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Related to Certification of the AP1000 Standard Plant Design

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ABSTRACT

This report supplements the final safety evaluation report (FSER) for the AP1000 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 design. The application for the AP1000 design was submitted on June 28, 2002, by Westinghouse Electric Corporation (Westinghouse) in accordance with Subpart B, "Standard Design Certifications," of Title 10 of the *Code of Federal Regulations* (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 AP1000 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 certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

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9. AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.1 Summary of Technical Information

Section 9.1.1, "New Fuel Storage" of the Westinghouse Electric Company, LLC (Westinghouse or the applicant) AP1000 design control document (DCD), Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 17, the applicant has proposed to make the following changes to Section 9.1.1 of the certified design:

1. New fuel rack design change. The basis for this change is documented in technical report (TR)-44, "New Fuel Storage Rack Structural/Seismic Analysis," APP-GW-GLR-026, Revision 0 of May 2006, and TR-44, Revision 1 of July 2008. TR-44, Revisions 0 and 1 described the design details and design-basis analyses for the new fuel racks. To be consistent with the design of the new fuel racks and the analyses presented in these TR revisions, the applicant proposed several changes throughout Section 9.1.1 of DCD Revision 17.

2. Fuel handling crane change. The applicant proposed to replace references to the fuel handling jib crane for the new-fuel handling crane. The basis for this change is addressed in TR-106, "AP1000 Licensing Design Changes for Mechanical System and Component Design Updates," APP-GW-GLN-106, Revision 1 of September 2007.

3. The applicant proposed conforming changes to DCD Section 9.1.1 related to the new fuel storage rack criticality analysis. The basis for this change is documented in TR-67, "New Fuel Storage Rack Criticality Analysis," APP-GW-GLR-030, May 29, 2006.

9.1.1.2 Evaluation

The staff reviewed all changes identified in the proposed amendment to the AP1000 DCD. The staff did not re-review descriptions and evaluations of the new fuel storage in the AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. All technical changes in the DCD are supported by information presented in the applicant's TRs.

The regulatory basis for AP1000 DCD, Section 9.1.1, is documented in NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," of September 2004. The staff has reviewed the proposed changes to DCD Section 9.1.1 against the applicable acceptance criteria of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Sections 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling" and 9.1.2, "New and Spent Fuel Storage." The following evaluations discuss the results of the staff's review.

The specific criterion that applies to the changes evaluated in this section is Title 10 of the *Code of Federal Regulations* (10 CFR) 52.63(a)(1)(vii), "Finality of standard design certifications," which concerns contribution to increased standardization of the certification information.

9.1.1.2.1 New Fuel Rack Design Change

9.1.1.2.1.1 Summary of Technical Information

The applicant proposed the following changes for the new fuel racks:

1. In DCD Section 9.1.1.1, deleted "... to supporting grid structures at the top and bottom elevations." and replaced it with "... to a thick base plate at the bottom elevation."

2. In DCD Section 9.1.1.1, deleted "... but may be braced as required to the pit wall structure."

3. In DCD Section 9.1.1.1, added reference to DCD Figure 9.1-1 for rack layout.

4. In DCD Section 9.1.1.1, replaced "new fuel handling crane" with "fuel handling machine."

5. In DCD Section 9.1.1.1, added the sentence "The stress analysis of the new fuel rack satisfies all of the applicable provisions in NRC Regulatory Guide (RG) 1.124, Revision 1 for components design by the linear elastic method."

6. In DCD Section 9.1.1.2, deleted "and laterally supported as required at the rack top by the pit wall structures."

7. In DCD Section 9.1.1.2.1, added the sentence "The new fuel storage rack array center-to-center spacing of nominally 10.9 inches provides a minimum separation between adjacent fuel assemblies sufficient with neutron absorbing material to maintain a subcritical array."

8. In DCD Section 9.1.1.2.1, deleted "racks are purchased equipment. The purchase specification for the new fuel storage racks will require the vendor to perform confirmatory dynamic and stress analyses."

9. In DCD Section 9.1.1.2.1, changed future tense "will be done by the combined operating license (COL) applicant" to present perfect tense "has been done by DC applicant".

10. In DCD Section 9.1.1.2.1, deleted "and is braced as required to the pit wall structures."

11. In DCD Section 9.1.1.2.1 and Section 9.1.1.3, changed the maximum uplift load from 907 kilograms (kg) (2000 pounds (lb)) to 1814 kg (4000 lb).

12. In DCD Section 9.1.1.2.1, added the weight of the fuel handling tool and the control rod assembly to the weight of the fuel assembly in the fuel drop analysis. This changed the total drop weight from 850 kg (1875 lb) to 919 kg (2027 lb).

13. In DCD Section 9.1.1.2.1, replaced "The crane and the attachment to the building structure ..." with "The fuel handling machine ..."

14. In DCD Section 9.1.1.3, changed future tense "will be done by combined operating license (COL) applicant" to present perfect tense "has been done by DC applicant."

15. In DCD Section 9.1.1.3, deleted "The new fuel storage rack is purchased equipment. The purchase specification for the new fuel storage rack requires a criticality analysis of the new fuel storage racks."

16. In DCD Section 9.1.1.3, identified that venting of the neutron absorbing material is "considered in the detailed design of the storage rack."

The staff confirmed that the changes to DCD Revision 17, Section 9.1.1 were consistent with the design changes identified in TR-44, Revisions 0 and 1. TR-44, Revision 3, was issued in May 2010, to update both the design and the design basis, and to address unresolved issues. TR-44, Revision 4, was issued in July 2010, to clarify the analyses conducted to address sliding of the racks. TR-44, Revision 5, was issued in August 2010, to incorporate the applicant's new position on accidental fuel assembly drops over the new fuel pit.

9.1.1.2.1.2 Evaluation

TR-44, Section 2.8.5, Revision 0, indicated that there were two postulated fuel drop scenarios over the new fuel pit, both from a height of 91.4 cm (36 in) above the top of the new fuel storage rack. In request for additional information (RAI)-TR44-01, the staff requested that the applicant describe the fuel handling operations that lead to the assumed 91.4 cm (36 in) drop height. The staff also issued RAI-TR44-02 through RAI-TR44-07, requesting specific information related to the accidental drop analysis.

In August 2010, the applicant submitted TR-44, Revision 5, in which the applicant deleted all reference to postulated new fuel assembly drop scenarios over the new fuel pit. The basis for this deletion was submitted in a revised response to RAI-TR44-01, dated August 13, 2010. This is further supported by the applicant's revised response to RAI-TR44-06, dated September 13, 2010. The applicant stated that the new fuel is moved from the rail car bay to a cell in the new fuel storage rack by the single-failure proof hoist. The same hoist then moves the fuel assembly from the cell in the new fuel storage rack to the new fuel elevator. This sequence is repeated for each new fuel assembly, 67 per outage, on an 18-month schedule. The applicant stated that the single-failure-proof hoist is designed to meet the requirements of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and is the only hoist capable of moving the new fuel above the operating floor. The applicant further stated that there are no safe shutdown systems or components currently housed in the new fuel pit or the resin transfer pump/valve room below it, and there are no criticality concerns for new fuel storage. The new fuel handling tool incorporates the same design features as the spent fuel handling tool (SFHT), which prevents inadvertent release of the new fuel assembly during handling operations. Based on the information in the revised RAI responses, the staff finds that the applicant has adequately justified that a new fuel assembly drop is unlikely and does not require analysis. RAI-TR44-02, RAI-TR44-03, RAI-TR44-04, RAI-TR44-05, and RAI-TR44-07 are no longer relevant, and are not discussed in this report.

As indicated in Table 2-3 of TR-44, one of the fuel handling accident loads that needs to be considered is uplift force on the rack caused by a postulated stuck fuel assembly. TR-44 Section 2.8.3 states: "An evaluation of a stuck fuel assembly, leading to an upward load of 2,000 lb has been performed. The results from the evaluation show that this is not a bounding condition as the local stresses do not exceed 2,500 psi." The staff determined that the information provided was not sufficient for the staff to reach a conclusion that this load had been adequately considered. In RAI-TR44-08, the staff requested the applicant to provide a detailed

description of the assumptions, the analyses conducted, the results obtained, and the basis for the conclusion that this is not a bounding condition.

In a letter dated June 7, 2007, the applicant stated that a nearly empty rack with one corner cell occupied is subject to an upward load of 907 kg (2000 lb), which is assumed to be caused by the fuel sticking while being removed. The ramification of the loading is two-fold:

1) The upward load creates a force and a moment at the base of the rack;
2) The loading induces a local tension in the cell wall.

The applicant attached a calculation documenting the maximum stress in the rack cell structure due to a postulated stuck fuel assembly. This local stress is well below the yield stress of the cell wall material (i.e., 207 MPa (30,000 psi)).

The basis for resolution of this RAI is similar to that of RAI-TR54-14, which is discussed in Section 9.1.2 of this report. The applicant re-defined the uplift force to be 1814 kg (4,000 lb), and showed that the induced stress is still well below the material yield stress. The applicant made appropriate changes in TR-44, Revision 1, and identified proposed changes to DCD Revision 17. The staff found this acceptable. However, the applicant's proposed changes to DCD Revision 17 were not completely implemented. During the June 2-3, 2010, regulatory audit, the staff asked the applicant to submit a supplemental RAI response, addressing the omission. In a letter dated July 20, 2010, the applicant submitted a supplemental response, explaining that the paragraph omitted from the DCD revision discussed the new fuel handling crane, which has been superseded. Therefore, it is no longer relevant. The staff finds this explanation acceptable. Therefore RAI-TR44-08 is resolved.

The staff noted that insufficient descriptive information was included in TR-44, Revision 0, to permit an adequate review of the structural/seismic analysis of the new fuel rack. In RAI-TR44-09, the staff requested the applicant to provide descriptive information, including plans and sections showing the new fuel rack and vault walls. All of the major features of the rack, including the cell walls, baseplate, pedestals, bearing pads, neutron absorber sheathing, any impact bars, welds connecting these parts, and any other elements in the load path of the rack should be shown on one or several sketches.

In a letter dated July 17, 2007, the applicant stated that TR-44 Figures TR-44-9.1 through TR-44-9.5 provide additional descriptive information on the new fuel rack and new fuel storage pit floor and walls. The staff confirmed that appropriate additions were made in TR-44, Revision 1. In a letter dated July 20, 2010, the applicant submitted a revised response to RAI-TR44-9, because several dimensions on drawings had been updated. The applicant stated that the most recent change to DCD Figure 9.1-1 (Sheet 1 of 2) is included in the response to RAI-TR44-17 Revision 3, which was also submitted in a letter dated July 20, 2010. The staff reviewed the change to DCD Figure 9.1-1 and found it acceptable. RAI-TR44-09 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text.

The staff noted that TR-44, Revision 0, did not provide sufficient data regarding the input loads used for the seismic analysis of the new fuel rack. The staff issued RAI-TR44-11, which reads as follows:

a. Floor response spectra (X, Y, and Z - vertical directions) at or near the elevation of the top of the fuel rack and near the bottom of the fuel rack or vault floor corresponding to the damping value used for the analysis.

b. An explanation of why the envelope of these two sets of spectra were not used.

c. An explanation of the range of soil and rock properties used in enveloping the seismic floor spectra. Given that the certified DCD is applicable to a hard rock site and the location of e are these range of soil/rock properties specified for confirmation by a future COL applicant?

d. For the synthetic time histories, plots of the three time histories, the cross correlation coefficients, the comparisons of the spectra from the synthetic time histories to the enveloped target response spectra, and the comparisons of the power spectral density plots to the target power spectral density function associated with the target response spectra.

e. Which time history was used (displacement, velocity, or acceleration)? Were all three directions input simultaneously? Was gravity included in the time history analysis?

In a letter dated May 3, 2007, the applicant provided the following response:

a) Floor response spectra (X, Y, and Z - vertical directions) near the elevation of the bottom of the new fuel storage vault corresponding to the damping value used for the analysis are provided in the attachment to this response. No floor response spectra are provided near or at the elevation of the top of the new fuel rack.

   The ASB99 floor response spectra (FRS) represent the enveloping response spectra for the auxiliary and shield building (ASB) at elevation 99 feet (ft) for a range of soil/rock condition. FRS of various soil/rock analyses were first enveloped for various locations of the ASB. All of the ASB locations at elevation 99 ft were then grouped and enveloped to develop the ASB99 floor response spectra.

b) It is probable that the floor response spectra will be revised for various reasons and that a revision to the new fuel storage rack structural/seismic analysis report (TR-44) will be required. The methodology for developing the spectra is described in TR-44-11 a, d and e responses.

c) The range of soil and rock conditions for which the seismic floor spectra applies is described in Westinghouse technical report (TR)-03, APP-GW-S2R-010, Revision 0, "Extension of NI Structures Seismic Analysis to Soil Sites."

d) The synthetic time histories, the response spectrum curves, and the power spectral density plots for the Auxiliary and Shielding Building (ASB) at Elevation 99 feet are provided, with this response. The cross correlation coefficients for

the three orthogonal components (East-West, North-South, and Vertical) of the ASB99 synthetic time histories are summarized in the table below.

Description	Cross Correlation Coefficient
East-West to North-South	-0.0414
East-West to Vertical	0.0088
North-South to Vertical	0.0536

e) Acceleration time histories are used as the input motion for the seismic analysis of the spent fuel racks. The acceleration input is defined by three orthogonal components, which are input and solved simultaneously. Gravity is also included in the time history analysis.

The staff found this RAI response acceptable because it adequately addressed all of the staff's questions. RAI-TR44-11 was initially resolved. However, subsequent to the initial resolution of this RAI, the applicant revised the seismic design loads twice. Therefore, during the June 2010 audit, the staff requested that the applicant update this RAI response to reflect the current seismic design loads for the new fuel rack. In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-TR44-11, updating the seismic design loads. The staff finds that the revised RAI response adequately describes the current seismic design loads for the new fuel rack. TR-44, Revision 3, includes the numerical results for the current design loads. Therefore, RAI-TR44-11 is resolved.

In RAI-TR44-12, the staff requested the applicant to address how the different impact stiffness values are determined for the fuel assembly-to-cell wall, rack-to-wall, and pedestal-to-bearing pad. In addition, since the impact forces can be greatly affected by the impact spring constant, the staff asked the applicant to address the sensitivity of the impact forces and rack responses to variations in these spring constants and whatever impact forces are imparted directly onto the cell walls or impact bars that are used?

In a letter dated July 5, 2007, the applicant stated that the impact stiffness values for the rack-to-wall and pedestal-to-bearing pad (concrete floor) are calculated as shown in Attachment 1 to the response. The fuel-to-cell wall impact stiffness is determined based on the solution for a simply supported circular plate under a concentrated load applied at its center, where the plate diameter is equal to the cell inner dimension and the plate thickness is equal to the cell wall thickness. The stiffness of the annular plate is then multiplied by the number of loaded storage cells for the new fuel storage rack, since the stored fuel assemblies are assumed to rattle in unison. A sensitivity study has not been performed specifically for the AP1000 new fuel rack to quantify the effect of variations in the impact stiffness values. However, sensitivity studies have been performed in the past for similar spent fuel rack applications submitted by HOLTEC, which employed the same method of computing the impact stiffness values, and the impact forces were found to be insensitive to small variations in the stiffness values provided that the integration time step was sufficiently small. There are no impact bars at the top of the new fuel storage rack. However, the new fuel storage rack is braced against the north and south walls of the new fuel storage pit by inserting stainless steel wedges in the interstitial space between the top of the new fuel storage rack and the pit opening.

The applicant subsequently changed the design of the new fuel rack to be free-standing in the new fuel pit. The steel wedges are no longer used. This is documented in TR-44, Revision 1. During the August 6-7, 2009 regulatory audit, the staff reviewed the updated calculations for the

free-standing new fuel rack. The calculations support the applicant's conclusion in TR-44, Revision 1, that there are no credible impacts between the new fuel rack and the fuel pit walls, for the free-standing configuration.

Therefore, RAI-TR44-12 was resolved. A new issue arose related to new fuel rack sliding and potential wall impact, after the resolution of RAI-TR44-12. This is addressed in the discussion of RAI-SRP9.1.2-SEB1-02 in this report.

Section 2.2.2.2 of TR-44 describes the modeling of a single rack. It indicates that the rack cellular structure elasticity is modeled by a 3-D beam having three translational and three rotational degrees-of-freedom (DOFs) at each end so that two-plane bending, tension/compression, and twist of the rack are accommodated. In RAI-TR44-14, the staff requested the applicant to explain why shear stiffness/deformation is not also included and to provide more detailed information about how the beam model of the rack was developed, considering that it is an assembly of many square-celled structures welded at discrete locations.

In a letter dated April 13, 2007, the applicant stated that shear deformation is included in the rack dynamic model. The beam model of the rack was developed based on the applicable codes, standards and specifications given in Section IV(2) of the Nuclear Regulatory Commission (NRC) guidance on spent fuel pool (SFP) modifications entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, which states that "Design ... may be performed based upon the AISC specification or Subsection NF requirements of Section III of the ASME Code for Class 3 component supports." The rack modeling technique is consistent with the linear support beam-element type members covered by these codes.

The basis for resolution of this RAI is similar to that of RAI-TR54-23, which is discussed in Section 9.1.2 of this report. The staff confirmed that appropriate changes were made in TR-44, Revision 1. Therefore, RAI-TR44-14 was resolved.

Section 2.2.2.2 of TR-44 refers to Figure 2-2 for the dynamic beam model of a single rack. The text and figure do not adequately describe the model. The staff issued RAI-TR44-15, which reads as follows:

a. Define what each series of nodal degrees-of-freedom (DOFs) correspond to (i.e., nodes 1, 2; P1, P2, ...; q4, q5, ..., 1*, 2*, ...). While some of these may be deduced by judgment, the report should clearly define all of these.

b. Explain whether there are five (5) nodes and four (4) beams along the rack beam model to coincide with the five (5) nodes and four (4) elements of the fuel assemblies.

In a letter dated June 7, 2007, the applicant provided the following response:

a. The attached table defines the nodal DOFs for the dynamic beam model of a single rack as depicted in Figure 2-2 of the technical report.

b. The rack cell structure is modeled as a single beam between two nodes, which are located at the top of the rack and at the baseplate elevation. This is consistent with HOLTEC's standard model for seismic analysis of spent fuel racks, which has been reviewed and approved by the NRC on numerous dockets. Although there is not a one-to-one correspondence between beam nodes and fuel assembly nodes, fuel-to-cell

> wall impact loads, which can occur at elevation 0, 0.25H, 0.5H, 0.75H, and H (where H is the height of the cell structure), are properly transmitted to the rack beam in accordance with the methodology outlined in Reference 12 in COLA technical report APP-GW-GLR-026 Revision 0.

The basis for resolution of this RAI is similar to that of RAI-TR54-24, which is discussed in Section 9.1.2 of this report. The staff confirmed that appropriate changes were made in TR-44, Revision 1. Therefore, RAI-TR44-15 was resolved.

In RAI-TR44-16, the staff requested the applicant to explain whether only a full new fuel rack is considered in the simulation, or if several scenarios are considered; i.e., different fill ratios, from empty to full; and provide the technical justification if only a full rack is considered.

In a letter dated June 7, 2007, the applicant stated that the new fuel rack is assumed to be fully loaded with maximum weight fuel assemblies in all three simulations. This scenario bounds any partially loaded configuration since it: (1) maximizes the vertical compression and lateral friction loads on the support pedestals; and (2) produces the maximum rack displacements and fuel-to-cell wall impacts. The displacements are larger for a fully loaded rack, as opposed to a partially filled rack, because the dynamic model conservatively assumes that all stored fuel assemblies rattle in unison. Hence, the momentum transferred between the rattling fuel mass and the new fuel rack is the maximum for a fully loaded rack. For a partially filled rack, the decrease in rattling fuel mass outstrips the destabilizing effect of an eccentric fuel loading pattern.

The staff determined that a quantitative evaluation of this issue would most likely be required. During the August 2009 audit, the staff reviewed Draft Revision 3 to HOLTEC Calculation HI-2063492, which includes analysis of a partial loading case. The staff requested that the applicant revise its RAI response to reflect this additional loading case, and also to include these results in the next revision of TR-44.

The applicant formally submitted Revision 3 to TR-44 in May 2010, which includes the results of the partial loading case. At the June 2010 audit, the staff and applicant discussed these results, which indicate that partial loading generates the limiting conditions for certain loading cases. The staff noted that the applicant's current RAI response indicates that the partial loading case is not controlling, and that the DCD does not identify that a partial loading was considered. In a letter dated July 20, 2010, the applicant submitted a revised response to RAI-TR44-16, correcting the statement about the partial loading case, and also proposing a DCD change. The staff finds the response acceptable because it adequately addresses the inconsistency identified 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 staff issued RAI-TR44-17 which reads as follows:

> What are the gaps and tolerances for the gaps between the fuel assembly and cell wall, and between the rack and vault wall? What are the assumed initial locations of the various components (fuel assemblies and rack) and what is the technical basis for this assumption? Were any studies done for different initial conditions (considering tolerances); if not, explain why it was not necessary. Are there requirements in the DCD to ensure that the assumed gaps (considering tolerances) are maintained throughout the operating license period?

In a letter dated July 17, 2007, the applicant stated that all gaps between fuel assemblies and cell walls and between the rack and vault walls are set to match the nominal gaps provided on the layout drawing. The applicant attached a table summarizing the gap information used in the dynamic analyses. The applicant also stated that the new fuel storage rack is braced against the north and south walls of the new fuel storage pit by inserting stainless steel wedges in the interstitial space between the top of the new fuel storage rack and the new fuel storage pit opening. Fuel is assumed centrally located in the cell. This is conservative, since minimizing the gap on one or two walls will generally produce a larger hydrodynamic coupling effect. Some numerical studies were done on other rack projects; the results generally showed a small influence on results. A larger influence occurs if the gaps are assumed to be displacement dependent, rather than always being held constant at their initial value. The neglect of this effect is conservative.

The applicant further stated that, once the new fuel rack is installed, the "as-built" gaps are reconciled with the gaps initially used for analysis by evaluation of the numerical results and the predicted motions. The new fuel rack will be positioned in the new fuel storage pit in accordance with the gap information provided in the table attached to the July 17, 2007, letter. The only way the gaps would change over time would be by the action of a seismic event. COL applicants will have a procedure in place to address measurement of the post design-basis seismic event gaps, and to evaluate the acceptability of the configuration or to take appropriate corrective actions. The applicant proposed that the following statement be added to TR-44:

> Per DCD Subsection 3.7.5.2, Combined License applicants will prepare site-specific procedures for activities following an earthquake. These procedures will be used to accurately determine both the response spectrum and cumulative absolute velocity of the recorded earthquake ground motion from the seismic instrumentation system. An activity will be to address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit and to take appropriate corrective actions.

During the August 2009 audit, the staff discussed the need to revise the HOLTEC drawing depicting the design gaps and tolerances, to reflect the change in the new fuel rack design to be free-standing.

During the June 2010 audit, the staff confirmed the HOLTEC drawings have been updated; and also requested the applicant to clarify the dimension and gap information for the new fuel racks in three related RAIs (RAI-TR44-09, RAI-TR44-017, and RAI-TR44-25), to be consistent with the current design basis. The staff noted that DCD Figure 9.1-1 needed to be updated. In a letter dated July 20, 2010, the applicant submitted a revised response to RAI-TR44-17, updating dimension and gap information. The applicant also proposed a change to DCD Figure 9.1-1. The staff finds the response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 2.3.4.3 of TR-44, fourth bullet, develops the faulted (Level D) allowable maximum weld stress for the weld material. In RAI-TR44-22, the staff asked the applicant whether an allowable maximum weld stress based on the base metal was also developed. The staff noted that normally welds are checked for both weld material and base metal, as was done for Levels A and B in TR-44 Section 2.3.4.1.

In a letter dated June 7, 2007, the applicant provided a response that is essentially the same as its response to RAI-TR54-33 on the same topic. The detailed review is discussed in

Section 9.1.2 of this report. Based on that discussion, the staff found this RAI response acceptable. Therefore, RAI-TR44-22 was resolved.

Section 2.3.5 of TR-44 discusses dimensionless stress factors. It states that "R1 is the ratio of direct tensile or compressive stress on a net section to its allowable value (note pedestals only resist compression)." In RAI-TR44-23, the staff requested the applicant to explain why the pedestals only resist compression, since horizontal forces are also generated due to friction during a seismic event. These forces could be quite high and also would introduce shear and moments into the pedestal and rack structure.

In a letter dated July 17, 2007, the applicant stated that Section 2.3.5 of TR-44 defines seven stress factors (R1 through R7), which correspond to the American Society of Mechanical Engineers (ASME) Code Section III, Subsection NF stress limits for Class 3 components. R1 is defined as the ratio of direct tensile or compressive stress on a net section to its allowable value. Since the new fuel rack is freestanding, the net cross section of the support pedestals can only be subjected to direct compressive stress. The applicant further stated that horizontal forces are generated due to friction between the support pedestals and the floor and that these forces produce shear and bending stresses in the pedestals. The shear and bending stresses in the support pedestals, as well as the combined compression and bending stress, are measured by the other six stress factors (i.e., R2 through R7), which are defined in Section 2.3.5 of TR-44. The staff reviewed the RAI response and found it acceptable, because the applicant clarified that shear force and moment due to friction have been calculated and evaluated against applicable code limits. Therefore, RAI-TR44-23 was resolved.

Some of the information provided in Section 2.8.2 (Rack Structural Evaluation) and Tables 2-6 through 2-14 (stress results) of TR-44 is not clear. The staff issued RAI-TR44-24, which reads as follows:

a. Section 2.8.2.1, 2nd paragraph, indicates that the tables also report the stress factors for the AP1000 new fuel storage rack cellular cross section just above and below the baseplate. This implies that the fuel cells continue below the baseplate. Explain.

b. The same paragraph refers to "pedestal five in the first sheet of the summary tables for each simulation (that is, 9.M.0 where M stands for run number)." Explain what this means since the tables do not reflect this terminology.

c. The same paragraph refers to "ensures that the overall structural criteria set forth in Subsection 2.2.3 are met." Structural criteria are not presented in Subsection 2.2.3.

d. Section 2.8.2.2 a. refers to a stress factor of 2.1516, which it states is given in the tables. However, no such stress factor is given, please explain. Also, are all cells welded to the baseplate on all four sides?

e. Section 2.8.2.2 b. indicates that a separate finite element model is used to check the baseplate to pedestal welds. Provide a short description of the model, computer code, loading, and location of the maximum tabulated stress in the weld referred to in Table 2-12.

f. Section 2.8.2.2 c. indicates that for calculation of cell welds, the fuel assemblies in adjacent cells are conservatively calculated by assuming that the fuel assemblies in adjacent cells are moving out of phase with one another. It then states that cell to cell

weld calculations are based on the maximum stress factor from all runs. However, elsewhere in the report, it was stated that all of the fuel assemblies in the simulation are assumed to vibrate in phase. Provide more information to explain this.

g. Section 2.8.2.3 refers to Tables 2-6 through 2-13 for limiting thread stresses under faulted conditions for every pedestal. These tables do not seem to apply to pedestal thread shear stress. Therefore, clarify or correct this information.

h. For Table 2-6, Results Summary, please identify what rack component/element applies to each of the column headings (i.e., Max Stress Factor, Max. Shear Load, Max Fuel to Cell Wall Impact). Similarly, for Tables 2-11, 2-13, and 2-14, identify what rack component/element the table applies to.

i. Why is Table 2-14 labeled "Allowable Shear Stress for Level D"? This is inconsistent with the other tables. Explain.

In a letter dated July 17, 2007, the applicant provided the following response:

(a) The fuel cells do not continue below the baseplate. Stress factors are computed just above the baseplate, where the fuel cells are welded to the baseplate, and just below the baseplate where the support pedestals are welded. Section 2.8.2.1 (2nd paragraph, 2nd sentence) will be revised as follows:

> "The tables also report the stress factors for the AP1000 new fuel storage rack cellular cross section just above the baseplate."

(b) The computer code DYNAPOST, which is listed in Table 2-15, computes the stress factors for the four support pedestals and for the cellular structure just above the baseplate based on the time history analysis results. For convenience, these five locations are identified as pedestal numbers 1 through 5 in the DYNAPOST output tables, which are not included in TR-44. Therefore, the sentence, "The locations above the base plate ... are referred to as pedestal five in the first sheet of the summary tables for each simulation (that is, 9.M.0 where M stands for run number)" is not relevant to the report and will be deleted.

(c) The reference to Subsection 2.2.3 is a typographical error. The correct reference is Subsection 2.3.3.

(d) The factor of 2.1516 is not provided in the tables as stated in text. Section 2.8.2.2 a. (2nd paragraph) will be revised as follows:

> "Weld stresses are determined through the use of a simple conversion (ratio) factor (based on area ratios) applied to the corresponding stress factor in the adjacent rack material. This conversion factor is developed from the differences in base material thickness and length versus weld throat dimension and length." All fuel cells are welded to the baseplate on all four sides.

(e) The finite element code ANSYS is used to resolve the tension and compression stresses in the pedestal weld due to the combined effects of a vertical compressive load in the pedestal and a bending moment caused by pedestal

friction. The compression interface between the baseplate and the pedestal is modeled using contact elements. The perimeter nodes on the pedestal are connected to the baseplate by spring elements in order to simulate tension in the weld. The maximum instantaneous friction force on a single pedestal from the rack seismic analysis is conservatively applied to the finite element model in the horizontal x- and y-directions simultaneously, along with the concurrent vertical load, at the appropriate offset location. The perimeter nodes on the pedestal are restrained to move only in the vertical direction so that the spring elements only resist bending. The limiting ANSYS results are combined with the maximum horizontal shear loads to obtain the maximum weld stress. The maximum weld stress reported in Table 2-12 occurs at the corner of the pedestal where the tensile stress in the weld due to bending is maximum.

(f) All stored fuel assemblies within a rack are assumed to rattle in phase for the seismic analysis of the new fuel rack using the HOLTEC proprietary computer code MR216 (a.k.a. DYNARACK). This analysis yields the maximum impact force between a single fuel assembly and the surrounding cell walls. When evaluating the weld connection between adjacent storage cells, the maximum fuel-to-cell impact force from the dynamic analysis is conservatively multiplied by a factor of 2 to consider out-of-phase fuel rattling.

(g) The reference to "Tables 2-6 through 2-13" in Section 2.8.2.3 is incorrect. The first sentence in Section 2.8.2.3 should be revised as follows: "Table 2-14 provides the limiting thread stress under faulted conditions."

(h) In Table 2-6, the "Max. Stress Factor" column applies to the rack cell structure. The "Max. Vertical Load" and "Max. Shear Load" columns apply to a single rack pedestal. The "Max. Fuel-to-Cell Wall Impact" column provides the maximum impact force between a single fuel assembly and the surrounding cell wall at any of the five rattling fuel mass elevations (refer to Figure 2-5 of the report). Table 2-11 applies to the base metal adjacent to the baseplate to cell welds. Table 2-13 provides the shear stress in the cell to cell welds as well as the adjacent base metal. Table 2-14 applies to the pedestal internal threads.

(i) Table 2-14 should be labeled "Pedestal Thread Shear Stress" instead of "Allowable Shear Stress for Level D." The allowable stresses reported in Tables 2-10 through 2-14 are Level D stress limits since the design basis ASB99 earthquake is a faulted condition (Level D).

The basis for resolution of this RAI is similar to that of RAI-TR54-36, which is discussed in Section 9.1.2 of this report. Based on that discussion and confirmation that the proposed changes were made in TR-44, Revision 1, the staff found this RAI response acceptable. Therefore, RAI-TR44-24 was resolved.

In the markup of the DCD, provided in Section 5 of TR-44, Revision 0, DCD Figure 9.1-1, new fuel storage rack, is identified for deletion. In RAI-TR44-25, the staff requested the applicant to explain why this figure was marked for deletion.

In a letter dated April 13, 2007, the applicant stated, "We are in agreement Revision 16 of the DCD will have a revised Figure 9.1-1 New Fuel Rack Layout. This figure will show the new fuel rack configuration in plan and elevation views identifying significant features and dimensions".

The staff reviewed the RAI response and concluded that review of DCD Revision 16 would be necessary to determine if sufficient information is included in the revised Figure 9.1-1.

In TR-44, Revision 1, the applicant changed the new fuel rack design to be free-standing, which affects DCD Figure 9.1-1. This was discussed with the applicant and HOLTEC at the August 2009 audit. The applicant agreed to revise the figure as necessary to reflect the design change.

In a letter dated July 20, 2010, the applicant submitted a revised response to RAI-TR44-25, referencing its revised response to RAI-TR44-17, also submitted by letter dated July 20, 2010, for the updated dimension and gap information. RAI-TR44-25 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff noted that computer code MR216 (a.k.a. DYNARACK), as well as the other computer analysis codes used for the rack analyses, should have complete validation documentation, available for review during an audit. To streamline such an audit, in RAI-TR44-26, the staff inquired whether any of the computer codes have been previously reviewed and approved by the staff in other licensing reviews.

In a letter dated June 7, 2007, the applicant stated that computer analysis codes used to perform the seismic analysis of the new and spent fuel racks have been validated in accordance with HOLTEC's 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," quality assurance program. The validation documentation will be available for review during the audit. The validation documentation for the computer code MR216 has been previously submitted by HOLTEC International to the NRC staff for review and approval several times. Most recently it was reviewed by the NRC in 1998 in Docket 50-382 for the Waterford 3 Steam Electric Station.

The basis for resolution of this RAI is similar to that of RAI-TR54-39, which is discussed in Section 9.1.2 of this report. Based on that discussion, the staff found this RAI response acceptable. Therefore, RAI-TR44-26 was resolved.

In its review of TR-44, the staff did not identify any information related to inservice inspection of the new fuel rack. In RAI-TR44-27, the staff requested the applicant explain what provisions are provided for performance of inservice inspection of the rack, in accordance with 10 CFR 50.55a, "Codes and standards" and/or 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," as applicable.

In a letter dated June 7, 2007, the applicant stated that the new fuel rack is passive in nature. There are no moving parts on the new fuel rack, and it does not require any instrumentation. Therefore, there is no compelling need to perform inservice examination of the new fuel rack. Nonetheless, the new fuel rack can be accessed from above by way of an empty storage cell location(s) to enable the performance of inservice examination, as mandated by 10 CFR 50.55a(g)(3) for ASME Code Class 3 component supports. At the base of each storage cell (except at the four designated lifting locations), there is a 15.24 cm (6 in) diameter thru-hole in the baseplate, which provides access below the baseplate. The new fuel rack contains new fuel only during a short period prior to refueling. When it does not have new fuel, it could be lifted from the new fuel storage pit for inspection.

In summary, the new fuel rack is designed to provide access to surfaces that may come in contact with new fuel assemblies and to the support pedestals beneath the baseplate to support inservice examinations as needed.

From the information provided by the applicant, the staff concluded that there is ample access to the new fuel rack for inservice inspection. Therefore, RAI-TR44-27 was resolved.

The treatment of the new fuel storage rack as a safety class/seismic Category I component appears to represent a departure from past practice in the nuclear power industry. The draft update to RG 1.29, "Seismic Design Classification," (DG-1156) does not identify new fuel storage racks as seismic Category I. In RAI-TR44-28, the staff requested the applicant to: (1) describe the technical basis for treating the new fuel storage rack as a safety class/seismic Category I component; and (2) explain how the safety significance of the AP1000 new fuel storage rack differs from prior nuclear power plant designs.

In a letter dated June 7, 2007, the applicant provided the following response:

1) We understand that both Regulatory Guide 1.29, Revision 3 and draft update to RG 1.29 (DG-1 156) do not identify new fuel storage racks as seismic Category I. However, the Westinghouse decided that all racks in the AP1000 plant would be seismic Category I. Holtec has designed and fabricated new fuel storage racks to seismic Category I. There is no additional analysis or fabrication cost to have the new fuel storage rack as seismic Category I.

2) The safety significance of the AP1000 new fuel storage rack does not differ from prior nuclear power plant designs. It is both the applicant's and HOLTEC's position that the form, fit and function of the AP1000 new fuel storage rack is the same as new fuel racks in operating PWRs.

On the basis that the applicant has invoked more stringent seismic design requirements than RG 1.29, the staff found that its technical approach is acceptable. Therefore, RAI-TR44-28 was resolved.

Section I, "Introduction," was revised in TR-44, Revision 1 to add: "Per DCD Subsection 3.7.5.2, COL applicants will prepare site-specific procedures for activities following an earthquake. These procedures will be used to accurately determine both the response spectrum and cumulative absolute velocity of the recorded earthquake ground motion from the seismic instrumentation system. An activity will be to address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit and to take appropriate corrective actions."

The staff noted that DCD Section 3.7.5.2 does not discuss the need for COL applicants to prepare site-specific procedures for checking the gaps between the new fuel rack and walls of the new fuel storage pit following an earthquake. In RAI-SRP9.1.2-SEB1-01, the staff requested the applicant to explain how this requirement is conveyed to the COL applicants.

In its response dated February 24, 2009, the applicant proposed a revision to Section 3.7.5.2 of the DCD, requiring COL applications to include in their post-earthquake procedure a requirement to check the gaps between the new fuel rack and the new fuel pit walls and to take appropriate actions to restore the design gaps. Following the staff's August 2009 audit, the applicant submitted a revised response on August 31, 2009. The staff found this revised RAI

response acceptable, because it more clearly specifies the actions to be taken following an earthquake. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In its review of TR-44, Revision 1, the staff noted that Section 2.8.1.4 "Impact Loads" was not revised, even though shims between the new fuel rack and the fuel pit wall have been deleted from the design. Quoting from TR-44, Revision 1, Section 2.8.1.4, "The maximum impact load from the set of shims that close the north-south gaps at the top of the rack is summarized in Table 2-8." The staff also noted that the maximum rack-to-wall impact load in Table 2-8 increased from 50,802 kg (112,000 lb) in TR-44, Revision 0, to 69,853 kg (154,000 lb) in TR-44, Revision 1.

In RAI-SRP9.1.2-SEB1-02, the staff requested the applicant explain why the impact load increased, and describe how the design of the new fuel rack and the new fuel pit wall were evaluated for the significant increase (35 percent) in the impact load, in addition to other concurrent loadings.

In a letter dated April 1, 2009, the applicant submitted its response to RAI-SRP9.1.2-SEB1-02. The applicant stated that all references to shims will be deleted from TR-44. The applicant described the basis for the increased impact force, and clarified that the impact is the base plate against the wall, not the top of the rack against the wall. The applicant also stated that this impact only occurs for an unrealistically low coefficient of friction (COF) (0.2) between two dry steel surfaces. At realistic values (>0.5), the new fuel rack does not slide, and there is no rack-to-wall impact. The staff concluded that the detailed calculation(s) leading to the conclusion that there are no credible impacts between the new fuel rack and the fuel pit walls needed to be audited. At the August 2009 audit, the staff reviewed the detailed calculation, and found it acceptable to support the conclusion of no impact. The applicant agreed to revise TR-44 and Calculation APP-FS01-S3C-001 (HOLTEC Calculation HI-2063492), to clarify that there are no credible impacts between the new fuel rack and the new fuel pit walls, and to define the technical basis for this conclusion.

In May 2010, the applicant informed the staff that the COF could be as low as 0.24 because the actual surface condition is now steel on concrete, not steel on steel as it was previously. The applicant performed new calculations and evaluation of sliding and impact using a COF of 0.24. At the June 2010 audit, the applicant presented the results of the analysis for a 0.24 coefficient to the staff. TR-44, Revision 3, submitted just prior to the audit, includes the results of the new calculations. The staff reviewed the presentation material and related section of TR-44, Revision 3. The staff pointed out several apparent inconsistencies in the results presented, possibly indicative of an incorrect COF. Upon examination, HOLTEC confirmed that an incorrect COF had been used in one of the cases analyzed. During the audit, HOLTEC corrected the input file, re-executed the analysis, and presented the corrected results. The staff reviewed the corrected results, and found them acceptable.

The new results, for a 0.24 COF, show considerable sliding of the new fuel rack, but impact is avoided by increasing the minimum gap between the new fuel rack and the new fuel pit walls. The applicant agreed to revise the response to RAI-SRP9.1.2-SEB1-02 to describe the updated evaluation of sliding and impact, and also to identify necessary changes to TR-44.

In a letter dated July 30, 2010, the applicant submitted a revised response to RAI-SRP9.1.2-SEB1-02. The applicant stated that it had re-evaluated the range of appropriate friction values, to ensure that the interface between the new fuel storage rack and the new fuel

pit floor is accurately and conservatively represented. The applicant concluded that the appropriate credible lower-bound COF is presented in Run Number 5 (COF=0.24). Additionally, Run Number 1 (COF=0.2) has been demonstrated to be non-credible, as noted above. It will be maintained in the supporting calculations, but for clarity it will be eliminated from the tables in the next revision of TR-44.

The applicant also attached TR-44, Revision 4 (July 2010) to the RAI response. Revision 4 clarifies what run numbers apply to the design basis and provides the detailed results for the credible lower-bound COF (COF=0.24). The report eliminates reference to the non-credible case for Run Number 1 (COF=0.2).

The staff reviewed TR-44, Revision 4, and confirmed that it includes appropriate new information related to Run Number 5 and that Run Number 1 has been deleted. The staff finds this acceptable to properly document the design basis for the new fuel rack. Therefore, RAI-SRP9.1.2-SEB1-02 is resolved.

Section 2.8.4.1, "Cell Wall Buckling Evaluation," was revised in TR-44 Revision 1. The buckling equation and assumed rectangular plate boundary conditions were changed. The rectangular flat plate model representing the lower cell wall region was changed to clamped on all four edges. In the Revision 0 calculation, simple support was assumed. The staff determined that only one edge can truly be treated as clamped, and the other three edges can rotate somewhat due to the flexibility of the adjacent sections.

The staff issued RAI-SRP9.1.2-SEB1-03, which reads as follows:

> Section 2.8.4.1, "Cell Wall Buckling Evaluation" was revised in TR-44 Revision 1. A different buckling equation and different boundary conditions are indicated. The rectangular flat plate model representing the lower cell wall region is now assumed to be clamped on all 4 edges, Considering that only 1 edge can truly be treated as clamped, and the other 3 edges can rotate somewhat due to the flexibility of the adjacent sections, the staff requests Westinghouse to provide the technical basis for changing the boundary conditions to clamped on all 4 edges. Also, identify the minimum acceptable factor of safety and the technical basis for its selection.
>
> The staff notes that for K = 7.23, the revised σ_{cr} should be 15,600 psi, not 13,100 psi. Also, there is a typographical error: "31,100" should have been "13,100." The staff requests Westinghouse to correct the text of Section 2.8.4.1 accordingly.
>
> The staff also requests Westinghouse to identify the factor of safety based on the Rev 0 estimate of σ_{cr} = 7,090 psi.

In a letter dated April 14, 2009, the applicant submitted its response to RAI-SRP9.1.2-SEB1-03, in which it provided its technical basis for the revised calculation of buckling for the cell wall. The staff reviewed the response and determined that the information provided was insufficient, and that a significantly expanded technical basis would be needed before the staff could accept the cell wall buckling calculation. At the August 2009 audit, HOLTEC informed the staff that it was conducting a detailed nonlinear analysis of the bottom of the rack for vertical compressive load, and presented the ANSYS computer model and preliminary results. The staff found this to be a considerable analytical improvement. During the June 2010 audit, the staff reviewed

HOLTEC's final results of the ANSYS buckling evaluation of the cell walls, at the base of the new fuel rack. The calculation shows that a 1.5 factor of safety, in accordance with the acceptance criterion in ASME Code Section III, Subsection NF, has been achieved. The staff finds the analytical method used and the results obtained to be acceptable, based on its detailed review of HOLTEC's calculation. Therefore, RAI-SRP9.1.2-SEB1-03 is resolved.

9.1.1.2.1.3 Conclusion

The staff has conducted a detailed review of TR-44, which addresses DCD Revision 15 COL Information Item 9.1-1: "Perform a confirmatory structural dynamic and stress analysis for the new fuel rack, as described in AP1000 DCD Subsection 9.1.1.2.1. This includes the structural adequacy of the proposed AP1000 new fuel storage rack under postulated loading conditions and effects on the structure described in Subsection 3.8.4." The staff finds the new fuel rack design, as described in TR-44, Revision 5, to be acceptable. On the basis of its review, the staff concludes that the substance of the COL information item is completely addressed by TR-44, and that this COL information item is no longer needed.

In its previous evaluations of AP1000 DCD Section 9.1.1, the staff identified acceptance criteria based on the design's meeting relevant requirements in 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 2, "Design Bases for Protection Against Natural Phenomena," and in GDC 4, "Environmental and Dynamic Effects Design Bases." The above evaluation concludes that the new fuel rack design meets these 10 CFR Part 50 requirements.

9.1.1.2.2 Fuel Handling Crane Change

The applicant proposed the following changes for the fuel handling crane:

1. Deleted references to the fuel handling jib crane and replaced them with references to the new-fuel handling crane in DCD Sections 9.1.1.1; 9.1.1.2; 9.1.1.2.1; and 9.1.1.3.

2. The capacity of the fuel handling crane was limited to 907 kg (2000 lb), the applicant is now proposing that the capacity of the fuel handling crane be limited to lifting a fuel assembly, control rod assembly, and handling tool.

3. The uplift capability of the new-fuel handling crane was increased from 907 kg (2000 lb) to 919 kg (2027 lb) in DCD Section 9.1.1.3.

However, in response to RAI-SRP9.1.4-SBPB-01, the applicant stated in a letter dated June 26, 2008, that the function of moving new fuel will be transferred to the fuel handling machine (FHM) and that the new-fuel handling crane will be eliminated.

The evaluation of this change is reviewed in Section 9.1.4 of this report. The staff determined that the changes made to DCD Section 9.1.1 are conforming changes that do not impact the staff's safety evaluation of DCD Section 9.1.1. Therefore, the staff finds the proposed change acceptable.

9.1.1.2.3 New Fuel Criticality Analysis

9.1.1.2.3.1 Summary of Technical Information

In the certified DCD Revision 15, Section 9.1.1, "New Fuel Storage," it is stated in Section 9.1.6.2 that the COL applicant is responsible for a confirmatory criticality analysis for the

new fuel rack, as described in Section 9.1.1.3. This is COL Information Item 9.1-2. In DCD Revision 17, the applicant proposed to perform the confirmatory criticality analysis so that COL action is no longer necessary. DCD Section 9.1.1.3 is revised to reflect that the criticality analysis is now complete and Section 9.1.6.2 is revised to state that the COL information requested in this section has been completely addressed in TR-67, and the applicable changes are incorporated into the DCD. The applicant stated that no additional work is required by the COL applicant. The technical details of the criticality analysis for the AP1000 new fuel storage design is presented in TR-67. This report provides the technical support for the changes found in Section 9.1.1 of the DCD. The staff's review of the criticality analysis of AP1000 new fuel storage includes DCD Revision 17 Section 9.1.1 and TR-67.

The staff based its review of the AP1000 new fuel storage on the information in the DCD and the TRs referenced by the applicant. The review was limited in scope to the changes to the new fuel storage criticality analysis of DCD Revision 15 (NUREG-1793) as presented in Revision 17. The staff conducted its review of the criticality analysis of the new fuel storage in accordance with the guidelines provided in Section 9.1.1 of NUREG-0800.

10 CFR Part 50, Appendix A, provides a list of the minimum design requirements for nuclear power plants. According to GDC 62, "Prevention of Criticality in Fuel Storage and Handling," the licensee must limit the potential for criticality in the fuel handling and storage system by physical systems or processes.

The regulations regarding criticality for the AP1000 new fuel storage rack are as follows:

- 10 CFR 50.68(B)(2), "Criticality accident requirements," states the maximum K_{eff} value, including all biases and uncertainties, must not exceed 0.95, at a 95 percent probability and a 95 percent confidence level, with full density unborated water.

- 10 CFR 50.68(B)(3) states the maximum K_{eff} value, including all biases and uncertainties, must not exceed 0.98, at a 95 percent probability and a 95 percent confidence level, with optimum moderation and full reflection conditions.

- 10 CFR 50.68(B)(7) states the maximum enrichment of fresh fuel assemblies must be less than or equal to 5.0 weight-percent (w%) U-235.

9.1.1.2.3.2 Evaluation

The criticality analysis of new fuel storage racks for the AP1000 design is presented in TR-67.

Methodology

The analysis methodology uses SCALE-PC, a personal computer version of the SCALE-4.4a code system with the updated SCALE-4.4a version of the 44 group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross-section library.

SCALE-PC, used in both the benchmarking and the fuel assembly storage configurations, includes the control module CSA25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. The SCALE system was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on personal computers.

Validation of SCALE-PC for fuel storage rack analyses is based on the analysis of representative critical experiments from two experimental programs: the Babcock & Wilcox (B&W) experiments in support of Close Proximity Storage of Power Reactor Fuel and the Pacific Northwest Laboratory Program in support of the design of Fuel Shipping and Storage Configurations. DCD Revision 15 and "Criticality Safety Criteria" (TANS Volume 35, page 278 dated 1980) were useful in updating pertinent experimental data for the Pacific Northwest Laboratory experiments. The validation of SCALE-PC was limited to the 44 group library provided with the SCALE-PC version 4.4a package (Westinghouse calculation CN-CRIT-206).

The approach used for the determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of "Criticality Safety Criteria (TANS Volume 35, page 278 dated 1980) and DCD Revision 17. For a given KENO-calculated value of K_{eff} and associated one sigma uncertainty, the magnitude of $k_{95/95}$ is computed. By this definition, there is a 95 percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than $k_{95/95}$.

Assumptions

The assumptions used for the AP1000 new fuel storage rack are as follows:

- The Westinghouse AP1000 17x17 fuel assembly was modeled as the design basis fuel assembly with an enrichment equal to 5.0 w% U-235.

- Storage cell material and Metamic® poison material were modeled with a length of 426 cm (168 in). The storage cell material and Metamic® poison above and below the active fuel length were conservatively omitted.

- Fresh fuel assemblies were conservatively modeled with a UO_2 density of 10.686 gram per cubic centimeters (g/cm^3) (97.5 percent of theoretical density). This translates into a pellet density equal to 98.6 percent of theoretical density with a 1.1 percent dishing (void) fraction.

- All fresh fuel assemblies were conservatively modeled as containing solid right cylindrical pellets and uniformly enriched over the entire length of the fuel stack height. This conservative assumption bounds fuel assembly designs that incorporate lower enrichment blanket or annular pellets.

- All fuel assemblies were conservatively modeled as containing no burnable absorber material.

Design Input

The fuel assembly modeled in this analysis is the Westinghouse AP1000 17xl7 fuel assembly present in Figure 2-1 of TR-67. The bottom elevation of the Metamic® poison panel (436.9 cm (172 in) long) conservatively covers the active fuel length (427 cm (168 in)) with a 10.16 cm (4 in) overlap (5.08 cm (2 in) overlap on each end of the active fuel). Design data related to the AP1000 new fuel storage rack were obtained from HOLTEC. In addition, HOLTEC supplied Westinghouse a pressurized-water reactor (PWR) rack data sheet, which provides detailed new fuel storage rack design information (HOLTEC International Letter Number 1540001.doc, "PWR Rack Data Sheet," Revision 4, April 27, 2006).

Westinghouse general arrangement and concrete outline drawings were used to determine the new fuel storage vault geometry. A plan view of the AP1000 new fuel storage rack is shown in Figure 2-2 of TR-67. The AP1000 new fuel storage rack is located inside a concrete room (vault) in the AP1000 Auxiliary Building. The AP1000 new fuel storage rack is centered inside the vault and is an 8x9 array of storage cells, which provides 72 total storage locations. A hatch lid is provided for the vault for security, and for foreign material exclusion (FME).

The individual storage cells of the AP1000 new fuel storage rack are centered on a nominal pitch of 27.7 cm (10.9 in). Each storage cell consists of an inner stainless steel box, which has a nominal inside dimension of 22.4 cm (8.8 in) and is 0.19 cm (0.075 in) thick. Metamic® panels are attached to the outside surfaces of all storage cells except for the outside cell walls directly facing the North and South walls of the vault. No poison is required on these outside cell faces since there is just a small amount of space between the rack and storage vault concrete. However, poison is required on the outside cell faces in the East and West directions (see Figure 2-2 in TR-67) to mitigate the effects of an inadvertent placement of a fuel assembly outside of the rack, but within the vault on these two sides. Each Metamic® poison panel is held in place and is centered on the surface of the stainless steel box by an outer stainless steel sheathing panel. There is a small void space between the sheathing and the Metamic® panel. The dimensions of the Metamic® poison panel are 19.06 cm (7.5 in) in width by 0.27 cm (0.106 in) in thickness. The sheathing panels are 0.089 cm (0.035 in) in thickness.

Each storage cell is nominally 506.7 cm (199.5 in) long, and it rests on top of a base plate whose top is 12.7 cm (5 in) above the concrete floor. As stated above, each Metamic® poison panel is 436.9 cm (172 in) long overlapping the 426.7 cm (168 in) active fuel length. The Metamic® poison material is nominally a mixture of 31 w% B_4C and 69 w% aluminum.

KENO Model and Assumptions

The KENO V.a model is a three-dimensional representation of the AP1000 new fuel storage rack and vault. The 17x17 fuel assemblies are explicitly modeled as shown in Figure 2-1 of TR-67, and each assembly is fully enriched with 5 w% U-235 over the entire length of the active fuel (426.7 cm (168 in).

The 8x9 array of storage cells is modeled with the active fuel of a 17x17 fuel assembly in each location. The 8x9 array of storage cells is centered with the four walls of the concrete vault. The top of the vault is conservatively modeled without a lid. This is a conservative omission because a metal lid would absorb neutrons. Also included in the criticality model is the reflection provided by a postulated 30.48 cm (12 in) of full density water.

The fuel rod, guide tube, and instrumentation tube claddings are modeled with Zircaloy. This is conservative with respect to the Westinghouse ZIRLO product, which is a Zirconium alloy containing additional elements including Niobium. Niobium has a small absorption cross section, which causes more neutron capture in the cladding regions resulting in a lower reactivity. Therefore, this criticality analysis is conservative with respect to fuel assemblies containing ZIRLO cladding in fuel rods, guide tubes, and the instrumentation tube.

There are no burnable absorbers modeled in any of the fuel assemblies. The Zirconium grid straps are conservatively omitted. The 8x9 array of storage cells and vault are modeled at room temperature conditions (20 °C), and the system reactivity is evaluated at 11 moderator densities ranging from 1.0 g/ cm^3 down to 0.001 g/cm^3. A total of 1.2 million neutron histories are

modeled in 1003 generations with 1200 neutrons per generation. It is noted that all KENO V.a results for the first three neutron generations are skipped to eliminate preliminary estimates of the system reactivity.

The methodology bias and uncertainty are discussed and evaluated in "Criticality Safety Criteria," TANS Volume 35, page 278, dated 1980. The results of these KENO calculations showing final K_{eff} values (including bias and uncertainties) versus water density are given in Table 2-2 of TR-67. During a Regulatory Audit on November 16, 2006, the staff reviewed the applicant's calculation, "AP1000 New Fuel Storage Rack Criticality Analysis," which includes the uncertainty calculations as well as the final summation of uncertainties to be applied when calculating K_{eff} values.

Hypothetical Fuel Assembly Drop and Impact on Criticality Analysis

It is possible to drop a fresh fuel assembly into or on top of a storage cell in the AP1000 new fuel storage rack as described in Section 9.1.1.2.1 C of DCD Revision 17. In the event that the dropped fuel assembly hits the top of a storage cell, the applicant's analyses indicate that neither the Metamic® nor the active fuel is adversely impacted. The applicant states there is no degradation of the criticality safety margin as a result of dropping a fuel assembly on top of a storage cell. The staff reviewed the applicant's analyses and agrees with the approach and conclusions.

To conservatively bound the resulting deformation on the base plate following a drop of fuel assembly straight through an empty cell impacting the rack baseplate, the bottom elevations of 25 fuel assemblies were lowered by 12.7 cm (5 in). Even with the bottom elevation of the active fuel in 25 fuel assemblies lowered by 12.7 cm (5 in), the criticality design criteria given in Section 2.0 are still met. This conclusion is based upon the observation that the AP1000 new fuel storage rack is normally dry and not flooded with water. The applicant notes that for this hypothetical dropped fuel assembly accident the AP1000 new fuel storage rack would need to contain at least several feet of water in the bottom of the AP1000 new fuel storage rack before the criticality design basis limits would be exceeded. The accident scenario analyzed is beyond the design basis. Based on its review of the analysis, the staff finds the applicant's design meets the criticality design basis limits for a hypothetical dropped fuel assembly.

Evaluation Results

Figure 2-2 of TR-67 displays the final K_{eff} values (including all biases and uncertainties) versus water density for the AP1000 new fuel storage rack. The maximum fresh fuel enrichment limit for the AP1000 new fuel storage racks is determined to be 5.0 w% U-235 since the final K_{eff} values at this enrichment are less than 0.98 at optimum moderation conditions and less than 0.95 at fully flooded conditions, assuming no soluble boron.

The staff has reviewed the changes submitted in TR-67, Revision 0. Given that: (1) the maximum K_{eff} value, including all biases and uncertainties, is less than 0.95 with full density unborated water; (2) the maximum K_{eff} value, including all biases and uncertainties, is less than 0.98 with optimum moderation and full reflection conditions; and (3) the maximum enrichment of fresh fuel assemblies is less than or equal to 5.0 w/o U-235, the AP1000 new fuel storage rack fully loaded with Westinghouse AP1000 17x17 fuel assemblies with an enrichment less than or equal to 5.0 w/o U-235 satisfies the criticality safety criteria specified in 10 CFR Part 50.68 and GDC 62 and, therefore, is acceptable.

9.1.1.2.3.3 Conclusion

The staff has reviewed the DCD Section 9.1.1 changes provided by DCD Revision 17 and supported by TR-67 submitted by the applicant, describing the new fuel storage racks, the criticality analyses performed, and the methods used. Based on this review, the staff concludes that the appropriate documentation was submitted and that the criticality aspects of the new fuel storage racks meet the requirements of GDC 62 related to the prevention of criticality.

9.1.1.3 Conclusion

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

In its previous evaluations of AP1000 DCD, Section 9.1.1, the staff identified acceptance criteria based on the design meeting relevant requirements in GDC 2, GDC 4, GDC 5, "Sharing of Structures, Systems and Components," GDC 61, "Fuel Storage and Handling and Radioactivity Control," in GDC 62, and in GDC 63, "Monitoring Fuel and Waste Storage." The staff found that the AP1000 new fuel storage design was in compliance with these requirements, as referenced in NUREG-0800 Section 9.1.2 and determined that the design of the AP1000 new fuel storage, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 new fuel storage as documented in AP1000 DCD against the relevant acceptance criteria as listed above and in SRP, Sections 9.1.1, and 9.1.2. The staff finds that the applicant's proposed changes do not affect the ability of the new fuel storage to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD. The staff concludes that the AP1000 new fuel storage design continues to meet all applicable acceptance criteria. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes incorporated into Revision 17 contribute to the increased standardization of the certification information in the AP1000 DCD and, thus, meet the requirements of 10 CFR 52.63(a)(1)(vii). Therefore, the staff finds that the proposed changes to AP1000 Section 9.1.1 are acceptable.

9.1.2 Spent Fuel Storage

9.1.2.1 Summary of Technical Information

Section 9.1.2, "Spent Fuel Storage" of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In AP1000 DCD, Revision 17 the applicant has proposed to make the following changes to Section 9.1.2 of the certified design:

1. Spent fuel rack design change. The basis for this change is documented in TR-54, "Spent Fuel Storage Racks Structure and Seismic Analysis," APP-GW-GLR-033, Revision 4 of June 2010. The applicant added detailed design information for the spent fuel racks and proposed several changes throughout DCD Section 9.1.2 to reflect the changes addressed in TR-54.

2. SFP water level increase. The basis for this change is documented in TR-121, "Spent Fuel Pool Water Level and Dose," APP-GW-GLN-121, of May 2007. The applicant proposed several changes throughout DCD Section 9.1.2 to reflect the changes addressed in TR-121.

3. Fuel handling crane change. The applicant proposed to replace references to the fuel handling jib crane for the new-fuel handling crane. The basis for this change is addressed in TR-106.

4. The applicant proposed several changes throughout DCD Section 9.1.2 to reflect the spent fuel rack criticality analysis change addressed in TR-65, "Spent Fuel Storage Racks Criticality Analysis," APP-GW-GLR-029, Revision 3 of March 3, 2011 including resolving COL Information Item 9.1-4 by performing a confirmatory criticality analysis for the spent fuel racks.

9.1.2.2 Evaluation

The staff reviewed all changes identified in Section 9.1.2 of the AP1000 DCD. The staff did not re-review descriptions and evaluations of the spent fuel storage in AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. All technical changes in the DCD are supported by information presented in the applicant's TRs.

The regulatory basis for AP1000 DCD, Section 9.1.2, is documented in NUREG-1793. The staff has reviewed the proposed changes to DCD Section 9.1.2 against the applicable acceptance criteria of NUREG-0800 Sections 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling" and 9.1.2, "New and Spent Fuel Storage." The following evaluations discuss the results of the staff's review.

The specific criterion that applies to the changes evaluated in this section is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

9.1.2.2.1 Spent Fuel Rack Design Change

TR-54, Revisions 0 and 1, described the design details and design-basis analyses for the spent fuel racks. To be consistent with the design of the spent fuel racks and the analyses presented in TR-54, Revisions 0 and 1, the following changes were proposed in DCD Revision 17, Section 9.1.2:

1. Delete references "to supporting grid structures at the top and bottom elevations" and replaced them with "a thick base plate at the bottom elevation" in DCD Section 9.1.2.1.

2. Increase SFP capacity from 619 fuel assembly locations to 889 fuel assembly locations.

3. Replace Figures 9.1-2 and 9.1-3, which show the rack array center-to-center spacing, with new Figures 9.1-2, 9.1-3, and 9.1-4, which show the SFP rack layout in Region 1, Region 2, and overall SFP rack layout, respectively.

4. Add the following statement in DCD Section 9.1.2.1: "All spent fuel racks will be in place whenever fuel is stored in the spent fuel racks. See DCD Section 3.7.5.2, for discussion

of site-specific procedures for activities following an earthquake. An activity will be to address measurement of the post-seismic event gaps between spent fuel racks and to take appropriate corrective actions."

5. Add the following statement in DCD Section 9.1.2.1: "The stress analysis of the spent fuel racks satisfies all of the applicable provisions in NRC RG 1.124, Revision 1 for components designed by the linear elastic analysis method."

6. Add design scope, in addition to SFP, the fuel transfer canal, and the cask loading pit, the FHM traverses the new fuel storage pit and the rail car bay in DCD Section 9.1.2.1.

7. Delete the following statement in DCD Section 9.1.2.2.1: "The purchase specification for the spent fuel storage racks requires the vendor to perform confirmatory dynamic and stress analyses."

8. Add the additional design information on the spent fuel racks center-to-center spacing; and material characteristics and recommended monitoring schedule of the neutron absorbing material (Metamic) coupons used in the fuel storage racks in DCD Section 9.1.2.2.1.

9. Add the following statement in DCD Section 9.1.2.2.1: "Both of these rack module configurations provide adequate separation between adjacent fuel assemblies with neutron absorbing material to maintain a subcritical array."

10. Change design activities: Reference to seismic and stress analyses of the spent fuel racks are changed to present tense (e.g., are performed) from future tense (e.g., will be performed by the vendor) in DCD Section 9.1.2.2.1.

11. Add item F in DCD Section 9.1.2.2.1, addressing loads due to "Internally Generated Missiles."

12. Changed Equipment Classification: In DCD Section 9.1.2.3, the racks are changed from Equipment Class 3 to Equipment Class D.

13. Delete the following statement in DCD Section 9.1.2.3: "because of the close spacing of the cells, it is impossible to insert a fuel assembly in other than design locations."

9.1.2.2.1.1 Spent Fuel Rack Structural

9.1.2.2.1.1.1 Summary of Technical Information

The staff confirmed that the proposed changes to DCD Revision 17, Section 9.1.2 are consistent with the design changes identified in TR-54, Revisions 0 and 1. In order to resolve the outstanding issues, and also to evaluate subsequent changes to the spent fuel rack design and changes to the spent fuel rack design basis, submitted to the staff in TR-54, Revisions 2 and 3. TR-54, Revision 4, was issued in June 2010, to update both the design and the design basis, and to address the unresolved RAIs.

In total, the staff issued 44 RAIs for TR-54. The significant RAIs are discussed below. Any RAIs not specifically discussed herein were for clarification only, and did not require a detailed technical response by the applicant.

9.1.2.2.1.1.2 Evaluation

Section 2.8.5 of TR-54 indicates that the three drop scenarios all assume a 91.44 cm (36 in) drop height above the top of the spent fuel storage rack. In RAI-TR54-01, the staff requested that the applicant describe the fuel handling operation that determines this drop height.

In a letter dated April 9, 2007, the applicant stated that the fuel handling operations for the Section 2.8.5 scenarios are the normal fuel handling operations such as those performed during refueling outages, i.e., core fuel offloaded and reloaded from/to the Reactor Building via the fuel transfer system into and out of the SFP storage racks. There are also instances where fuel inspection and/or fuel repair will require the fuel to be removed from the spent fuel storage racks and moved to the designated fuel inspection or fuel repair workstation. These fuel handling operations include the transfer of fuel from the rack cells and into the cask area during dry cask storage operations.

The applicant also stated that the current fuel transfer system in the spent fuel building, which lifts and lowers the fuel during normal fuel handling operations, consists of the FHM and the SFHT. The FHM is a fixed mast manipulator-type bridge crane similar in design to the manipulator crane bridges used in numerous existing Westinghouse operating plants' Reactor Buildings. The SFHT is a long handled tool which latches onto the fuel assembly top nozzle via a manually actuated gripper. Lifting of the SFHT and attached fuel assembly is performed using an auxiliary hoist on the FHM bridge. The design of the SFHT is very similar in design to the fuel handling tool currently in use at numerous existing Westinghouse operating plants.

The applicant indicated that the current designs of the AP1000 SFP, spent fuel storage racks, FHM, and SFHT limit the height that the fuel assembly can be lifted above the spent fuel racks to 22.86 cm (9 in) maximum. This height is limited by the water coverage above the fuel assembly and is limited by the physical design of the FHM and SPHT via mechanical stops and/or tool length. The maximum fuel drop height will be approximately 22.86 cm (9 in); which is bounded by the Section 2.8.5 scenarios of 91.44 cm (36 in).

At the June 2010 audit, the staff inquired about the operational controls on spent fuel handling that will ensure no exceedance of the analyzed 91.44 cm (36 in) drop height. In a letter dated July 13, 2010, the applicant submitted a supplemental response to RAI-TR54-01, describing how this will be accomplished. The staff reviewed the RAI response and concluded that there are sufficient hardware, software, and administrative controls in place to ensure that the design basis fuel assembly drop height of 91.44 cm (36 in) will not be exceeded. Therefore, RAI-TR54-01 is resolved.

As described in Section 2.8.5, the objective of the LSDYNA impact analyses is to assess the extent of the permanent damage to the rack and the structural integrity of the SFP liner. In RAI-TR54-02, the staff inquired whether the analyses are also intended to address the structural condition of the dropped fuel assemblies. If the analyses are also intended to address damage to the fuel assemblies, the staff would need additional fuel assembly design details and LSDYNA analysis details. If not, the applicant was asked to identify where this is addressed in the DCD or other technical report(s).

In a letter dated April 9, 2007, the applicant provided the following response:

> The LSDYNA impact analyses are not intended to address the structural condition of a dropped fuel assembly. The analysis addresses the structural condition of the rack and its ability to maintain subcriticality. A fuel handling accident and its radiological consequence is addressed in DCD Subsection 15.7.4 Fuel Handing Accident. The fuel handling accident described in Subsection 15.7.4 is defined as the dropping of a spent fuel assembly such that every rod in the dropped assembly has its cladding breached so that the activity in the fuel/cladding gap is released.

The applicant referenced report, APP-FS02-Z0C-001, Revision 0, "Analysis of AP1000 Fuel Storage Racks Subjected to Fuel Drop Accidents," for the requested LSDYNA analysis details.

The staff confirmed that TR-54, Revision 1, removed any information describing the fuel assembly capacity and margins, since this is not the purpose of the TR-54. The staff also confirmed that TR-54, Revision 2, Section 2.8.1.4, and related sections in DCD Revision 17 were revised to include the maximum calculated fuel impact load on the racks (demand), and show it is less than the allowable impact load (capacity). Therefore, RAI-TR54-02 was resolved.

Section 2.8.5 of TR-54 indicates that a quarter of the spent fuel rack and a single fuel assembly were modeled appropriately using LSDYNA's shell and solid elements. The rack is submerged under the water when an impact occurs. In RAI-TR54-04, the staff requested the applicant confirm whether the water-structure interaction has been accounted for or to provide an explanation why this effect is not important.

In a letter dated April 9, 2007, the applicant stated that the fuel assembly drop analyses conservatively neglect the water structure interaction in the wake of an impact. By assuming that the impact occurs in air, as opposed to water, none of the impact energy is diverted to fluid kinetic energy (i.e., fluid damping), which would attenuate the deformation to the fuel rack. The applicant referenced report APP-FS02-Z0C-001, Revision 0.

The staff reviewed the applicant's response and concluded that neglecting the mitigating effects of the water is conservative; therefore, the response is acceptable, and RAI-TR54-04 was resolved.

Section 2.8.5 of TR-54 states that appropriate non-linear material properties have been applied to the rack components to permit yielding and permanent deformation. TR-54 Table 2-6 provides Young's modulus, yield strength, and ultimate strength. LSDYNA requires a true stress-strain relation for nonlinear materials. In RAI-TR54-05, the staff requested the applicant to provide the following: (1) a complete description of the material stress-strain curve and confirmation that a true stress-strain curve was used in these impact analyses; and (2) a description of the fuel assembly model, including the element properties and material properties for the dropped fuel assembly.

In a letter dated April 9, 2007, the applicant provided the following response:

> The spent fuel racks are fabricated from SA240-304 and SA564-630 stainless steel. For the impact analyses, a true stress-strain curve, which is obtained from Atlas of Stress Strain Curves (2nd Edition, ASM International) and [included] as

Figure TR-54-5.1 [in the RAI response], is used to define the strength properties of SA240-304 stainless steel.

The properties of SA564-630, which is used to fabricate the adjustable support pedestals, are input in terms of engineering stress-strain, based on material data taken from the ASME Code. Also, the welds that connect the rack components are modeled as a bi-linear elasto-plastic material having the engineering stress-strain properties of the adjoining base metal (i.e., SA240-304). The material property values, which are used to define the engineering stress-strain curves for SA564-630 stainless steel and for the structural welds, are [summarized in the table included in the RAI response].

The fuel assembly is modeled by a rigid bottom end fitting and a mass at the top (representing the weight of lifting tool) connected by an elastic beam (with a Young's modulus of 1.04×10^7 psi and a Poisson's ratio of 0.3 for typical rod material) that has an equivalent mass and total cross sectional area of all fuel rods in an AP1000 fuel assembly. In addition, a very thin rigid shell is attached to the bottom end fitting to represent the side surfaces of the fuel assembly that might be in contact with rack cell walls in a shallow drop event. To maximize the damage in the rack, the fuel assembly is only allowed to move in the vertical direction.

The staff reviewed the applicant's response, and determined that additional clarifications were needed, related to representation of nonlinear material properties and modeling of the fuel assembly for the drop analysis. During April 2007 and October 2007 regulatory audits, the staff discussed the needed clarifications with the applicant.

Since curves are only provided at 21.11 °Celsius (C) (70 °Fahrenheit (F)) and 430 °C (806 °F), the staff was not sure how the stress-strain curve at the required temperature (65.55 °C (150 °F)) was obtained from Figure TR- 54-5.1. The applicant indicated that the true stress-strain curve for the stainless steel material at the appropriate temperature (65.55 °C (150 °F)) is derived by manual interpolation of the true stress-strain curves at 21.11 °C (70 °F) and 430 °C (806 °F), which are provided in Atlas of Stress Strain Curves (2nd Edition, ASM International) for Type 304 stainless. The properties were linearly interpolated to obtain the values at 65.55 °C (150 °F). However, the staff requested the applicant to demonstrate whether using a linear interpolation approach is conservative. Using data from the ASME Code Section II, Part D, the applicant showed that the temperature versus yield stress and ultimate stress for stainless steel materials is not linear; the applicant's linear interpolation resulted in a slight overestimation of these values in the LSDYNA drop analyses. The overestimation was less than 4 percent for the ultimate strength and less than 10 percent for the yield. The applicant concluded that the results would not change significantly. The staff reviewed the differences in the two interpolated stress-strain curves, and concluded that the applicant's assessment that the results would not change significantly is acceptable.

The staff was uncertain of the effect that modeling the welds using bi-linear elasto-plastic material having engineering stress/strain properties, rather than the true stress-strain curve, has on the results of the analysis. For the weld, the applicant indicated that the use of the engineering stress strain curve is conservative because it is lower than the true stress strain curve and the failure strain is less. The staff found this acceptable.

From the response dated April 9, 2007, it was not clear to the staff whether using the given Young's modulus and total cross-sectional area of all the fuel rods provides an equivalent axial stiffness of the fuel pellet material and fuel cladding material. The applicant indicated that the axial stiffness of the dropped fuel assembly is based on the cladding material only, rather than the cladding and fuel pellets. The mass of the lifting tool is placed on top of this elastic beam. The mass of the cladding is included within the density of the beam. The fuel mass is lumped at the bottom of the fuel assembly model. The thin rigid shell surrounds the entire fuel assembly to position the single line beam representation within the storage cell.

The applicant agreed to expand the existing information for the fuel assembly model, in the next revision to TR-54. The staff verified that Section 2.8.5 of the TR-54, Revision 1, included sufficient information describing the fuel assembly model. Therefore, RAI-TR54-05 was resolved. The staff confirmed that TR-54, Revision 4, includes the same descriptive information.

The fuel rack baseplate shown in Figures 2-7 and 2-8 of TR-54 appears to have only one layer of 8 node brick element through its thickness. It is not clear if a solid or a thick shell element is used. In RAI-TR54-06, the staff requested the applicant to clarify the type of element used for the baseplate.

In a letter dated April 9, 2007, the applicant stated that the baseplate is modeled using 8-noded solid elements arranged in a single layer. The staff determined that the applicant would need to provide justification, demonstrating that modeling the baseplate using an 8-node solid element can adequately capture the behavior of the plate, including bending through the thickness. The staff noted that the model is used for the drop analyses and the seismic analyses of the racks.

During the April 2007 audit, the staff discussed this concern with the applicant. The applicant agreed to provide a supplemental response to address the staff's concern.

During the October 2007 audit, the applicant indicated that the rack baseplate model was revised to utilize thick shell elements in HOLTEC Report No. HI-2063519, Revision 1. The applicant also indicated that they revised their model to use strain rate effects for the material properties. The net effect of both improvements resulted in lower deformations. The staff found the use of the thick shell element representation of the baseplate to be appropriate because it more closely simulates the true behavior under dynamic impact loadings.

During the May 2008 regulatory audit, the staff verified that HOLTEC Report No. HI-2063519, Revision 2, shows the rack baseplate modeled with thick shell elements, which can capture the bending behavior of the plates. In addition, strain rate effects were also included in the spent fuel rack analysis. These new analyses demonstrate that the maximum downward deformation of the rack baseplate is 7.98 cm (3.14 in), which is less than the 8.89 cm (3.5 in) used in the criticality analysis. Also, the criticality analysis was conservatively performed assuming a total of nine fuel assemblies deform the maximum value of 8.89 cm (3.5 in) rather than the single drop at 7.98 cm (3.14 in). Therefore, RAI-TR54-06 was resolved.

Section 2.8.5 of TR-54 indicates that the baseplate of the rack is connected to the cells by appropriate welding. However, the cells are described in Sections 2.1.1.1 and 2.1.1.2 as resting on top of the baseplate. Welded connections between the cells and the baseplate would greatly increase the strength of the whole rack system. The staff issued RAI-TR54-07, which reads as follows:

a) Confirm there is a welded connection between the baseplate and the cells.
b) Describe the design details of this connection.
c) Describe how this connection is modeled in LS-DYNA.

In a letter dated April 9, 2007, the applicant provided the following response:

a) The base of every storage cell is welded to the rack baseplate.

b) Each cell is welded to the baseplate on four sides by 1/16 in fillet welds having a minimum length of 7 ".

c) The cell to baseplate weld connection is modeled in LSDYNA by shell elements, which join the bottom of the cell and the baseplate top surface, with a thickness equal to the corresponding throat dimension of the weld.

The staff determined that the applicant provided sufficient information to clarify the cell-to-baseplate attachment, and found the response acceptable. Therefore, RAI-TR54-07 was resolved.

For the drop case in which the impact occurs directly above a rack pedestal, Section 2.8.5 of TR-54 provides the concrete strength of the pool floor and the thickness of the stainless steel liner, but does not provide the thickness of the pool floor. There is a possibility that the impact could also cause damage to the concrete floor, and pose a more severe consequence than causing the liner to yield. The maximum Von Mises stress in the SFP liner is reported as 161.3 MPa (23.4 kips per square inch (ksi)), which is much larger than the concrete strength of 27.6 MPa (4 ksi); the concrete may crush and crack locally at this level of stress. In RAI-TR54-08, the staff requested the applicant to provide additional details on the modeling of the concrete floor (including a figure of the concrete model, element type, boundary conditions, material properties, etc.) and the analysis results for the concrete floor (in addition to Figure 2-11).

In a letter dated April 10, 2007, the applicant stated:

> The spent fuel pool concrete floor is modeled only in the vicinity of the impacted rack pedestal with an assumed thickness of two ft and compressive strength of 4,000 psi. The pool liner and rack pedestal bearing pad are also modeled as shown in Figure TR54-8.1 [attached to the RAI response]. The periphery surface nodes of the SFP pool liner and the underlying concrete slab in the LSDYNA model are restrained from moving in the vertical direction and in the horizontal direction normal to the periphery surface to simulate the confining effect of the surrounding structure.
>
> The maximum compressive stress in the concrete floor, resulting from the fuel assembly deep drop event in which the impact occurs directly above a rack pedestal, is predicted to be 4,557 psi as shown in the Figure TR-54-8.2 [attached to the RAI response]. This maximum compressive stress slightly exceeds the assumed concrete compressive strength and is limited to the top surface of the concrete near the bearing pad edge. The very limited local damage to the concrete floor surface is acceptable since the acceptance criterion for the fuel deep drop accident is no gross failure of the SFP floor leading to an uncontrolled loss of SFP water.

The staff determined that the response is acceptable because the bearing capacity of confined concrete is greater than the compressive strength of 27.6 MPa (4,000 psi). As follow-up, the staff requested the applicant to describe the concrete material model used in LSDYNA. During the April 2007 audit, the applicant indicated that concrete nonlinear material model "Material 16" was used in the LSDYNA analysis. The staff requested the applicant to explain how the input parameters for Material 16 were developed and verified, since they do not appear to be derivable from commonly known material properties of concrete.

During the October 2007 audit, the applicant indicated that the properties of Material 16 in LS-DYNA were obtained from NUREG/CR-6608, "Summary and Evaluation of Low Velocity Impact Tests of Solid Steel Billet into Concrete Pads," (February 1998). This concrete model was used by HOLTEC in the generic license for HI-STORM 100 storage casks, which received approval by the NRC. The staff requested the applicant confirm that the 27.6 MPa (4,000 psi) concrete compressive strength for the AP1000 fuel pool structure is consistent with the concrete strength used in the referenced NUREG/CR-6608 report. The applicant stated that the concrete pad modeled in NUREG/CR-6608 has a compressive strength of 29.0 MPa (4,200 psi), which is slightly greater than the compressive strength of the AP1000 SFP slab (27,6 MPa (4,000 psi)). In order to account for the difference in strength, the input parameters specified in Appendix C of NUREG/CR-6608, for use with LS DYNA Material Model 16, have been modified in accordance with HOLTEC Position Paper DS-240 for a compressive strength of 27.6 MPa (4,000 psi).

Since the concrete input parameters were adjusted to account for the differences in compressive strength between the concrete used in NUREG/CR-6608 and the concrete used in the AP1000 design, and considering the small difference in compressive strengths, the staff found this acceptable. Therefore, RAI-TR54-08 was resolved.

Section 2.8.5 of TR-54 does not indicate whether other fuel assemblies are in place when a fuel assembly drops through an empty cell and impacts the baseplate at its center. Depending on how the baseplate is designed, a full load of fuel assemblies may introduce progressive deformation after a fuel assembly impacts at the center of the baseplate. The maximum downward deformation of the baseplate is about 10.92 cm (4.3 in), as shown in TR-54 Figure 2-10. This may be significant enough to initiate a progressive deformation. Therefore, in RAI-TR54-09, the staff requested the applicant to provide: (1) the assumption for in-place fuel assemblies when the impact occurs; (2) the design basis for the baseplate; and (3) a figure similar to Figure 2-10 that shows the cells together with the severely deformed baseplate.

In a letter dated April 10, 2007, the applicant provided the following response:

(1) The spent fuel storage rack is assumed to be empty (i.e., no fuel assemblies in place) when a fuel assembly drops through an empty cell and impacts the baseplate at its center. This is a simplifying assumption, which is reasonable considering that the buoyant weight of a fuel assembly is approximately 1,525 lb whereas the impact load transmitted by the dropped fuel assembly is roughly 268,000 lb based on the LSDYNA solution.

(2) The design basis for the baseplate is to provide vertical support for the stored fuel assemblies and to protect the Spent Fuel Pool liner from a fuel assembly strike. In other words, a dropped fuel assembly should not pierce the baseplate and result in a direct impact with the liner.

During the April 2007 audit, the staff discussed the response with the applicant and requested a supplemental response for items (1) and (3). For item (1) the applicant indicated they would provide a supplement to the response, to address the effects of the additional fuel assemblies during the vertical drop accidents. For item (3) the applicant agreed to provide another detail showing the bottom plate and cell walls in the surrounding region where the maximum deformation and stresses occur.

In a letter dated May 17, 2007, the applicant submitted the revised RAI response. The applicant's response to items (1) and (2) was unchanged. For item (3), the applicant attached a figure to the response, showing the cells together with the severely deformed baseplate, for the same LS DYNA solution as shown in Figure 2-10 of TR-54. The applicant noted that the deformation of the cells is not significant compared to the baseplate, because the cell to baseplate weld connections broke as a result of the postulated fuel impact load before the cell walls were permanently deformed. The staff reviewed the applicant's revised response, and found the response to item (3) acceptable.

During the October 2007 audit, in response to item (1), the applicant indicated that the rack model, represented in HOLTEC Report No. HI-2063519, Revision 1, had been revised to consider the effects of all of the stored fuel assemblies in the rack, by modifying the density of the rack baseplate. The modeling of the baseplate was also changed to thick shell elements, and strain rate effects were included. The staff reviewed the HOLTEC calculation and confirmed that it includes the mass effect of all of the fuel assemblies by increasing the baseplate density. The staff concluded that the consideration of the rest of the fuel assemblies by increasing the mass of the baseplate is an acceptable approach to simulate the dynamic effects. The staff requested the applicant to finalize the HOLTEC calculation and revise TR-54, to describe the modeling approach used. The staff noted that its concern about the large vertical deformation (10.92 cm (4.3 in)) is addressed under RAI-TR54-10.

During the May 2008 audit, the staff verified that HOLTEC Report No. HI-2063519, Revision 2, shows that the rack is conservatively assumed to be fully loaded with fuel assemblies, except at the center cell where the dropped fuel hits the rack baseplate. Therefore, RAI-TR54-09 was resolved.

In Section 2.8.5 of TR-54, a vertical movement of 5.08 cm (2 in) of a fuel assembly is defined as the criticality limit, and the impact analysis shows that quite a number of fuel assemblies will have more than 5.08 cm (2 in) displacement. It appears that a rack design with only a 5.08 cm (2 in) space between the bottom of the baseplate and the top of the floor would eliminate this risk. In RAI-TR54-10, the staff requested the applicant to explain why the design has a space larger than 5.08 cm (2 in).

In a letter dated April 9, 2007, the applicant stated that each spent fuel rack storage cell is 506.73 ± 0.159 cm (199.5 ?± 0.0625) in) in length and rests on top of a base plate whose top is 5 in above the SFP liner. Each Metamic poison panel is 436.9 cm (172 in) long and has a bottom elevation that is 15.8 cm (6.23 in) above the top of the base plate. The active fuel region of each fuel assembly begins at an elevation 20.9 cm (8.23 in) above the top of the base plate. Therefore, the bottom elevation of the Metamic poison panel is positioned to be 5.08 cm (2 in) lower than the bottom elevation of the active fuel.

Therefore, the results of the criticality analyses are bounding even if the fuel assembly is vertically displaced downward by up to 5.08 cm (2 in) as a result of the hypothetical fuel

assembly drop. The 5.08 cm (2 in) vertical displacement of the fuel assemblies, mentioned above in RAI-TR54-10, is not a criticality limit. The criticality analyses summarized in TR-65, Revision 0, addressed the hypothetical fuel assembly drop in Section 2.4.6.3 as follows:

> A fuel assembly (with a control rod and attached to the fuel assembly handling tool) is dropped and impahcts the baseplate as discussed in Subsection 2.8.5 of Reference 4. The analysis in Subsection 2.8.5 of Reference 4 indicates that the base plate will be defomed as a result of this drop. For conservatism, the fuel assemblies in nine storage cell locations will be modeled as vertically dropped by 3.5 inches to bound the consequences of the base plate deformation.

Since the criticality analysis demonstrates that the stored fuel assemblies remain subcritical following a hypothetical fuel assembly drop, the space between the bottom of the baseplate and the top of the floor is not designed to control criticality, but to protect the SFP liner from an impact strike. In other words, the rack baseplate is raised high enough above the floor (10.8 cm (4.25 in)) to prevent the baseplate from contacting the SFP liner when the baseplate deforms under impact.

The staff reviewed the above response and concluded that the response is acceptable. However, it should be noted that the maximum displacement of 8.89 cm (3.5 in) reported here does not agree with the 10.9 cm (4.3 in) indicated in Figure 2-10 of TR-54. During the April 2007 audit, the staff noted that the applicant needed to address the higher base plate deformation (10.9 cm (4.3 in)) in the vertical direction and how this may affect the criticality analysis.

During the October 2007 audit, the applicant indicated that the hypothetical drop, wherein a fuel assembly travels downward through an empty storage cell and impacts the baseplate, has been re-analyzed in HOLTEC Report No. HI-2063519, Revision 1, for both the Region 1 and Region 2 spent fuel racks. The new analysis model incorporates the following changes (as discussed in the RAI responses to RAI-TR54-06, RAI-TR54-09, and RAI-TR54-11): (1) the baseplate is modeled with thick shell elements; (2) the effect of the stored fuel assemblies is accounted for by increasing the mass density of the baseplate; and (3) strain rate effects are considered for the welds only. Based on the new analyses for the Region 1 and Region 2 spent fuel racks, the maximum vertical displacement of the rack baseplate is 7.98 cm (3.14 in), which is less than the 8.89 cm (3.5 in) displacement considered in the criticality analysis. Therefore, the existing criticality analysis remains bounding.

The improvements made to the fuel rack models were reviewed by the staff and found to be technically acceptable. The staff requested the applicant to finalize the HOLTEC calculation and to revise TR-54 to describe the modeling approach used.

During the May 2008 audit, the staff verified that HOLTEC Report No. HI-2063519, Revision 2, shows that: (1) the baseplate is modeled with thick shell elements; (2) the effect of the stored fuel assemblies is accounted for by increasing the mass density of the baseplate; and (3) strain rate effects are considered for the welds. The resulting vertical deformation due to the drop is 7.98 cm (3.14 in). Therefore, RAI-TR54-10 was resolved.

Figure 2-9 of TR-54 shows the permanent deformation at the top of a cell wall at Region 2. The permanent deformation is measured as 50.8 cm (20 in), which is just slightly smaller than the limit of 52.07 cm (20.5 in). Since the deformation at the impact location is so close to the limit (i.e., very little margin exists), the mesh should be locally refined, to ensure convergence with

mesh size. Therefore, in RAI-TR54-11, the staff requested the applicant to perform an additional analysis with a finer mesh at the impact region to confirm that the LSDYNA model is suitable.

In a letter dated April 9, 2007, the applicant stated that the general acceptance criterion for the 91.44 cm (36 in) fuel assembly drop onto the top of a Region 2 rack is to maintain the stored fuel assemblies in a subcritical configuration. In measurable terms, the permanent deformation of the rack (measured downward from the top of rack) is limited to 52.07 cm (20.5 in), which is equal to the distance from the top of the rack to the top of the neutron absorber panel. This limit is conservative because the active fuel region begins 5.08 cm (2 in) below the top of the neutron absorber panels. Therefore, more margins exist than TR-54 indicates, and a mesh convergence study is not required.

The staff noted that, although the distance from the top of the rack to the top of the fuel is 57.15 cm (22.5 in), which gives a margin of 6.35 cm (2.5 in), this is still a relatively small margin. Because of the small margin, the staff requested the applicant to demonstrate whether the finite element model is sufficiently refined in the impact region.

During the April 2007 audit, the staff discussed this with the applicant. The applicant noted that the neutron absorber panels, beyond the 50.8 cm (20 in) deformation, respond in the elastic range and would not be damaged. The applicant also noted that the criticality analysis was performed for 9 fuel assemblies having the active fuel region exposed 8.89 cm (3.5 in) (see response to RAI-TR54-10).

During the October 2007 audit, the applicant indicated that the 91.44 cm (36 in) fuel assembly drop onto the top of a Region 2 rack has been re-analyzed in HOLTEC Report No.HI-2063519, Revision 1, with consideration of strain rate effects for the welds. The new analysis shows that the maximum permanent deformation of the rack cell wall is only 35.7 cm (14.06 in) (measured from the top of rack) versus the allowable limit of 53.71 cm (21.145 in) as defined in the analysis report. Since the margin between the calculated deformation and the allowable limit is greater than 17.78 cm (7 in), the applicant stated that there is no longer a need to demonstrate that the refinement of the model is adequate in the localized region of the impact zone.

The staff, however, requested the applicant to confirm the adequacy of the rack model in the crushed zone region by providing curves that compare the hourglass energy to the kinetic, internal, and/or total energy. The applicant provided these curves, which showed that the hourglass energy was essentially negligible in comparison to the internal energy of the cell structure and impact bar that were being plastically deformed during these drop accident cases. In view of the now much larger margins in the extent of plastic deformation in the new revised model, and the comparison of the hour glass energy, the staff found the response technically acceptable, pending the finalization of the HOLTEC calculation and revision of TR-54 to describe the new results.

During May, 2008 audit, the staff verified that HOLTEC Report No. HI-2063519, Revision 2, shows the results of the drop analyses onto the top of the fuel racks considering the strain rate effects for the welds. In view of the now much larger margins in the extent of plastic deformation in the new revised model, and the comparison of the hour glass energy, the staff found the response technically acceptable. The TR does not go to this level of detail; however, the staff confirmed the HOLTEC calculation incorporated the above modeling change. Therefore, RAI-TR54-11 was resolved.

There are a total of six impact analyses for the Region 1 and Region 2 racks (3 drop cases for each rack region). TR-54 only presents the results for three analyses, on the basis that these are the bounding conditions. In RAI-TR54-13, the staff requested the applicant to explain the technical basis for concluding that these are the bounding conditions, or provide the results for the three analyses not presented in the report.

In a letter dated May 17, 2007, the applicant stated that the analysis was performed for both the Region 1 and Region 2 racks. The bounding analysis was reported. The shallow drop event involving a Region 1 rack was analyzed and found to yield a plastic deformation of 38.1 cm (15.0 in) measured vertically from the rack top, which is bounded by the reported Region 2 rack shallow drop analysis with a predicted rack top plastic deformation of 50.8 cm (20 in). Similarly, Region 2 rack baseplate was found to deform less than 8.38 cm (3.3 in), which is smaller than the reported bounding baseplate deformation of 10.83 cm (4.264 in) for the Region 1 rack. Finally, because the Region 1 rack is lighter than the Region 2 rack, more impact energy can be transferred into the SFP floor and therefore results in bounding SFP floor damage in an event where the fuel assembly drops directly over a Region 1 rack pedestal; this bounding case was analyzed and reported.

The staff evaluated the RAI response and determined that the response provided an adequate explanation of why the three accident drop cases that were analyzed bound the other three drop cases. Therefore, the staff found the response acceptable and RAI-TR54-13 was resolved.

In accordance with NUREG-0800 Section 3.8.4, Appendix D, one of the fuel handling accident loads that need to be considered is uplift force on the rack caused by a postulated stuck fuel assembly. Section 2.8.3 of TR-54, Revision 0, states: "An evaluation of a stuck fuel assembly, leading to an upward load of 2,000 lb has been performed. The results from the evaluation show that this is not a bounding condition because the local stresses do not exceed 2,500 psi." The staff determined that additional information was needed in order to assess whether this load has been adequately considered. In RAI-TR54-14, the staff requested that the applicant provide a detailed description of the assumptions, the analyses conducted, the results obtained, and the basis for the conclusion that this is not a bounding condition.

In a letter dated May 17, 2007, the applicant stated: “A nearly empty rack with one corner cell occupied is subject to an upward load of 2000 lbf, which is assumed to be caused by the fuel sticking while being removed." The applicant attached a calculation, and stated that the local stress is well below the yield stress of the cell wall material (i.e., 147 MPa (21,300 psi) pursuant to Table 2-6). The applicant also noted that the value of 17.2 MPa (2,500 psi) will be changed in Section 2.8.3 to 20.7 MPa (3,000 psi) for the local stresses resulting from a stuck fuel assembly.

In a letter dated April 18, 2008, the applicant augmented its initial RAI response, stating that the calculation provided in the RAI response is excerpted from HOLTEC Report No. HI-2063523, (APP-FS02-S3C-001, Revision 0). The staff evaluated the supplemental RAI response, including the calculation that demonstrates the adequacy of the vertical welds along the height between adjoining cells and the horizontal welds at the base (cell walls to baseplate). The staff found all responses to staff’s concerns are satisfactory, pending revision to the HOLTEC calculation and the TR.

In December 2008, the staff reviewed TR-54, Revision 2, and related sections in DCD Revision 17. The staff noted that a new Section 2.8.6 has been added to TR-54, Revision 2, to describe the stuck fuel assembly evaluation. Section 2.8.6 states, “A nearly empty rack with

one corner cell occupied is subject to an upward load of 5,000 lbf, which is assumed to be caused by the fuel sticking while being removed." This increase in uplift force was necessitated by changes in the spent fuel handling operations. The staff found the re-analysis to be acceptable. The predicted stresses are still below allowable stresses. Therefore, RAI-TR54-14 was resolved.

The staff noted that the descriptive information included in TR-54 was not sufficient to permit an adequate review of the structural/seismic analysis of the spent fuel racks, in accordance with NUREG-0800 Section 3.8.4, Appendix D. In RAI-TR54-15, the staff requested the applicant provide descriptive information, including plans and sections showing the spent fuel racks, pool walls, liner, and concrete walls. All of the major features of the racks including the cell walls, baseplate, pedestals, bearing pads, neutron absorber sheathing, any impact bars, welds connecting these parts, and any other elements in the load path of the racks should be shown on one or several sketches. These sketches should also indicate related information which includes key: cutouts, dimensions, material thicknesses, and gaps (fuel to cell, rack to rack, rack to walls, and rack to equipment area). In addition to the above, for review of postulated fuel handling drop accident and quantification of the drop parameters, sketches with sufficient details for the fuel handling system should be provided to facilitate the review as indicated in NUREG-0800 Section 3.8.4, Appendix D.

In a letter dated June 8, 2007, the applicant provided sketches in the attachment to the RAI-TR54-15 response, showing the major features of the racks and SFP. The applicant stated that these sketches would be incorporated in TR-54. The detailed design of fuel handling equipment and detailed sketches are not available. However, the quantification of the drop parameters has been established in the DCD (both maximum drop weights and heights). The DCD drop heights are much greater than what is being designed for the fuel handling equipment, which is stated in the RAI-TR54-01 response.

During the April 2007 audit, the complete design drawings of the spent and new fuel racks were available to the NRC for review. In addition, HOLTEC explained how the rack features were incorporated into the seismic/structural models.

During the October 2007 audit, the staff discussed five specific items with the applicant: key dimensions of the male and female pedestal components and bearing plates; welds connecting the pedestals to the baseplate; welds connecting the baseplate to the fuel cell walls; leak chase channels; and gaps between the racks. The applicant provided additional information to the staff, describing these details.

During the May 2008 audit, the staff reviewed all the pending revisions to TR-54 and the DCD. The staff identified the need for additional changes to certain figures in TR-54 and the need to include them in the DCD for consistency.

In a letter dated June 20, 2008, the applicant provided its supplemental response to this RAI. The staff reviewed the response and found that TR-54, Revision 2, and related sections in DCD Revision 17 had been appropriately revised. Therefore, RAI-TR54-15 was resolved.

The staff noted that TR-54 is a summary report. However, to adequately perform a technical review of the analysis and design of the spent fuel racks, a more detailed report should have been submitted, similar to those provided in past technical reviews of spent fuel racks for specific nuclear power plants. Therefore, in RAI-TR54-16, the staff requested the applicant provide the detailed spent fuel storage rack report/calculation for review.

In a letter dated April 9, 2007, the applicant stated that TR-54 is a summary report; and two calculations [HOLTEC Report No. HI-2063523, "Spent Fuel Rack Structural/Seismic Analysis for Westinghouse AP1000," Revision 0, 08/15/2006 (APP-FS02-S3C-001); and HOLTEC Report No. HI-2063519, "Analyses of AP1000 Fuel Storage Racks Subjected to Fuel Drop Accidents," Revision 0, 08/15/2006 (APP-FS02-Z0C-001)] form the basis for TR-54.

During the April 2007 audit, the staff reviewed both HOLTEC calculations. Based on this review, the staff requested the applicant to address nine specific questions related to gaps between racks, modeling assumptions, and solution convergence in the seismic analysis. Two significant issues required the applicant to conduct additional analysis. The staff requested that the applicant provide a justification for the assumptions used in developing the various spring stiffnesses, including how variations in the spring stiffnesses affect the results (e.g., sensitivity studies). The staff also requested the applicant to address the sensitivity of the numerical results to the integration time step used in the analysis. The remaining seven questions were requests for clarifications, not requiring additional analysis. The staff also requested the applicant to address 11 specific questions related to design parameters (e.g., weight, temperature, flow) and to analysis methods used for the drop accident analysis. These questions were requests for clarifications, not requiring additional analysis. The applicant agreed to address all of the staff's questions by making appropriate changes and/or additions to the calculations.

During the October 2007 audit, the applicant indicated that, in order to address possible variations in the spring constants, a sensitivity study was performed in which the calculated impact spring constants were uniformly increased and decreased by 20 percent in two separate computer runs. The COF used for all three computer runs (i.e., 80 percent, 100 percent, and 120 percent of the calculated spring constants) is 0.8. The applicant also indicated that the time integration step used for the computer runs is 1×10^{-5} sec. In order to verify that the runs are converged, an additional computer run (Run 6) was performed using a time integration step of 5×10^{-6} sec (i.e., half of the original time step). The differences in results are minimal. HOLTEC Calculation HI-2063523 was updated to include the results of these sensitivity studies, and the final safety conclusions are based on the maximum results from all computer runs, including the sensitivity studies. The staff reviewed HOLTEC Calculation HI-2063523, Revision 1, and noted that the calculation had been revised to include the numerical results for the sensitivity studies, and that these results were considered with all the other runs in the structural assessments.

During the August 2009 audit, the staff confirmed that all 20 of the staff's questions had been appropriately addressed in updates to the HOLTEC calculations and corresponding Westinghouse reports. Therefore, RAI-TR54-16 was resolved.

The staff noted that insufficient data were provided in TR-54 describing the seismic input loads used for analysis of the spent fuel racks. The staff issued RAI-TR54-17, which reads as follows:

a. Floor response spectra (X, Y, and Z - vertical directions) at or the near the elevation of the top of the fuel racks and near the bottom of the fuel rack or pool floor corresponding to the damping value used for the analysis.

b. Explain why the envelope of these two sets of spectra was not used.

c. The current DCD is applicable for the hard rock site. Therefore, provide further explanation for the range of soil and rock properties used in enveloping the seismic floor spectra. Where are these ranges of soil/rock properties specified for confirmation by a future COL applicant?

d. For the synthetic time histories, provide plots of the three time histories, the cross correlation coefficients, the comparisons of the spectra from the synthetic time histories to the enveloped target response spectra, and the comparisons of the power spectral density plots to the target power spectral density function associated with the target response spectra.

e. Which time history was used (displacement, velocity, or acceleration)? Were all three directions input simultaneously? Was gravity included in the time history analysis?

In a letter dated April 9, 2007, the applicant provided the following response:

a. Floor response spectra (X, Y, and Z vertical directions) near the elevation of the bottom of the spent fuel pool floor corresponding to the damping value used for the analysis are provided in the PDF attachment RAI-TR54-17a. No Floor response spectra are provided near or at the elevation of the top of the spent fuel racks (See response to RAI-TR54-17b).

   The ASB99 floor response spectra represent the enveloping response spectra for the auxiliary shield building at elevation 99 ft for a range of soil/rock condition. FRS of various soil/rock analyses were first enveloped for various locations of the ASB. All of the ASB locations at elevation 99 ft were then grouped and enveloped to develop the ASB99 floor response spectra. The spent fuel pool is at a lower elevation but the dynamic response is essentially the same as at elevation 99 ft.

b. The spent fuel racks are free-standing in the spent fuel pool. They are not anchored to the spent fuel pool walls. The spent fuel racks are excited in a seismic event by the floor response spectra representing the spent fuel pool floor (ASB99). There is no need to envelope multiple sets of floor response spectra.

c. The range of soil and rock conditions for which the seismic floor spectra applies is described in Westinghouse technical report "Extension of NI Structures Seismic Analysis to Soil Sites."

d. The synthetic time histories, the response spectrum curves, and the power spectral density plots for the Auxiliary Shield Building at Elevation 99 ft are provided in Figures TR54-17.1 through TR54-17.9 (attached to RAI response). The cross correlation coefficients for the three orthogonal components (East West, North South, and Vertical) of the ASB99 synthetic time histories are summarized in the table [submitted as part of this response].

e. Acceleration time histories are used as the input motion for the seismic analysis of the spent fuel racks. The acceleration input is defined by three orthogonal components, which are input and solved simultaneously. Gravity is also included in the time history analysis.

The staff found this RAI response to be acceptable because it adequately addressed all of the staff's questions. RAI-TR54-17 was initially resolved. However, subsequent to the initial resolution of this RAI, the applicant revised the seismic design loads twice. Therefore, during the June 2010 audit, the staff requested that the applicant update this RAI response to reflect the current seismic design loads for the spent fuel racks. In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-TR54-17, updating the seismic design loads. The staff finds that the revised RAI response adequately describes the current seismic design loads for the spent fuel racks. TR-54, Revision 4, includes the numerical results for the current design loads. Therefore, RAI-TR54-17 is resolved.

The staff noted that the seismic analyses only considered the bounding values (0.2 and 0.8) for the COF between the pedestal and the pool liner. In RAI-TR54-18, the staff requested the applicant to provide the technical basis for only considering these two bounding values and not other intermediate values. The staff also requested clarifications about sliding. What is assumed to slide, the pedestal to bearing plate or bearing plate to pool liner? If it is the surface between the bearing plate and pool liner, how is damage to the pool liner due to horizontal forces avoided? Are there any physical provisions to prevent the bearing plate and pedestal from sliding to the point that the pedestal centerline would be at or beyond the edge of the bearing plate?

In a letter dated April 10, 2007, the applicant stated that the coefficients of friction used in the seismic analyses, namely 0.2 and 0.8, are consistent with previous spent fuel rack license applications, and they are based on the experiments performed by E. Rabinowicz (TR-54, Reference 21: "Friction Coefficients of Water Lubricated Stainless Steels for a Spent Fuel Rack Facility," MIT, a report for Boston Edison Company, 1976). The lower value of 0.2 produces the maximum sliding displacement between the rack pedestals and the bearing plates. The higher value of 0.8 increases the rocking motion of the spent fuel racks and produces the maximum stress in the rack pedestals. Sliding occurs between the rack pedestal and the bearing plate since these two items are made of different materials (SA564-630 vs. SA240-304), whereas the pool liner and the bearing plate are made of the same material (SA240-304) and are more likely to gall. There are no physical provisions to prevent the rack pedestal from sliding beyond the edge of the bearing plate. The seismic analyses, however, demonstrate that the maximum sliding displacement at the base of the rack is less than the distance between the pedestal outside diameter and the edge of the bearing plate.

The staff reviewed the applicant's response and noted that three COF values (0.2, 0.5, and 0.8) are used for the new fuel rack analysis. The applicant needed to justify why all three values are not utilized for the spent fuel rack analysis. Consideration of an intermediate value is appropriate because the analyses are highly nonlinear and it is not evident which value(s) would govern. The need to consider other values is also discussed in Section 6.4.2 of NUREG/CR-5912, "Review of the Technical Basis and Verification of Current Analysis Methods Used to Predict Seismic Response of Spent Fuel Storage Racks." The staff also requested the applicant to provide the maximum horizontal displacement from the analyses and compare that against the distance between the pedestal centerline and the edge of the bearing plate.

During the October 2007 audit, the applicant indicated that an additional computer run had been performed for an intermediate COF of 0.5. During the May 2008 audit, the staff reviewed HOLTEC Calculation HI-2063523, Revision 1, and confirmed that the report describes the analysis and results for the additional case of a 0.5 COF. In addition, the staff reviewed TR-54, Revision 1, and confirmed that the 0.5 COF case was included. Therefore, RAI-TR54-18 was resolved.

The staff noted that the SFP is divided into 2 regions, with different rack designs. In RAI-TR54-20, the staff requested the applicant to explain the reason for the different type racks (i.e., Region 1 and Region 2). If it is because of different fuel assembly types, explain how the analysis considers the various types and combinations of fuel assemblies (e.g., mass, sizes, gaps, fluid coupling, etc.).

In a letter dated April 9, 2007, the applicant stated that the AP1000 uses only one fuel assembly type. The purpose of the Region 1 racks is to provide storage for up to 243 fresh fuel assemblies with a maximum initial enrichment up to 5.0 w% U-235. This is accomplished by spacing these storage cells on a pitch equal to 27.67 cm (10.9 in.) and employing a "flux trap" poison configuration between consecutive storage cells.

The purpose of the Region 2 storage racks is to provide storage for up to 646 fuel assemblies in a high density configuration. These storage cells employ a pitch equal to 22.93 cm (9.028 in.) and a single poison panel separates consecutive fuel assemblies.

The response clarified that all of the fuel assemblies are the same; only the rack configuration is different between Region 1 and Region 2. Since all of the racks are included in the 3-dimensional pool rack model, the staff found the response acceptable. Therefore, RAI-TR54-20 was resolved.

In RAI-TR54-21, the staff requested the applicant to explain how the different impact stiffness values are determined for the fuel to cell wall, rack to rack, rack to wall, and pedestal to floor. Since the impact forces are affected by the impact spring constants, how is the sensitivity of the impact forces and rack responses to variation in these spring constants addressed? Are impact forces imparted directly onto the cell walls or are there impact bars?

In a letter dated May 17, 2007, the applicant stated that the impact stiffness values for the rack to rack, rack to wall, and pedestal to floor are calculated as shown in the attachment to the RAI response. The fuel to cell wall impact stiffness is determined based on the solution for a simply supported circular plate under a concentrated load applied at its center, where the plate diameter is equal to the cell inner dimension and the plate thickness is equal to the cell wall thickness. The stiffness of the annular plate is then multiplied by the number of loaded storage cells for each rack, since the stored fuel assemblies are assumed to rattle in unison. A sensitivity study has not been performed specifically for the AP1000 spent fuel racks to quantify the effect of variations in the impact stiffness values. However, sensitivity studies have been performed in the past for similar spent fuel rack applications submitted by HOLTEC, which employed the same method of computing the impact stiffness values, and the impact forces were found to be insensitive to small variations in the stiffness values provided that the integration time step was sufficiently small. There are impact bars around the entire perimeter of each Region 2 spent fuel rack at the top of the rack. These bars prevent impact forces from being imparted directly onto the cell walls, and they reinforce the rack cell structure at the point of impact.

During the October 2007 audit, the staff discussed with the applicant the development of the spring constant for impact loads between the stored fuel assemblies and the cell walls, which is based on the solution for a circular plate with simply supported edges subjected to a uniform pressure load. This approach has been used consistently by HOLTEC since the mid 1980's, when the computer code DYNARACK was first developed. As a result, numerous spent fuel rack licensing applications over the past 20 years have relied on this approach. The applicant

also noted that, in response to RAI-TR54-16, a sensitivity analysis was performed in which all spring constants used in the DYNARACK model were uniformly increased and decreased by 20 percent. The stiffer springs resulted in only a 0.4 percent increase in the fuel-to-cell impact load.

The staff concluded that the applicant adequately addressed the sensitivity of the response to variations in the spring constants. Therefore, RAI-TR54-21 was resolved.

Section 2.2.2.2 of TR-54 describes some modeling information for a single rack. It indicates that the rack cellular structure is modeled by a 3-D beam having 3 translational and 3 rotational DOFs at each end, so that two-plane bending, tension/compression, and twist of the rack are accommodated. In RAI-TR54-23, the staff requested the applicant to explain why shear stiffness/deformation is not also included, and to provide more detailed information about how the beam model of the rack was developed, considering that it is an assembly of many square-celled structures welded at discrete locations.

In a letter dated April 9, 2007, the applicant stated that the shear deformation is included in the rack dynamic model. The beam model of the rack was developed based on the applicable Codes, Standards and Specifications given in Section IV(2) of the NRC guidance on SFP modifications entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, which states that "Design ... may be performed based upon the AISC specification or Subsection NF requirements of Section III of the ASME Code for Class 3 component supports." The rack modeling technique is consistent with the linear support beam element type members covered by these codes.

The staff verified that Section 2.2.2.2 of the TR-54, Revision 1, was revised to clarify that shear stiffness/deformation is also included in the rack model. Therefore, RAI-TR54-23 was resolved.

Section 2.2.2.2 of TR-54 refers to Figure 2-2 for the dynamic beam model of a single rack. The text and figure do not adequately describe the model. The staff issued RAI-TR54-24 which reads as follows:

a. Define what each series of nodal degrees-of-freedom (DOFs) correspond to (i.e., nodes 1,2; P1, P2, ...; q4, q5, ..., 1*, 2*, ...). While some of these may be deduced by judgment, the report should clearly define all of these.

b. Explain whether there are 5 nodes and 4 beams along the rack beam model to coincide with the 5 nodes and 4 elements of the fuel assemblies?

In a letter dated April 10, 2007, the applicant provided the following response:

a. Table 54-24.1 defines the nodal DOFs for the dynamic beam model of a single rack as depicted in Figure 2-2 of the technical report.

b. The rack cell structure is modeled as a single beam between two nodes, which are located at the top of the rack and at the baseplate elevation. This is consistent with HOLTEC's standard model for seismic analysis of spent fuel racks, which has reviewed and approved by the NRC on numerous dockets. Although there is not a one to one correspondence between beam nodes and fuel assembly nodes, fuel to cell wall impact loads, which can occur at Elevation 0, 0.25H, 0.5H, 0.75H, and H (where H is the height of the

cell structure), are properly transmitted to the rack beam in accordance with the methodology outlined in TR-54 Reference 11.

The staff found part (a) of the response acceptable, and determined that part (b) would require a review of Reference 11. During the April 2007 audit, the applicant indicated that to satisfy part (a) of the response it will include the information provided in the response in the TR. For part (b,) the applicant provided an explanation of how the fuel-to-cell impact terms at each fuel mass elevation are related to the single beam shape function at the same elevation.

During the October 2007 audit, the applicant indicated that Table 2-18 would be added to TR-54. This table defines the nodal DOFs for a single rack dynamic model. The staff reviewed the proposed revisions to Section 2.2.2.2 and Table 2-18. The staff found this technically acceptable. The staff also reviewed the approach described in Reference 11 of technical report APP-GW-GLR-033, Revision 0, for the use of the coupling terms to relate the internal deformations of the fuel nodes to the external deformations of the fuel rack beam nodes at the two ends. Based on this review and the use of this method in prior licensing submittals to the NRC, the staff found this approach to be acceptable.

The staff determined that the applicant made appropriate changes in TR-54, Revision 1. Therefore, RAI-TR54-24 was resolved.

In RAI-TR54-25, the staff requested the applicant to explain whether only full fuel racks are included in the two simulations, or if several scenarios are considered (i. e., different fill ratios, from partially full to full within a given rack; varying fuel locations within the partially filled rack; varying fill and locations in adjacent racks); and to provide the technical justification if only full racks are considered. The staff also asked whether it would be possible to have less than all fuel racks (8) in the pool. If so, then additional simulations would be needed. If not, is there a requirement in the DCD that specifies all fuel racks must always be in place whenever fuel is stored in any of the racks?

In a letter dated May 17, 2007, the applicant stated that all spent fuel racks, in both simulations, are assumed to be fully loaded with maximum weight fuel assemblies. This scenario bounds any partially loaded configuration since it (1) maximizes the vertical compression and lateral friction loads on the support pedestals and (2) produces the maximum rack displacements and fuel to cell wall impacts. The displacements are larger for a fully loaded rack, as opposed to a partially filled rack, because the dynamic model conservatively assumes that all stored fuel assemblies rattle in unison. Hence, the momentum transferred between the rattling fuel mass and the spent fuel rack is at a maximum for a fully loaded rack. For a partially filled rack, the decrease in rattling fuel mass outstrips the destabilizing effect of an eccentric fuel loading pattern.

The applicant further stated that the SFP rack analysis was performed with all eight fuel racks installed during operation of the SFP, which is consistent with the design intent of the AP1000 Spent Fuel Storage Racks. DCD Section 9.1 will include the statement that all spent fuel racks will be in place in the SFP whenever fuel is stored in the spent fuel racks.

The staff reviewed the response and concluded that the explanation provided in the response appears to support the conclusion that fully loaded racks would be expected to maximize impact forces and displacements. In addition, the use of the maximum weight for the fuel assemblies, the analysis assumption that all stored fuel assemblies rattle in unison, and consideration of the

upper and lower bound COF at all support legs provide added conservatism to bound the results.

However, the staff noted that Section 9.1.2.2.1 of DCD Revision 16, states "The purchase specification for the spent fuel storage racks requires the vendor to perform confirmatory dynamic and stress analyses. The seismic and stress analyses of the spent fuel racks consider the various conditions of full, partially filled, and empty fuel assembly loadings." Therefore, the applicant needed to explain this inconsistency between the DCD and the analyses actually performed. The staff noted that this statement occurs in two locations in DCD Section 9.1.

During the October 2007 audit, the applicant indicated that HOLTEC would perform additional analyses considering partially filled racks (from empty to full condition of various racks) in order to address the requirements in the DCD.

During the May 2008 audit, the staff reviewed HOLTEC Report No. HI-2063523, Revision 2, and draft TR-54, Revision 2, which show that additional cases were analyzed, which considered different racks in the pool having varying fill of fuel assemblies ranging from 0 percent, 25 percent, 33 percent, 66 percent, 75 percent, and 100 percent fill. These results, along with the other variations were presented in the report, and subsequent stress evaluations for the racks were based on the worst-case results from all nine cases considered.

TR-54, Revision 2, and related sections in DCD Revision 17 indicated that additional analysis for mixed loading of spent fuel racks was considered. Table 2-4 in TR-54, Revision 2, was revised to add three more simulations consisting of "fully loaded with modified gaps," "mixed loadings," and "empty" rack cases. Based on its review of these additional analyses, the staff requested the applicant to address the following items:

1. The report does not describe these additional cases, as discussed during the May 2008 audit. The staff's understanding from the audit was that in the mixed loading case: (1) different racks had varying fill from 0 percent through 100 percent; and (2) the partially filled racks had the fill offset (i.e., center of mass was offset from the centerline of the rack in the computer model) to simulate the possibility of placing fuel assemblies closer to the side of the fuel rack rather than uniformly distributed. Provide the following information in TR-54: (1) the fill used for each rack and the location of the assumed fuel assemblies to represent possible offsets in the Run 5 simulation (mixed loading case); (2) the modified gaps used in the Run 4 simulation versus the gaps used for the other cases; and (3) a full description of the three additional analyses (Runs 4, 5, and 9). The description should discuss the purpose, approach, results in comparison to the other cases, and clearly state that any subsequent stress evaluations for the racks were based on the worst-case results from all nine runs.

2. Explain why the tabulated displacements in Table 2-10 have not been revised in view of the statement made in Section 2.8.1.4 of TR-54, Revision 2 that the racks did impact the pool walls. The statement implies that the racks displace at least 8.13 cm (3.2 in), which is the available gap between the racks and the pool walls shown on Figure 2-1. In addition, if the displacement of Rack A1, adjacent to the tool storage area, is greater than 8.13 cm (3.2 in) in the south direction, then the 86.36 cm (34 in) referenced clearance shown on Figure 2-1 would need to be revised to ensure that the rack does not displace into the tool storage area boundary.

During the August 2009 audit, the staff discussed with the applicant the need to address the items described above. In a letter dated September 14, 2009, the applicant submitted a revision to its RAI response, addressing these items. The staff confirmed that the necessary changes to TR-54 are included in TR-54, Revision 3. Therefore, RAI-TR54-25 was resolved.

In RAI-TR54-26, the staff requested the applicant address the following: what are the gaps and tolerances for each of the gaps between the fuel to cell wall, rack to rack, and rack to wall? What are the assumed initial locations of the various components (fuel assemblies and each rack) and what is the technical basis for this assumption? Were any studies done for different initial conditions (considering tolerances); if not, explain why. What requirements are in the DCD to ensure that the assumed gaps (considering tolerances) will always be maintained throughout the licensing period?

In a letter dated May 17, 2007, the applicant stated that all gaps between fuel assemblies and cell walls, between racks, and between racks and pool walls are set to match the nominal gaps provided on the Westinghouse Drawing APP-FS02-V2-002, Revision 0, "Discrete Zone Two Region Spent Fuel Rack Pool Layout." An attached table summarized the gap information used in the dynamic analyses. The applicant further stated that fuel is assumed centrally located in the cell. This is conservative since minimizing gap on one or two walls will generally produce a larger hydrodynamic coupling effect. Numerical studies were done on other HOLTEC rack projects; the studies generally showed a small influence on results. A larger influence occurs if the gaps are assumed to be displacement-dependent, rather than always being held constant at their initial value. Neglecting this effect is conservative.

The applicant also stated that, once racks are installed in the SFP, the only way the rack-to-rack and rack-to-wall gaps would change over time would be by the action of a seismic event. COL applicants will have a procedure in place to measure the post-earthquake gaps, to evaluate the acceptability of the configuration, or to take appropriate corrective actions.

During the May 2008 audit, the staff reviewed HOLTEC Report HI-2063523, Revision 2, and confirmed that an additional case (Run 4) was analyzed to consider the effect of the installation tolerances for the nominal gaps. Run 4 corresponded to the base case conditions (Run 1 - COF equal to 0.8, fully loaded racks, integration time step = 1×10^{-5}) but it increased all of the gaps between adjacent racks by 1.27 cm (0.5 in). The results of this additional run demonstrated that the response of the racks was within 8 percent of the base case except for the maximum shear force on the rack pedestal, which was 32 percent higher. The applicant demonstrated that even with this higher pedestal shear force, the pedestal stress factor (ratio of calculated maximum shear stress to allowable shear stress) is equal to 0.092, which is very low. For completeness, the results of Run 4 were included in the design of the racks.

During the August 2009 audit, the staff determined that the drawings showing gaps and tolerances needed to be revised again, to reflect the latest analysis results for revised seismic loading. Subsequent rack design changes and a final revision to the seismic loading, following the August 2009 audit, required further revision of the drawings.

During the June 2010 audit, the staff and the applicant reviewed the final gaps and tolerances. In a letter dated July 20, 2010, the applicant submitted a revised RAI response that addressed the applicable figures in TR-54, Revision 4. The staff reviewed the response, and found the figures to be consistent with the gap and tolerance information in TR-54, Revision 4 and presented at the June 2010 audit.

In RAI-TR54-27, the staff requested the applicant provide more detailed information about how the fluid coupling was calculated and implemented in the AP1000 simulations. Describe the approaches used for fluid coupling of fuel assemblies to fuel cell walls, rack to rack, and rack to pool wall because there would be some differences among these. For the rack to rack and rack to wall fluid coupling, explain how fluid flow was considered horizontally as well as vertically over the top of the racks and flow to the bottom of the rack. Describe and justify any assumptions made in the approach. For example, small vibratory deflections relative to the gaps are probably assumed and the fluid gaps are not updated according to the rack displacements (see Section 2.4 of this report).

In a letter dated May 17, 2007, the applicant provided the following response:

> A mathematical explanation of the manner in which fluid coupling is calculated and implemented in the AP1000 simulations is provided.
>
> The problem to be investigated is shown in Figure TR54-27.1, which shows an orthogonal array of 8 rectangles which represent a unit depth of the 8 spent fuel racks in the AP1000 Spent Fuel Pool. The rectangles are surrounded by narrow fluid filled channels whose width is much smaller than the characteristic length or width of any of the racks. The spent fuel pool walls are shown enclosing the entire array of racks.
>
> The dimensions of the channels are such that an assumption of uni-directional fluid flow in a channel is an engineering assumption consistent with classical fluid mechanics principles. Each rectangular body (fuel rack) has horizontal velocity components U and V parallel to the x and y axes, and the channels are parallel to either the x or y axes. The pool walls are also assumed to move.
>
> It is conservatively assumed that the channels are filled with an inviscid, incompressible fluid. Due to a seismic event, the pool walls and the spent fuel racks are subject to inertia forces that induce motion to the rectangular racks and to the walls. This motion causes the channel widths to depart from their initial nominal values and causes flow to occur in each of the channels. Because all of the channels are connected, the equations of classical fluid mechanics can be used to establish the fluid velocity (hence, the fluid kinetic energy) in terms of the motion of the spent fuel racks.
>
> For the case in question, there are 22 channels of fluid identified.
> Figure TR54-27.2 shows a typical rack (box) with four adjacent boxes with the fluid and box velocities identified. [See Westinghouse response for calculation and figures.]
>
> There are a total of 15 + 8 = 23 equations which can be formally written; one circulation equation, however, is not independent of the other. This reflects the fact that the sum total of the 8 circulation equations must also equal zero, representing the fact that the circulation around a path enclosing all racks is equal to zero. Thus, there are exactly 22 independent algebraic equations to determine the 22 unknown mean velocities in this configuration.
>
> Once the velocities are determined in terms of the rack motion, the kinetic energy can be written and the fluid mass matrix identified using the HOLTEC

> International QA validated pre processor program CHANBP6. The fluid mass matrix is subsequently apportioned between the upper and lower portions of the actual rack in a manner consistent with the assumed rack deformation shape as a function of height in each of the two horizontal directions. The HOLTEC International pre processor program VMCHANGE performs this operation. Finally, structural mass effects and the hydrodynamic effect from fluid within the narrow annulus in each cell between the fuel assembly and the cell wall are incorporated using the HOLTEC International pre processor program MULTI155.

The staff reviewed the RAI response and concluded that the response only addressed how fluid coupling of rack to rack and rack to wall is calculated and implemented in the AP1000 simulation. As requested in the RAI, the applicant should also provide a description of the approach used for simulating the fluid coupling of the fuel assemblies to the fuel cell walls. Due to the complex nature of the fluid coupling approach used for the rack to rack and rack to wall, and the use of several HOLTEC in-house computer codes, how has the approach been verified (e.g., test data or alternate analytical methods with known solutions)?

In a letter dated April 18, 2008, the applicant revised its response, replacing the last two paragraphs, and adding references as shown below:

> "Once the velocities are determined in terms of the rack motion, the kinetic energy can be written and the fluid mass matrix identified using the HOLTEC International QA validated pre processor program CHANBP6. The fluid mass matrix is subsequently apportioned between the upper and lower portions of the actual rack in a manner consistent with the assumed rack deformation shape as a function of height in each of the two horizontal directions. The HOLTEC International pre processor program VMCHANGE performs this operation.
>
> "The approach used for fluid coupling between the fuel assemblies and the cell walls is presented in Reference 2, and it is based upon Fritz's classical two body fluid coupling model (Reference 3). References 2 and 3 were previously provided to the NRC as part of the April 9, 2007 RAI response submittal (Reference 4 - Westinghouse Letter DCP/NRC1860). The structural mass effects and the hydrodynamic effect from fluid within the narrow annulus in each cell between the fuel assembly and the cell wall is incorporated using the HOLTEC International preprocessor program MULTI155."

The staff noted that the fluid coupling between the fuel assemblies and the cell walls is described in a paper "Seismic Responses of Free Standing Fuel Rack Constructions to 3 D Motions," by Soler, A.I. and Singh, K.P., Nuclear Engineering and Design, Volume 80, pp. 315 329 (1984), and it is based upon Fritz's classical two-body fluid coupling model presented in "The Effects of Liquids on the Dynamic Motions of Immersed Solids," by Fritz, R.J., Journal of Engineering for Industry, Trans. of the ASME, February 1972, pp. 167-172. This methodology for modeling the fluid coupling effects has been accepted for predicting the response of spent fuel racks in NUREG/CR-5912. It has been used in past licensing of spent fuel racking, and reviewed and approved by the NRC. Therefore, RAI-TR54-27 was resolved.

The load combinations specified in Table 2-5 of TR-54 and Table 9.1-1 (markup version of the DCD provided with the subject report) did not match NUREG-0800 Section 3.8.4, Appendix D criteria. The staff issued RAI-TR54-29, which reads as follows:

a. No load combinations are specified for the spent fuel racks corresponding to service Level A.

b. Temperature conditions T_o and T_a are not included in Table 2-5; however, they are included in the markup DCD Table 9.1-1. A footnote in the markup of DCD Table 9.1-1 states that "For the faulted load combination, thermal loads will be neglected when they are secondary and self limiting in nature and the material is ductile. In freestanding spent fuel racks, thermal effects mainly affect the temperature that is used in specifying the allowable stress and Young's Modulus." Based on this statement:

   (i) Regarding the first quoted sentence above, Table 2-5, Load Combination corresponding to service levels A and B (which are not the faulted condition) should include T_o.

   (ii) Regarding the last quoted sentence above, SRP 3.8.4, Appendix D indicates that thermal loads due to temperature effects and temperature gradients across the rack structure need to be considered. Temperature gradients can occur due to differential heating effects between one or more filled cell(s) and one or more adjacent empty cell(s). The stresses from these types of thermal loads should be considered because they can still lead to localized failure of the structure. When responding to this, consider temperature loads due to normal and accident conditions, as noted in your Table 9.1-1 and SRP 3.8.4, Appendix D.

c. Table 2-5 in the report and DCD Table 9.1-1 indicate that the load term P_f is the uplift force on the rack caused by a postulated stuck fuel assembly accident condition or the force developed on the rack from the drop of a fuel assembly during handling to the top of the rack or the baseplate through an empty cell. SRP 3.8.4, Appendix D, separates these two accident events into Pf for the uplift force event and Pd for the drop load event. This is necessary because SRP 3.8.4, Appendix D specifies that the acceptance limits for these two events (in combination with deadweight + live load + thermal) are different.

d. Table 2-5, last load combination with E', does not provide the Service Limit. If the same Service Limit, $D^{(1)}$, as indicated in the load combination above the last load combination was intended, then explain whether the functionality capability requirement in footnote (1) (which is applicable to only the new racks) is in addition to or in-place of Level D limits.

In a letter dated May 17, 2007, the applicant provided the following response:

Table 2-5 of technical report (TR)-54 and DCD Table 9.1 1 will be revised as follows (which is derived from Appendix D to SRP Section 3.8.4):

a. Table 2 5 of the subject report and DCD Table 9.1 1 will be modified to specify the load combinations D + L and D + L + T_o for Service Level A, as shown above.

b. (i) Table 2 5 of the subject report will be modified to include T_o for Service Levels A and B, as shown above.

(ii) The temperature gradients across the rack structure caused by differential heating effects between one or more filled cells and one or more adjacent empty cells are considered. The worst thermal stress field in a fuel rack is obtained when an isolated storage location has a fuel assembly generating heat at maximum postulated rate and the surrounding storage locations contain no fuel. This secondary stress condition is evaluated alone and not combined with primary stresses from other load conditions. A thermal gradient between cells will develop when an isolated storage location contains a fuel assembly emitting maximum postulated heat, while the surrounding locations are empty. A conservative estimate of the weld stresses along the length of an isolated hot cell is obtained by considering a beam strip uniformly heated by 10.00 °C (50 °F), and restrained from growth along one long edge. The 50 °F temperature rise envelops the difference between the maximum local spent fuel pool water temperature (174 °F) inside a storage cell and the bulk pool temperature (140 °F) based on the thermal hydraulic analysis of the spent fuel pool. The cell wall configuration considered here is shown in figure below.

c. The definition of P_f in Table 2-5 of the subject report and DCD Table 9.1-1 is incorrect. The referenced tables will be revised to clearly distinguish between P_f and F_d.

d. Level D service limits apply to load combination D + L + Ta + E'. Per Appendix D of SRP Section 3.8.4, the functional capability of the fuel racks should be demonstrated for the accidental drop event (D + L + F_d). This requirement is in place of the Level D service limits since it is recognized that the rack may sustain permanent damage due to the impact force, and therefore it may not be possible to meet Level D service limits at all locations within the rack. The functional capability of the spent fuel racks is generally defined as the continued ability of rack to store spent fuel assemblies in a subcritical arrangement.

The staff reviewed the RAI response, and concluded that several additional items needed to be addressed by the applicant. In a letter dated April 18, 2008, the applicant provided a revised response to this RAI, which provided additional explanations.

The staff reviewed the revised response, and noted that proposed TR-54 Table 2-5 and DCD Table 9.1-1 include the missing load combination. In addition, the applicant: (1) explained why a secondary stress is not combined with primary stresses from other loads; (2) provided the calculation for the analysis of thermal stresses in the rack cell considering the differential heating between a cell containing a fuel assembly emitting maximum postulated heat and empty cells surrounding the heated cell; and (3) explained that the shear stress in the weld is caused by thermal loading, which is classified by the ASME Code as a secondary stress. In summary, the staff accepted the applicant's explanations pending submittal of revisions to TR-54 and the DCD. The staff subsequently confirmed that TR-54, Revision 2, and DCD Revision 17 had been appropriately revised. Therefore, RAI-TR54-29 was resolved.

The staff noted that TR-54 does not address seismic-induced sloshing effects. In RAI-TR54-31, the staff requested that the applicant provide a description of the sloshing calculation approach and results for both horizontal directions.

In a letter date May 17, 2007, the applicant stated that "sloshing" may be defined as the dynamic behavior and associated load of the water produced by wave-like motion at the surface of the pool. TID-7024, "Nuclear Reactors and Earthquakes," Chapter 6, is commonly used to evaluate the dynamic response of the water within the SFP. Figure 6.2(a) of TID-7024 depicts the two masses of water that the total bulk is considered to be split into, as described in the text. The upper portion of the water, denoted in the figure as "water in motion," produces convective forces and the lower portion of the water, denoted as "constrained water," produces impulsive forces. The latter bulk of water has an associated mass (identified as weight W0) and is effectively a rigid body that moves along with the tank (refer to Figure 6.1 and the first paragraph of Section 6.4). The horizontal force produced by this mass of water when accelerated by the earthquake acts at a height of h_o from the bottom of the tank. This parameter is determined in the table given at the end of Section 6.3 to be equal to 3/8 times the height of the fluid. This height is not dependent upon the magnitude of the earthquake. For the SFP, the water depth is approximately 12.19 m (40 ft) and the height h_o would be 4.57 m (15 ft) from the bottom. Since the impulsive force acts at the approximate centroid of the rigid water mass, the top elevation of this bulk of water is above this point. As the racks are approximately 515.6 cm (203 in) tall, which is only slightly higher than the height h_o, the racks reside in the impulsive water mass at the bottom of the pool and the sloshing portion of the water is above this elevation. Therefore, seismic sloshing of the SFP water does not influence the dynamic response of the spent fuel racks in either horizontal direction.

The staff concluded that the applicant's description of the sloshing effect in the SFP, based on using the method presented in TID-7024, Chapter 6, demonstrates that the racks reside in the impulsive water mass region of the pool and the sloshing portion of the water is above the top of the racks. Therefore, RAI-TR54-31 was resolved.

Section 2.3.4.3 of TR-54, fourth bullet, develops the faulted allowable maximum weld stress for the weld material. In RAI-TR54-33, the staff asked the applicant why an allowable maximum weld stress based on the base metal isn't also developed. Normally welds are checked for both weld material and base metal, as was done for Levels A and B in Section 2.3.4.1.

In a letter dated May 17, 2007, the applicant stated:

> The required capacity evaluation for Level A conditions are presented below using the material properties associated with the material. Using the ASME Code allowable strengths for weld and base metal in Subsection NF, the shear capacities are:
>
> V(base) = (0.4Sy)Al; V(throat) = (0.3Su)(0.707Al);
> V(throat)/V(base) = 0.2121Su/(0.4Sy) = 0.53025Su/Sy, where Su = ultimate strength of weld material (assumed equal to that of the base metal for purposes of this calculation); Sy = yield strength of base metal; Al = fillet weld leg area; At = fillet weld throat area = 0.707Al.
>
> The above result for Level A conditions shows that the weld throat controls the capacity only if 0.53025Su < Sy. For the AP1000 spent fuel racks, Su=66.2 ksi; Sy=21.3 ksi at temperature, so that V(throat)/V(base) = 1.648, indicating that

base shear capacity controls the joint for a Level A event. For Levels B, C, and D, the joint capacities are simply increased by a factor so that the determination of the governing section remains the same.

Appendix F of the ASME Code does not explicitly require weld calculations for Level D events. If, however, the weld capacity evaluations are performed using material strengths inferred by certain sub sections of Appendix F, HOLTEC evaluates the capacity of the weld throat by using the amplifier 1.8 on the Level A capacity to obtain V(throat) = 1.8 (0.2121SuAl) = 0.38278SuAl

ASME Code Appendix F contains the following subsections that refer to allowable strengths for shear calculations. Using the 1998 Edition,
F-1331 - Criteria for Components (F-1331.1(d)) - The average primary shear stress across a section loaded in pure shear shall not exceed 0.42Su.
F-1332 - Criteria for Plate and Shell Type Supports (F-1332.4 Pure Shear) - The average primary shear stress across a section loaded in pure shear shall not exceed 0.42Su.
F-1334 - Criteria for Linear Type Supports (F-1334.2 Stresses in Shear) - The shear stress on the gross section shall not exceed the lesser of 0.72Sy and 0.42Su. Gross section shall be determined in accordance with NF 3322.1(b). [Note that Code reference to NB 3322.1(b) is a typo as the referenced NB section has nothing to do with section evaluation.]

It is stipulated that F-1334.2 is intended for setting limits for the shear stress in the base metal of gross sections associated with steel structural members and should not be applied to any weld calculation (as can be inferred by the title of Subsection NF-3322 B Design Requirements for Structural Steel Members). Even if one accepts that there is an implied requirement in Appendix F to check weld capacity for Level D events, the appropriate base metal shear stress limit should be 0.42Su (viz. F-1331.1(d), F-1332.4, or F-1334.2), which would therefore give the capacity of the base metal as V(base) = 0.42SuAl.

V(throat)/V(base) = 0.911 indicating that weld throat shear capacity always controls the joint for a Level D event independent of the material. This is why only the weld throat is checked when examining welds in the Level D configuration.

The staff reviewed the above explanation and found that it demonstrates that following the requirements in Appendix F for components and supports, and using the ultimate strength value and yield strengths for the materials of the spent fuel racks, welded joints are governed by the weld throat shear capacity, not the base metal capacity. That is why only the weld throat was checked for Level D when examining the structural adequacy of welds. The staff found this acceptable. Therefore, RAI-TR54-33 was resolved.

Section 2.3.5 of TR-54 discusses dimensionless stress factors. It states that "R1 is the ratio of direct tensile or compressive stress on a net section to its allowable value (note pedestals only resist compression)." In RAI-TR54-34, the staff requested the applicant explain why this indicates that pedestals only resist compression, since horizontal forces are also generated due to friction during a seismic event. These forces could be quite high and also would introduce shear and moments into the pedestal and rack structure.

In its response, the applicant stated that Section 2.3.5 of the report defines seven stress factors (R1 through R7), which correspond to the ASME Code Section III, Subsection NF stress limits for Class 3 components. R1 is defined as the ratio of direct tensile or compressive stress on a net section to its allowable value. Since the spent fuel racks are freestanding, the net cross section of the support pedestals can only be subjected to direct compressive stress. This is the explanation for the note in parentheses. The applicant further stated that horizontal forces are generated due to friction between the support pedestals and the SFP floor and that these forces produce shear and bending stresses in the pedestals. The shear and bending stresses in the support pedestals, as well as the combined compression and bending stress, are measured by the other six stress factors (i.e., R2 through R7), which are defined in Section 2.3.5 of this report. The staff reviewed the RAI response and concluded that the explanation is acceptable. Therefore, RAI-TR54-34 was resolved.

Section 2.8.1.4 of TR-54 describes the impact loads and states that these loads do not result in damage to the racks that would prevent retrievability. In RAI-TR54-35, the staff requested the applicant confirm that the acceptance criteria for these impacts include both retrievability and meeting the stress limits for Level D in accordance with ASME Code, Section III, Subsection NF; and to provide the stress ratios for the most critical cells adjacent to the worst case impact.

In a letter dated May 17, 2007, the applicant stated that the ability to retrieve the fuel is based solely on evaluating the rack structure to show that there is no instability that would collapse the cell. Subsection NF stress limits for Level D do not apply to the local stress state in the impacted cells because: (a) the fuel racks are analyzed as linear type supports (i.e., beam type members) in accordance with Appendix D of NUREG-0800 Section 3.8.4; and (b) rack to rack impact loads near the top of the rack produce secondary stresses, for which there is no prescribed limit in ASME Code, Section III, Subsection NF for Level D. Away from the point of impact, the rack to rack impact loads do produce primary bending and shear stresses in the rack beam, which are reflected in the maximum stress factors reported in TR-54, Table 2-9.

The staff reviewed the response, and requested additional information explaining what acceptance criteria are used to ensure that the fuel assemblies can be retrieved following the impact: If a quantitative stress limit criteria is not utilized, then what is the specific criterion used (e.g., are the cross-sectional opening dimensions in each cell checked for any permanent deformation that would infringe on the fuel assembly outside dimension)?

In a letter dated April 18, 2008, the applicant stated that in order to ensure fuel retrievability is maintained, the impact loads at the rack top elevation are compared against 2/3 of the critical buckling load for the cell walls, as required by Table NF-3523(b)-1 of the ASME Code for primary plus secondary stresses. The critical buckling load calculation is included in HOLTEC Report No. HI-2063523, "Spent Fuel Rack Structural/Seismic Analysis for Westinghouse AP1000," Revision 0. The impact load is assumed to spread uniformly over a 15.24 cm (6 in) vertical span of the cell wall, which is equal to the minimum length of the intermittent cell to cell welds. The average compressive stress in the cell walls due to the maximum rack-to-rack impact load is 132 MPa (19,120 psi). This stress is less than two thirds of the critical buckling load. Therefore, the spent fuel rack design meets the requirements of Table NF-3523(b)-1 of the ASME Code for the primary plus secondary stress category.

The staff concluded that the calculation adequately demonstrated that the stress due to the maximum impact load is below 2/3 of the critical buckling load, and that the spent fuel racks meet the requirements of Table NF-3523(b)-1 for primary plus secondary stress category, which

covers buckling. The staff found this approach to demonstrate retrievability acceptable. The staff requested the applicant expand the limited one-sentence conclusion presented in Section 2.8.1.4 of the TR-54, to summarize the information in the RAI response and the buckling calculation. The staff subsequently reviewed TR-54, Revision 2, and related sections in DCD Revision 17, to confirm the inclusion of additional descriptive information. Therefore, RAI-TR54-35 was resolved.

Some of the information provided in Section 2.8.2 (Rack Structural Evaluation) and Tables 2-9 through 2-15 (stress results) of TR-54, Revision 0, was not clear. The staff issued RAI-TR54-36 which reads as follows:

a. Section 2.8.2.1, 2nd paragraph, indicates that the tables also report the stress factors for the AP1000 Spent Fuel Storage Racks cellular cross section just above and below the baseplate. This implies that the fuel cells continue below the baseplate. Explain.

b. The same paragraph refers to "pedestal five in the first sheet of the summary tables for each simulation (that is, 9.M.0 where M stands for run number)." Explain what this means since the tables do not reflect this terminology.

c. The same paragraph refers to "ensures that the overall structural criteria set forth in Subsection 2.2.3 are met." Structural criteria are not presented in Subsection 2.2.3.

d. Section 2.8.2.2 a., refers to a stress factor of 2.1516 which it states is given in the tables. However, no such stress factor is given, please explain. Also, are all cells welded to the baseplate on all four sides?

e. Section 2.8.2.2 a., first bullet, calculates the stress in the weld, connecting the cell walls to the baseplate, equal to 25,047 psi; however, Table 2-12 shows a smaller (maximum) weld stress of 22,647. Explain.

f. Section 2.8.2.2 b. indicates that a separate finite element model is used to check the baseplate to pedestal welds. Provide a short description of the model, computer code, loading, and location of the maximum tabulated stress in the weld referred to in Table 2-14.

g. Section 2.8.2.2 c. indicates that for calculation of cell welds, the fuel assemblies in adjacent cells are conservatively calculated by assuming that the fuel assemblies in adjacent cells are moving out of phase with one another. It then states that cell to cell weld calculations are based on the maximum stress factor from all runs. However, elsewhere in the report it was stated that all of the fuel assemblies in the simulation are assumed to vibrate in phase. Provide more information to explain this. Also, this paragraph indicates that both the weld and the base metal shear results (for cell to cell) are reported in Table 2-14; however, Table 2-14 is labeled baseplate to pedestal welds. If reference was intended to Table 2-15, then note that Table 2-15 provides the shear stress only for the base metal.

h. Section 2.8.2.3 refers to Tables 2-9 through 2-14 for limiting pedestal thread shear stresses for every pedestal. These tables do not seem to apply to pedestal thread shear stress. Therefore, clarify or correct this information.

i. Table 2-9, Summary: identify what rack component/element applies to each of the column headings (i.e., Max. Stress Factor, Max. Shear Load, Max. Fuel to Cell Wall Impact). Similarly, for the other tables, identify what rack component/element the table applies to (e.g., Tables 2-13 and 2-15 are missing this information).

j. Table 2-10 provides maximum rack-to-rack displacements relative to the floor. Also provide maximum & minimum relative displacements to the walls.

k. Why are results for "Run 1 and 2" given for some tables and not others? Both should be provided or an explanation should be given why they are included for some tables and not for others.

l. Table 2-15, why is this table labeled "Allowable Shear Stress ..." versus the labeling of other tables and why is it labeled Level D, versus other tables where there is no indication of Levels? All tables should identify which load level they apply to.

In a letter dated June 14, 2007, the applicant provided the following response:

a. The fuel cells do not continue below the baseplate. Stress factors are computed just above the baseplate, where the fuel cells are welded to the baseplate and just below the baseplate where the support pedestals are welded. Section 2.8.2.1 (2nd paragraph, 2nd sentence) will be revised as follows:

> The tables also report the stress factors for the AP1000 Spent Fuel Storage Racks cellular cross section just above the baseplate.

b. The computer code DYNAPOST, which is listed in Table 2-8, computes the stress factors for the four support pedestals and for the cellular structure just above the baseplate based on the time history analysis results. For convenience, these five locations are identified as pedestal numbers 1 through 5 in the DYNAPOST output tables, which are not included in technical report APP-GW-GLR-033. Therefore, the sentence, "The locations above the base plate ... are referred to as pedestal five in the first sheet of the summary tables for each simulation (that is, 9.M.0 where M stands for run number)." is not relevant to the report and will be deleted.

c. The reference to Subsection 2.2.3 is a typo. The correct reference is Subsection 2.3.3.

d. The factor of 2.1516 is not provided in the tables as stated in text. Section 2.8.2.2 a. (2nd paragraph) will be revised as follows:

> Weld stresses are determined through the use of a simple conversion (ratio) factor (based on area ratios) applied to the corresponding stress factor in the adjacent rack material. This conversion factor is developed from the differences in base material thickness and length versus weld throat dimension and length:
>
> All fuel cells are welded to the baseplate on all four sides.

e. The correct stress in the weld is 25,047 psi. Table 2 12 will be revised to change 22,647 psi to 25,047 psi, as shown [in Table 2-12 of the RAI response].

f. The finite element code ANSYS is used to resolve the tension and compression stresses in the pedestal weld due to the combined effects of a vertical compressive load in the pedestal and a bending moment caused by pedestal friction. The compression interface between the baseplate and the pedestal is modeled using contact elements. The perimeter nodes on the pedestal are connected to the baseplate by spring elements in order to simulate tension in the weld. The maximum instantaneous friction force on a single pedestal from the rack seismic analysis is conservatively applied to the finite element model in the horizontal x and y directions simultaneously, along with the concurrent vertical load, at the appropriate offset location. The perimeter nodes on the pedestal are restrained to move only in the vertical direction so that the spring elements only resist bending. The limiting ANSYS results are combined with the maximum horizontal shear loads to obtain the maximum weld stress. The maximum weld stress reported in Table 2-14 occurs at the corner of the pedestal where the tensile stress in the weld due to bending is at its maximum.

g. All stored fuel assemblies within a rack are assumed to rattle in phase for the seismic analysis of the spent fuel racks using the HOLTEC proprietary computer code MR216 (a.k.a. DYNARACK). This analysis yields the maximum impact force between a single fuel assembly and the surrounding cell walls. When evaluating the weld connection between adjacent storage cells, the maximum fuel to cell impact force from the dynamic analysis is conservatively multiplied by a factor of 2 to consider out of phase fuel rattling. The reference to Table 2-14 in Section 2.8.2.2 c is incorrect. The shear stress results for the cell to cell weld connection are not provided in Table 2-14 or Table 2-15. The shear stress in the cell to cell weld and the adjacent base metal are 11,646 psi and 8,235 psi, respectively. The allowable stress limits are 35,748 psi and 18,000 psi, respectively. Tables 2-16 and 2-17 (see below) will be added to technical report APP-GW-GLR-033 to provide the shear stress results for the cell to cell weld and the adjacent base metal, respectively.

h. The reference to "Tables 2-9 through 2-14" in Section 2.8.2.3 is incorrect. The first sentence in Section 2.8.2.3 will be revised as follows: "Table 2-15 provides the limiting thread stress under faulted conditions."

i. In Table 2-9, the "Max. Stress Factor" column applies to the rack cell structure. The "Max. Vertical Load" and "Max. Shear Load" columns apply to a single rack pedestal. The "Max. Fuel to Cell Wall Impact" column provides the maximum impact force between a single fuel assembly and the surrounding cell wall at any of the five rattling fuel mass elevations (refer to Figure 2-5 of the report).

   Table 2-13 applies to the base metal adjacent to the baseplate to cell welds. Table 2-15 applies to the pedestal internal threads.

j. Table 2-10 provides the maximum displacement in any direction (x or y) for all racks, relative to the floor. In other words, the rack displacements in Table 2-10 are the bounding displacements for all rack to rack and rack to wall gaps. The results in Table 2-10 also represent the maximum rack displacements relative to the pool walls since the SFP structure is assumed to be rigid for the purpose of the rack seismic analysis (i.e., the SFP floor and walls displace equally). The minimum rack displacement relative to the SFP walls (which is interpreted as maximum distance that a rack displaces away from the SFP walls) is also bounded by the results in Table 2-10, since the reported displacements are the maximum (absolute value) displacements for all racks.

k. The stress results in Tables 2-12 through 2-15 are the maximum values from Run 1 and Run 2.

l. Table 2-15 should be labeled "Pedestal Thread Shear Stress" instead of "Allowable Shear Stress for Level D". The allowable stresses reported in Tables 2-12 through 2-15 are Level D stress limits since the design basis ASB99 earthquake is a faulted condition (Level D).

The staff reviewed the response, and concluded that for parts (a) through (d), (h), (i), (k), and (l), the clarifications and editorial corrections are acceptable; however, revision of TR-54 would be required as noted in the response. For part (e), correction of the stress result in Table 2-12 is acceptable; however, with this correction the safety factor noted in the table is no longer correct and needs to be revised. For part (f), the description of the separate finite element model to check the baseplate to pedestal welds is acceptable and should be included in the next revision of TR-54. For part (g), the response explained why in the dynamic analyses the in-phase assumption for fuel assembly motion was utilized, while for the design of the welds between adjacent cells, the out-of-phase motion of fuel rattling was used. This approach is considered to be acceptable because it would maximize the rack motion and impact forces in the dynamic analyses while the out of phase motion of the fuel assembly would be more conservative for the evaluation of the welds between adjacent cells. However, the shear stress results refer to the wrong table, as indicated in the response, and two new tables have been developed which need to be inserted in TR-54. For part (j), additional clarification is needed to explain: (1) whether the individual maximum displacements are at the base or at any elevation of the rack model including the top of the rack since the rotation of the racks about a leg would amplify the horizontal motion from the base; and (2) how do these displacements compare to the initial available gap to the pool walls and rack-to-rack gaps at the top and bottom. This would demonstrate whether impacts with the pool wall occur.

During the May 2008 audit, the staff reviewed TR-54, Revision 1, and concluded that the applicant's responses to all parts of this RAI are technically acceptable, and appropriately addressed in the TR and the DCD. Therefore, RAI-TR54-36 was resolved.

Section 2.8.4 of TR-54 indicates that this section presents evaluations for potential cell wall buckling and the secondary stresses produced by temperature effects. The staff noted that the description of secondary stresses produced by temperature effects is not included in this section. In RAI-TR54-37, the staff requested the applicant add this information to the report, and to confirm that the R5 stress factor used for the buckling calculation includes the worst impact forces generated, including the impacts at the top of the racks.

In a letter dated May 17, 2007, the applicant stated that the secondary stresses produced by temperature effects (an isolated hot cell) were inadvertently omitted, and that TR-54 Section 2.8.4 would be revised to include an evaluation of secondary stresses produced by temperature effects. The stress factor R5 is a stress factor that is used to get the vertical stress near the base of a corner cell and includes the effect of lateral impact forces at the top of the rack. That is, at any instant the rack is under beam action so that a lateral impact load at the top of a rack develops a vertical load at the base of the rack as the rack resists rocking.

During the May 2008 audit, the staff reviewed TR-54, Revision 2, and confirmed that the evaluation of the secondary stresses produced by temperature effects was included in Section 2.8.4.2 of the TR. The thermal analysis did not consider the contribution from seismic loads. The applicant explained that this was done because the extreme thermal analysis was based on the conservative case of a single cell with a fuel assembly surrounded by all empty cells. The seismic stress contribution for this rack configuration would be insignificant. The applicant demonstrated this by presenting the results for Run 9, which considered all racks empty. These results showed that the maximum stress factor was 0.074, which is extremely small compared to 1.0. Therefore, RAI-TR54-37 was resolved.

The staff noted that the computer code MR216 (a.k.a. DYNARACK) as well as the other computer analysis codes should have complete validation documentation, available for review during an audit. In RAI-TR54-39, the staff inquired if any of the computer codes have been previously reviewed and approved by the staff on other licensing applications, for the same version of the code.

In its response dated April 9, 2007, the applicant stated that all computer analysis codes used to perform the seismic analysis of the spent fuel racks have been validated in accordance with HOLTEC's 10 CFR 50 Appendix B quality assurance program. The validation documentation will be available for review during the audit. The validation documentation for the computer code MR216 has been previously submitted by HOLTEC International to the NRC staff for review and approval several times. Most recently it was reviewed by the NRC in 1998 in Docket Number 50-382 for the Waterford 3 Steam Electric Station.

During the April 2007 audit, the applicant indicated that the version of the MR216 code previously used on the Waterford 3 Steam Electric Station and the version used for the AP1000 are identical, except that the code was revised to accept an additional input at the top of the structure being analyzed. This change has been validated; however, this feature is not used in the AP1000 analyses.

During the October 2007 audit, the staff reviewed the DYNARACK computer validation package. The validation package provided for staff review was the HOLTEC I.D. No. HI-91700 (Generic), "DYNARACK Validation Manual," Revision 1, approved January 28, 1998. The approach used to validate DYNARACK was to demonstrate that it meets the validation requirements of NUREG-0800 Section 3.8.1. The procedure followed for the validation of the code was Section II.4.F (in the current March 2007 version) of NUREG-0800 Section 3.8.1. A series of validation problems were performed and described in the validation manual, demonstrating that criteria (ii) and (iii) in Section II.4.F were met. The staff reviewed a representative test problem included in the validation package. Based on this review, and the accepted use of DYNRACK on a number of other rack analyses for nuclear power plant licensing submittals, the staff concluded that this validation package for DYNARACK is acceptable.

The staff also reviewed the computer program validation package for the DYNAPOST code. The validation package is HOLTEC Report No. HI-971648, "QA Validation of Program DYNAPOST for Generic," Revision 1, approved October 31, 1997. This program was developed to post process the results obtained from the whole pool multi-rack analysis performed with DYNARACK. Based on the review, the staff concluded that the validation package for DYNAPOST is acceptable.

The staff concluded that the computer codes used for the seismic response analysis of the fuel racks have been validated and supporting documentation exists. Therefore, RAI-TR54-39 was resolved.

In RAI-TR54-40, the staff requested the applicant explain what provisions are provided for performance of inservice examination of the rack, as specified in 10 CFR 50.55a(g)(3) for ASME Code Class 3 component supports.

In a letter dated May 17, 2007, the applicant stated that the spent fuel racks are passive structures in the SFP. They operate in a relatively mild environment compared to reactor coolant system (RCS) primary components. There are no moving parts on the spent fuel racks, and they do not require any instrumentation. Therefore, there is no compelling need to perform inservice examination of the spent fuel racks. However, the spent fuel racks can be accessed from above by way of an empty storage cell location(s) to enable the performance of inservice examination, as mandated by 10 CFR 50.55a(g)(3) for ASME Code Class 3 component supports. At the base of each storage cell (except at the four designated lifting locations), there is a 15.24 cm (6 in) diameter thru-hole in the baseplate, which provides access below the baseplate. Also, access below the baseplate can be gained from the area of the SFP that does not contain spent fuel racks. In summary, the spent fuel racks are designed to provide access to all surfaces that may come in contact with spent fuel assemblies and to the support pedestals beneath the baseplate, to support inservice examinations as needed.

The staff concluded that adequate accessibility has been provided to accommodate inservice inspection of the spent fuel racks. Therefore, RAI-TR54-40 was resolved.

Section 2.1.1 was revised in TR-54, Revision 2, to state that "Per DCD Subsection 3.7.5.2, COL applicants will prepare site-specific procedures for activities following an earthquake. An activity will be to address measurement of the post-seismic event gaps between spent fuel racks and to take appropriate corrective actions." This statement was previously in Section 2.9 "Conclusions," in TR-54, Revision 0 and Revision 1, and was moved to Section 2.1.1 in TR-54, Revision 2.

The staff noted that DCD Section 3.7.5.2 does not discuss the need for COL applicants to prepare site-specific procedures for checking the gaps between the fuel racks following an earthquake. In RAI-SRP9.1.2-SEB1-04, the staff requested the applicant explain how this requirement is conveyed to the COL applicants and to identify the COL action item, inspections, tests, analyses, and acceptance criteria (ITAAC), or other interface requirement that addresses this.

In its response dated February 24, 2009, the applicant submitted a proposed markup to Section 3.7.5.2 of the DCD, requiring COL applications to include in their post-earthquake a procedure to check the gaps between racks and between the racks and walls, and to take appropriate actions to restore the design-basis gaps. The staff found this acceptable. In a

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

Section 2.4 "Assumptions" was revised in TR-54, Revision 2, to state that "Modeling the total effect of n individual fuel assemblies rattling inside the storage cells in a horizontal plane as one lumped mass at each of five levels in the fuel rack is a conservative assumption. Thus, the effects of chaotic fuel mass movement are incorporated into the analysis by introducing a fuel ratio factor of 0.75 (75 percent of the fuel weight is used in the analysis)."

The staff noted that the use of a 0.75 fuel ratio factor is a departure from prior revisions of TR-54, where a fuel ratio factor of 1.0 was assumed. In RAI-SRP9.1.2-SEB1-05, the staff requested the applicant provide a detailed technical basis for utilizing a fuel ratio factor of 0.75.

During the August 2009 audit, the staff and the applicant discussed this issue in depth. The applicant agreed to conduct additional analysis to justify a fuel ratio factor less than 1.0. In a letter dated November 11, 2009, the applicant reported that it could not justify a fuel ratio factor less than 1.0. Consequently, this approach to reducing the seismic loads was abandoned. The applicant implemented a number of rack design changes to demonstrate adequacy for the loads based on a 1.0 fuel ratio factor. The staff confirmed that in TR-54, Revision 3, reference to fuel ratio factor was deleted. Therefore, RAI-SRP9.1.2-SEB1-05 was resolved.

During the review of TR-54, Revision 2, the staff noted that Section 2.8.1.4, "Rack-to-Rack and Rack-to-Wall Impacts," was revised, and indicated that the re-analysis of the spent fuel racks, to incorporate the updated seismic loading and revisions in the design of the racks, resulted in two rack-to-wall impacts: in Run 5, rack A1 impacts the west wall at a force of 45,690 lb; and in Run 4, rack B4 impacts the north wall at a force of 67,800 lb. In RAI-SRP9.1.2-SEB1-06, the staff requested the applicant describe in detail how these additional impact loads had been considered in the design of the fuel pool structure (including the liner) and in the design of the fuel racks, and also to identify where this would be described in the AP1000 DCD.

In a letter dated June 12, 2009, the applicant submitted its initial response to RAI-SRP9.1.2-SEB1-06. The staff and the applicant discussed the response at the August 2009 audit, and concluded that the rack-to-rack and rack-to-wall impact loads could increase, depending on the final resolution of the fuel ratio factor issue discussed in RAI-SRP9.1.2-SEB1-05 above. In a letter dated November 11, 2009, the applicant submitted a revised response, documenting the increased impact loads and describing the analysis method used to evaluate cell wall buckling at the top of the rack for the worst case impact load. The applicant stated that the details of the analysis were included in TR-54, Revision 3 (November 2009). The staff reviewed TR-54, Revision 3. The applicant conducted a nonlinear analysis using the LS-DYNA computer code; the results showed that the required safety factor of 1.5 is achieved before failure of the cell wall. The staff determined that the applicant's analysis constituted an acceptable method to check the adequacy of the spent fuel rack design for the worst case top impact load. During the June 2010 audit, the staff audited the Westinghouse/HOLTEC calculation for the impact analysis, and discussed the results with the applicant. The staff found that the calculation is consistent with the information in TR-54, Revision 3, and is acceptable. The applicant also identified several DCD changes to describe the impact analysis.

In the November 11, 2009 RAI response, the applicant also stated that the impact load between the rack and the SFP wall increased from 36,741 kg (81,000 lb) to 163,293 kg (360,000 lb), but that this had only a marginal effect on the required steel liner thickness. The staff noted that it

would be necessary to audit the applicable calculations, before it could accept this result. The staff attempted to audit the applicable calculations at the June 2010 fuel rack audit; however, the applicant's staff was unable to answer staff questions on the calculations.

At the DCD Section 3.8 regulatory audit conducted during the week of June 28, 2010, the staff again attempted to audit the AP1000 calculations that evaluate the spent fuel rack impact forces on the SFP walls. Again, the applicant was not able to address the questions raised by the staff.

Therefore, the staff requested the applicant to augment its prior RAI response to address the following: (1) describe how the tri-axial state of stress in the impacted faceplate has been addressed in design check, when considering the impact load in addition to other concurrent loadings; and (2) provide a comparison between the load combination with seismic load only and the load combination with seismic load and impact load, in order to confirm that the impact load is insignificant.

At the structural issues regulatory audit conducted August 18-20, 2010, the staff and the applicant discussed the results of the applicant's analyses to address the staff's questions, and the applicant's initial draft of the RAI response. The staff requested several additions to the RAI response, to which the applicant agreed. In a letter dated August 25, 2010, the applicant formally submitted its revised response, which included: (1) the calculation for the third principal stress in the faceplate of the spent fuel wall, due to the rack impact load on the wall, and a comparison of the stress intensities with and without the third principal stress, at several locations in the face plate; and (2) a comparison of the element member forces at several critical locations on the SFP wall between the load combination with seismic load only and the load combination with seismic load and impact load.

The staff reviewed the revised response and found it acceptable because the calculation results demonstrate that: (1) the effect of the increased impact force from a spent fuel rack onto the SFP walls is insignificant, for the design of SFP wall; and (2) the design of the spent fuel wall still meets the specified acceptance criteria when the impact load is included with other concurrent loads. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 2.8.4.1, "Cell Wall Buckling Evaluation," was revised in TR-54, Revision 2. A different buckling equation and different boundary conditions are indicated. The rectangular flat plate model representing the lower cell wall region is now assumed to be clamped on all 4 edges. Even with the assumption of clamped on all 4 edges; a very small safety margin against buckling is indicated in Revision 2. The staff determined that only one edge can truly be treated as clamped, and the other 3 edges can rotate somewhat due to the flexibility of the adjacent sections.

In RAI-SRP9.1.2-SEB1-07, the staff requested the applicant: (1) provide the technical basis for changing the boundary conditions to clamped on all four edges; and (2) identify the minimum acceptable factor of safety and the technical basis for its selection.

In a letter dated April 19, 2009, the applicant submitted its response to RAI-SRP9.1.2-SEB1-07, in which it provided its technical basis for the revised calculation of buckling for the cell wall. The staff reviewed the response and determined that the information provided was insufficient, and that a significantly expanded technical basis would be needed before the staff could accept the cell wall buckling calculation. At the August 2009 audit, HOLTEC informed the staff that it

was conducting a detailed nonlinear analysis of the bottom of the rack for vertical compressive load, and presented the ANSYS computer model and preliminary results. The staff found this to be a considerable analytical improvement.

In a letter dated November 11, 2009, the applicant submitted a revised RAI response, indicating it had re-evaluated the buckling capacity of the spent fuel storage rack cells at the base of the rack using an ANSYS finite element analysis. The results show that the spent fuel rack cells remain in a stable configuration when subjected to 1.5 times the maximum seismic load without any gross yielding of the storage cell; therefore, the ASME Code requirements for Level D conditions in this area are satisfied. The ANSYS analysis and results were included in TR-54, Revision 3 (November, 2009).

During the June 2010 audit, the staff reviewed HOLTEC's final results of the ANSYS buckling evaluation of the cell walls, at the base of the spent fuel rack. The calculation shows that a 1.5 factor of safety, in accordance with the acceptance criterion in ASME Code Section III, Subsection NF, has been achieved. The staff found the analytical method used and the results obtained to be acceptable, based on its detailed review of HOLTEC's calculation. Therefore, RAI-SRP9.1.2-SEB1-07 is resolved.

In addition to the SFP rack and pool liner, the staff also performed a review of the integrity of the spent fuel assemblies during a design basis seismic event. NUREG-0800 Section 3.8.4, Appendix D, provides guidance for evaluating the consequences of seismic loads on the fuel assemblies. The applicant described its evaluation of the spent fuel rack design (including fuel assemblies) in TR-54, "Spent Fuel Storage Racks Structural/Seismic Analysis," Revision 4. The applicant calculated the maximum deceleration of a fuel assembly due to lateral SFP seismic demands as 3.38 g, assuming that all assemblies move in phase. The worst-case maximum deceleration of a fuel assembly due to out-of-phase motion, fuel-to-fuel impacts, or fuel-to-wall impacts was reported as 5.69 g. This seismic demand was less than the reported Westinghouse minimum allowable grid impact capacity of 8.92 g and results in a factor of safety of approximately 1.57.

The staff's review of TR-54, Revision 4, found that the technical basis for the minimum allowable grid impact capacity (8.92 g) was not sufficiently described in the report. To address this, the staff performed an audit on July 14, 2011, to review TR-54 supporting documentation. The staff reviewed the applicant's grid crush strength test results summarized in report NFRE-07-08, "AP1000 RFA-2 Mid Grid Dynamic Crush Strength Preliminary Verification Report," which was used to develop the AP1000 minimum allowable grid impact force. The staff confirmed that the tests were performed in accordance with previously accepted practices and that the resultant grid crush data supported the stated minimum allowable grid impact force presented in TR-54.

The staff ensured that the applicant identified the limiting failure mechanism by reviewing two research reports pertaining to spent fuel cladding integrity developed by Sandia National Laboratory (SNL) and Lawrence Livermore National Laboratory (LLNL). The staff's review of these reports (ML042710347 and ML061990439, respectively) found that spent fuel cladding can withstand forces (accelerations) of up to 10 times the grid loading limits proposed by the applicant. The staff could not, however, make conclusions with respect to grid integrity from the data within these reports, as both reports were focused on cladding ruptures and did not investigate buckling of the grids. However, data presented in the reports related to cladding were sufficient when combined with the information provided in TR-54 for the staff to conclude that grid buckling was the limiting failure mechanism for the AP1000 fuel design.

Based on the following review findings: (a) the applicant's fuel grid testing was performed in accordance with NUREG-0800 Section 4.2; (b) significant margin in the fuel assembly grid (factor of safety of 1.57); and (c) indication that grid failure is the limiting fuel assembly failure mode, the staff concludes that there is reasonable assurance that the AP1000 spent fuel assemblies, when stored in the spent fuel racks in the SFP, will retain their structural integrity when subjected to SSE level demands.

9.1.2.2.1.1.3 Conclusion

The staff has conducted a detailed review of TR-54, which addresses DCD Revision 15 COL Information Item 9.1-3: "Perform a confirmatory structural dynamic and stress analysis for the spent fuel rack, as described in Subsection 9.1.2.2.1. This includes reconciliation of loads imposed by the spent fuel rack on the SFP structure described in Subsection 3.8.4." The staff finds the spent fuel rack design, as described in TR-54, Revision 4, to be acceptable. On the basis of its review, the staff concludes that the substance of the COL information item is completely addressed by TR-54, and that completion of this COL information item is no longer needed.

In its previous evaluation of AP1000 DCD, Section 9.1.2, the staff identified that the spent fuel rack design must meet the relevant requirements in GDC 2 and GDC 4. Based on its review, the staff has concluded that the spent fuel rack design meets these 10 CFR Part 50 requirements.

9.1.2.2.1.2 Spent Fuel Rack Density

In the AP1000 DCD, Revision 17, the applicant increased the SFP storage rack density from high density to higher density racks. GDC 61 requires in part that fuel storage systems be designed with a residual heat removal capability having reliability that reflects the importance to safety of decay heat removal. As indicated in NUREG-0800 Section 9.1.2, III.1, the staff considers the design of high-density fuel storage systems to be acceptable in this regard if (among other things) low-density storage is used, at a minimum, for the most recently discharged fuel to enhance the capability to cool it. The applicant's fuel storage system design does not use low-density racks as specified by NUREG-0800 Section 9.1.2.III.1, and this difference between the proposed design and the staff's acceptance criteria has not been explained and justified. In RAI-SRP9.1.2-SBPA-14, the staff requested that the applicant address this difference and explain how the proposed fuel storage system design is adequate for satisfying GDC 61 requirements commensurate with the staff's review criteria. The staff requested that the AP1000 DCD be revised to include this information. The staff identified this as Open Item OI-SRP9.1.2-SBPA-14.

In a letter dated September 22, 2009, the applicant stated that in the previously approved DCD revision, the SFP uses only high density racks. In Revision 17 of the DCD, the applicant further increased the SFP density in order to increase storage capacity. The applicant clarified that the SFP cooling system is designed to remove the decay heat produced by the spent fuel assemblies during all modes of plant operation, regardless of the spent fuel assemblies' storage locations in the pool.

The staff evaluated the applicant's response and determined that using a SFP cooling system design with sufficient cooing capability to maintain the stored fuel cooled is an acceptable method of meeting the requirements of GDC 61. Therefore, the staff finds that the applicant's

response is acceptable and the staff's concerns discussed in Open Item OI-SRP9.1.2-SBPA-14 are resolved.

The staff's detailed evaluation of the SFP cooling system capacity is documented in Section 9.1.3 of this report

9.1.2.2.2 Spent Fuel Pool Water Level Increase

The applicant proposed a series of changes related to an increase in normal SFP water level in DCD Section 9.1.2.2. The bases for these changes are addressed in TR-121. These changes include:

1. Increase the normal water volume of the pool from 685,159 to 721,121 liters (181,000 to 190,500 gallons).

2. Raise the water level from 76 cm (30 in) below the operating deck to 38 cm (15 in) below the operating deck.

3. Delete reference to "a minimum of 10 feet of shielding water above the spent fuel assemblies" and replace with "a minimum of 8.75 feet of shielding water above the active fuel height of spent fuel assemblies."

In describing spent fuel transfer operation, AP1000 DCD, Revision 15, stated that waterways are of sufficient depth to maintain "a minimum of 10 feet of shielding water above the active fuel height." AP1000 DCD, Revision 17, proposes to change Section 9.1.2.2 to state that waterways are of sufficient depth to maintain "a minimum of 8.75 feet of shielding water above the active fuel height." This corresponds to a decrease in minimum shielding of 38.10 cm (15 in) in minimum shielding from DCD Revision 15.

Also, in the AP1000 DCD, Revision 17, Tier 1, Table 2.1.1-1, "Inspections, Tests, Analysis and Acceptance Criteria," line 5, the applicant proposes to change the maximum elevation to which the bottom of a fuel bundle can be lifted from 7.70 meters (m) (25 ft, 3 in) below the operating deck to 7.47 m (24 ft, 6 in) below the operating deck. This corresponds to an additional lift of 22.86 cm (9 in).

In AP1000 DCD, Revision 17, Table 9.1-2, "Spent Fuel Pool Cooling and Purification System Design Parameters," the applicant proposes to change the SFP normal water level from 30.48 cm (12 in) below the operating deck to 38.10 cm (15 in) below the operating deck. This corresponds to a decrease in normal water level of 7.62 cm (3 in).

With the increased fuel bundle lift of 22.86 cm (9 in) and a decreased normal SFP water level of 7.62 cm (3 in), the staff believes that the change in minimum water shielding is be a decrease of 30.48 cm (12 in), not a decrease of 22.86 cm (9 in).

In RAI-SRP9.1.2-SBPA-09, the staff requested the applicant to clarify the proposed changes described above in the DCD so that the decrease in minimum shielding can be accurately determined.

On June 25, 2009, the staff conducted a regulatory audit of the DCD Revision 17 documentation and met with the applicant's personnel to identify the specific information required in order to resolve this RAI and other Chapter 9 RAIs.

During the June 25, 2009 audit, the applicant stated that the change in minimum shielding from 2.89 m (9.5 ft) to 2.67 m (8.75 ft) does not exceed the limit of 2.5 millirem to the bridge operator. The justification for this change is documented in calculation APP-GW-N2C and is discussed in the applicant's response to RAI-SRP12.3-CHPB-02.

With respect to the maximum elevation to which the bottom of the fuel bundle can be lifted, the applicant identified an ITAAC limit for mechanical hard stops. The mechanical hard stops provide 2.59 m (102 in) of water shielding.

The staff's evaluation of the justification for the minimum shielding change, as discussed in the applicant response to RAI-SRP12.3-CHPB-02, was reviewed and documented in Section 12.3 of this report. The staff identified this as Open Item OI-SRP9.1.2-SBPA-09.

In a response dated September 18, 2009, the applicant stated that DCD Revision 17, Tier 1, Table 2.1.1-1, Paragraph 5, limits the fuel assembly raise height to 7.4 m (24 ft, 6 in) between the bottom nozzle and the operating floor; elevation 41.15 m (135 ft, 3 in). This corresponds to 2.6 m (8.5 ft) of water shielding above the active portion of the fuel when the refueling cavity/SFP water level is at the 40.84 m (134 ft) elevation (minimum water elevation for fuel movement). This limit is established as a mechanical hard stop limit for the refueling machine (RM) and the FHM.

DCD Revision 17, Tier 2, throughout Section 9.1 and Section 12 establishes the minimum water coverage as 2.7 m (8.75 ft), 32 m (105 ft) elevation, above the active portion of the fuel when the refueling cavity/SFP water level is at the 40.84 m (134 ft) elevation. This is a 0.23 m (9 in) reduction in shielding. The RM and FHM would have controls to limit the hoist up travel to satisfy this requirement. Section 12.3 of this report evaluates the impact of this change on radiation protection considerations.

The staff finds that the RAI response discussed above clearly shows how the minimum shielding has been impacted by the changes in the RM limit set points. The staff determined that, based on the above discussion, Open Item OI-SRP9.1.2-SBPA-09 is resolved. The remaining changes are reviewed in Section 9.1.3 and 12.3 of this report. The staff determined that these changes made to DCD Section 9.1.2 are conforming changes and do not impact the staff's safety evaluation of DCD Section 9.1.2. Therefore, the staff finds the proposed changes acceptable.

9.1.2.2.3 Fuel Handling Crane Change

The applicant proposed to delete references to the fuel handling jib crane and replace them with references to the new-fuel handling crane in DCD Section 9.1.2.2.1. However, in response to RAI-SRP9.1.4-SBPB-01, the applicant stated in a letter dated June 26, 2008, that the function of moving new fuel will be transferred to the FHM and that the new-fuel handling crane will be eliminated. The evaluation of this change is reviewed in Section 9.1.4 of this report. The staff determined that this change made to DCD Section 9.1.2 is a conforming change that does not impact the staff's safety evaluation of DCD Section 9.1.2. Therefore, the staff finds the proposed change acceptable.

9.1.2.2.4 Spent Fuel Criticality Analysis

9.1.2.2.4.1 Summary of Technical information

In the certified DCD Revision 15, Section 9.1.2, "Fuel Storage and Handling," it is stated in Section 9.1.6 that the COL applicant is responsible for a confirmatory criticality analysis for the spent fuel rack, as described in Section 9.1.2.3. In DCD Revision 17, the applicant proposed to change this COL action by performing the confirmatory criticality analysis so that the COL action item is no longer necessary. DCD Section 9.1.2.3 is revised to reflect that the criticality analysis is now complete, and Section 9.1.6 is revised to state that the COL information requested in this section has been completely addressed in TR-65, Revision 2, and the applicable changes are incorporated into the DCD. The applicant stated that no additional work is required by the COL applicant. The technical details of the criticality analysis for the AP1000 spent fuel storage design is presented in TR-65, Revision 2. This report provides the technical support for the changes found in Section 9.1.2 of DCD Revision 17. The staff's review of the criticality analysis of AP1000 spent fuel storage includes DCD Revision 17, Section 9.1.2 and the supporting TR-65, Revision 2.

In a letter dated September 16, 2009, the applicant stated that it will be submitting an alternate loading pattern with a restriction, which will preclude the need for using burnup credit in the Region 2 rack criticality analysis. This will result in a change to the technical specifications (TS). Open Item OI-SRP9.1.1-SRSB-08 was created to track all changes related to this restricted loading pattern and the corresponding analysis.

Subsequently, in a letter dated July 28, 2010, the applicant retracted the September 16, 2009 proposal that suggested a restricted loading pattern and clarified the applicant's intent to rely on the analysis presented in TR-65, Revision 2, with full loading as the basis for the AP1000 Revision 17 SFP criticality analysis. Furthermore, the applicant provided a response to Open Item OI-SRP9.1.1-SRSB-08 in the July 28, 2010, letter that demonstrates consistency with the burnup credit methodology used and approved in current reactors. The staff has reviewed TR-65, Revision 2, and the RAI response from the July 28, 2010, letter against NUREG-0800 Section 9.1.1, the guidance in the August 19, 1998, NRC memorandum authored by Larry Kopp "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (the Kopp guidance) and past precedence examples. The staff concludes that the applicant follows the current available guidance and therefore the response and burnup credit analysis are acceptable. Therefore, Open Item OI-SRP9.1.1-SRSB-08 is satisfied and closed.

The AP1000 design includes facilities for the onsite storage of irradiated spent fuel. The spent fuel storage facility is located within the seismic Category I auxiliary building fuel handling area.

Irradiated fuel is stored in a stainless steel-lined concrete SFP containing 8 racks capable of holding a total of 889 assemblies. The spent fuel racks are divided into two regions within the common pool. The Region 1 rack consists of stainless steel cells with neutron absorbing material (Metamic®) attached to all sides facing other storage locations. Cells are separated by a water gap. There is no absorber on the cell faces adjacent to the pool wall or when SFP geometry does not require neutron absorber panels to remain subcritical. Metamic® panels are located on Region 1 edges in locations that could physically hold an assembly between the Region 1 and 2 racks or between the Region 1 rack and pool wall. Region 2 racks consist of the same stainless steel box structure as Region 1 with fixed neutron absorber attached to the

outside of the walls, but there is no intervening water gap in this case. In addition, there are five damaged fuel locations, which are of the same design as the Region 1 storage racks.

The spent fuel storage facilities are designed to meet seismic Category I requirements and maintain a subcritical storage configuration of fuel during normal storage and accident conditions.

The applicant has provided a system description in Section 9.1.2 of the DCD. In addition, TR-65, Revision 2, is reviewed as part of the AP1000 DC application. The criticality analysis is summarized here, in part, as follows:

Criticality analyses are performed for the AP1000 spent fuel storage racks to demonstrate K_{eff} ≤.95 during normal conditions, assuming a maximum nominal initial enrichment of 4.95 w% U-235 and taking into consideration uncertainties due to fuel and rack manufacturing tolerances. In addition, the spent fuel storage racks will remain subcritical under optimum moderation conditions. TR-65, Revision 2, provides the criticality analyses, including a description of the analytical methods used in the criticality analyses, as well as a description of the analytical uncertainties, equipment manufacturing tolerances, and other analysis assumptions.

In DCD Tier 1 Table 2.1.1-1, ITAAC Item 7 addresses criticality control during normal operation, design basis seismic events, and design basis dropped fuel assembly accidents.

COL Information Item 9.1-4

In Revision 17 of the AP1000 DCD, the applicant proposed to resolve COL Information Item 9.1-4 by performing a confirmatory criticality analysis for the spent fuel racks. The applicant submitted TR-65, Revision 0, for the staff's review to demonstrate that it had met the requirements of COL Information Item 9.1-4. COL Information Item 9.1-4 in the DCD is also discussed in NUREG-1793 as COL Action Item 9.1.6-4. This evaluation is a secondary review for COL Information Item 9.1-4 with respect to the compatibility of the neutron absorbing materials used in the spent fuel racks with the spent fuel environment.

In Revision 15, Section 9.1.6 to the AP1000 DCD, COL Information Item 9.1-4 states:

> The Combined License applicant is responsible for a confirmatory criticality analysis for the spent fuel racks, as described in Subsection 9.1.2.3. This analysis should address the degradation of integral neutron absorbing material in the spent fuel pool storage racks as identified in GL-96-04, and assess the integral neutron absorbing material capability to maintain a 5 percent subcriticality margin.

In Revision 17 of the AP1000 DCD, the applicant proposed to resolve the COL information item with the following:

> The Combined License information requested in this subsection has been completely addressed in TR-65, and the applicable changes are incorporated into the DCD. No additional work is required by the Combined License applicant.

9.1.2.2.4.2 Evaluation

The staff's review of AP1000 DCD Section 9.1.2 follows the procedures outlined in NUREG-0800 Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3. Compliance with regulatory requirements was verified based on the criteria defined by GDC 62 and 10 CFR 50.68, "Criticality accident requirements."

The TS identified in NUREG-0800 Section 9.1.1, Section III, are reviewed in the applicable sections of this report.

Design Bases

DCD Tier 2, Section 4.3.2.6.1 is consistent with TR-65, Revision 2, in stating that the K_{eff} of fully loaded spent fuel storage racks will not exceed 0.95, assuming that the racks are flooded with potential moderator. DCD Tier 2, Section 4.3.2.6.1 also states that the maximum new fuel enrichment level must be less than or equal to 5.0 w% U-235. This is not consistent with the maximum nominal initial enrichment of 4.95 w% U-235 stated in TR-65, Revision 2.

Revision 17 of the DCD states that criticality analyses will demonstrate that the fuel storage rack geometry in combination with the integral neutron absorber material is sufficient to maintain the fuel in a subcritical condition as given above. TR-65 states that the applicant will follow the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.17, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors," with regard to criticality safety.

By following the guidelines in NUREG-0800 Section 9.1.1, the staff finds that the design bases described above for the fuel storage and handling systems meet the requirements of GDC 62 and 10 CFR 50.68(b).

Criticality Analysis Methodology

The primary method for determining the multiplication factor for the various configurations being considered in this analysis is the Monte Carlo Code MCNP-4a, with the attached nuclear data libraries ENDF/B-V and ENDF/B-VI. The applicant validated this combination against relevant critical experiments as defined in the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) International Handbook of Evaluated Criticality Safety Benchmark Experiments. This validation is used to determine the inherent bias implicit in this approach, which is added to ensure that the multiplication factor is below the acceptable limit. The MCNP calculations used an appropriate number of cycles, histories per cycle, and skipped cycles to ensure that the MCNP calculation was converged.

The design criteria are consistent with the requirements outlined in 10 CFR 50.68(b)(4) for spent fuel racks. The requirement states that: for fully loaded spent fuel racks the multiplication factor must be below 0.95 including all uncertainties and biases, and taking credit for soluble boron in the cooling water. In addition, the multiplication factor must be less than 1.0 when the fully loaded rack is flooded with fresh water. In all cases rack cells are assumed to be loaded with fuel of maximum reactivity. These criteria are used in conjunction with the latest evaluation techniques described in ANSI/ANS-8.17-2004, which can be summarized in the following relationship:

$$k_{eff} = k_c + \Delta k_p + \Delta k_c$$

Where

k_{eff} = maximum multiplication factor.
k_c = calculated multiplication factor determined by MCNP.
Δk_p = allowance for convergence, modeling, and manufacturing limitations.
Δk_c = bias uncertainty associated with code validation.

The allowance for convergence and manufacturing tolerances included the statistical accuracy of the MCNP calculations, rack tolerances, fuel tolerances, and depletion uncertainties. The code bias uncertainties are based on the comparisons with critical experiments.

The MCNP calculations are generally carried out on infinite arrays of fuel cells. This is achieved by assuming a single unit cell with reflecting or periodic boundary conditions.

The methodology presented in TR-65, Revision 2 is consistent with standard industry practice and with past NRC approved techniques. The staff, therefore, approves of the criticality analysis methodology as described in TR-65, Revision 2.

Assumptions

In Section 4 of TR-65, the applicant listed the modeling assumptions used in the analysis to ensure a conservative approach. The staff reviewed these assumptions to ensure that they maximized the k_{eff} calculations and would therefore be conservative. The staff concluded that all but one of the assumptions was conservative. The remaining assumption involved ignoring certain minor components, which the applicant claimed would in effect remove the neutron absorption by these components that would normally occur. While the staff agrees with this statement, the staff also notes that by ignoring the components in the modeling, the applicant is in effect replacing them with borated water which would also act as an absorber. The net effect is not specified; however, based on previous experience the staff feels that the change to k_{eff} based on this assumption would be insignificant and more than bounded by the other assumptions. Therefore, the staff approves of the approximations used in this calculation based on the preceeding discussion.

Input Data

The applicant used the basic input data required for the calculations as described in TR-65, Revision 2. This data covers geometric input for the racks and fuel assemblies, core operating history, burnable poison treatment and axial burnup distribution.

The staff found that borosilicate glass burnable absorber rods were not considered in the analysis. In RAI-SRP9.1.1-SRSB-07, the staff requested that the applicant describe how the use of borosilicate glass burnable absorbers affects the analysis. In its response dated September 29, 2009, the applicant stated that TR-65 was revised to include borosilicate glass burnable absorbers. The revised TR shows that they are bounded by other combinations of inserts and, therefore, have no impact on the results. The staff finds this response acceptable.

Computer Codes

The two computer codes used in this analysis are MCNP-4a and CASMO-4. They are used to determine the multiplication factors, and the core depletion behavior and sensitivity to manufacturing tolerances, respectively.

MCNP-4a

1) Appendix A of TR-65 presents a series of critical experiments that were analyzed to determine the bias and uncertainty associated with the use of MCNP-4a and its attached nuclear data library. In addition, the KENO5a Monte Carlo code and its nuclear data library are used to analyze the same experiments as a cross check of the calculated results

The staff reviewed the experiments chosen for this comparison and concludes that they include the magnitudes of the most important parameters of the fuel assemblies to be stored in the racks. The resulting values of the multiplication factor as a function of energy of the average lethargy causing fission (EALF) was analyzed using linear regression analysis. The result showed no strong trends, since the correlation coefficients are low. These calculated values for bias and uncertainties are used in calculating the maximum multiplication factor for the storage racks.

The applicant investigated the possibility of the multiplication factor having a systematic trend. The staff reviewed this investigation and agrees with the applicant's conclusion that the multiplication factor showed no discernable trend with the parameters investigated. In addition to the investigation of potential trends in the multiplication factor, the applicant investigated the sensitivity of several parameters on the multiplication factor. The staff concludes that the use of these systematic trend and parametric study investigations follows the guidance to account for all biases and uncertainties provided in the NRC memo authored by Larry Kopp, which is currently used by the industry and previously accepted by the agency.

The applicant analyzed a small number of critical experiments that included Mixed Oxide (MOX) fuel. In this manner the ability of the combination of MCNP-4a and its attached nuclear data library to handle MOX fuel could be estimated. The staff reviewed these comparisons of critical experiments to the results of the applicant's analysis. It is noted that there is a discrepancy introduced by ^{241}Pu decay and the implied growth of ^{241}Am; however, the staff concludes that this discrepancy does not appreciably affect the results. The staff concludes that the applicant's methodology provides overall conservative results and is therefore acceptable.

Based on the staff's review of the benchmark calculations presented in Appendix A of TR-65, Revision 2 as detailed in the preceding discussion, the staff finds the use of MCNP-4a and the calculated MCNP-4a bias and bias uncertainty values to be acceptable.

CASMO-4

The applicant uses CASMO-4 for the following two purposes in this analysis:

1) Depletion and decay calculations to determine the isotopic content of spent fuel, and
2) Determinations of changes in the multiplication factor introduced by perturbations in the storage rack.

CASMO-4 is an industry standard transport theory depletion code originally developed for core analyses. It has been used for SFP depletion calculations previously and approved by NRC staff. In Appendix B of TR-65, Revision 2, the applicant provides benchmarking results for CASMO-4 including reactor critical comparisons, cross section library comparisons, and code to code comparisons.

The applicant compared the k-inf values calculated with CASMO-4 to experimental reactor critical data. This database covered a wide range of design conditions to ensure the calculated uncertainties would be applicable to the AP1000 design.

The applicant investigated the effects of using different available cross-section data libraries. The results did not indicate a significant difference between the investigated libraries.

The applicant provided a code to code reactivity difference comparison with MCNP-4a as an additional check on top of the critical experiments comparisons. These comparisons showed good agreement and did not result in differences that required additional scrutiny.

The staff reviewed the methodology of the depletion code benchmarking against the Kopp memo as well as other recently approved license amendment request applications. The staff reached the following conclusions regarding the depletion code benchmarking:

(1) The applicant compared the calculated results to a variety of relevant critical experiments. While it is recognized that more recent relevant data have been made available, the experiments used by the applicant provide reasonable assurance that the relevant biases can be determined. In RAI-SRP9.1.1-SRSB-06, the staff requested additional information regarding the effects of performing the analyses at maximum water density (4 °C) and the lack of Tungsten gray rod data in the benchmark tests. In response, the applicant explained that the extrapolation from the minimum temperature used in the analysis to 4 °C results in a negligible effect on the uncertainty calculations. The staff determined that while this effect should normally be quantified, the additional analytical margin included on top of the calculated bias uncertainty is sufficient to bound the small effect that would result from slightly more dense water. The applicant addressed the question regarding lack of Tungsten gray rod data by stating that the effect of tungsten inserts would not appreciably affect the benchmark studies which already include various other absorber inserts. The staff agrees that this observation is most likely correct, but notes that there currently is no database to support this conclusion as this is a new material. The staff approves the use of the applicant's methodology based on the wide range of other parameters investigated, but recognizes that small future changes could occur based on the collection of new data.

(2) Based on the comparison of the cross-section libraries, the staff finds that the library selection between current industry standard libraries has little effect on the reactivity calculation. Therefore the staff finds that the library used by the applicant is acceptable.

(3) The staff confirmed that applicable permutations were investigated to determine the appropriate depletion uncertainties.

(4) The applicant included an additional analytical margin on top of the calculated bias uncertainty.

Therefore, the staff approves the use of CASMO-4 for the depletion calculations necessary for the AP1000 SFP criticality analysis.

Criticality Analysis

In this section the review of the criticality calculations for the Region 1 and Region 2 racks from TR-65, Revision 2 is presented. In addition, the review of the consequences due to possible abnormal and accident scenarios is discussed.

The Region 1 and Region 2 racks are represented by detailed 2-D MCNP models, with reflecting boundary conditions on the surfaces separating one cell from the other. In this manner, an infinite array of fuel cells is represented in the calculation. Additionally, MCNP models that explicitly involve more storage cells are used to analyze the abnormal or accident conditions.

The 3-D MCNP calculations have the same detail in the X-Y plane as the 2-D models, and are extended axially in the Z-direction. The model includes a 30 cm axial water reflector, which does not include any boron, even for those cases that include boron in the water in the storage cells.

The CASMO calculations are 2-D, and thus the regions above and below the fuel are not represented. CASMO is used to determine the perturbations in the multiplication factor due to manufacturing tolerances. These perturbations are presented as adjustment factors to the multiplication factor determined for a nominal fuel loading case.

Region 1 Storage Racks

Region 1 storage racks are qualified to store fresh fuel with an enrichment of up to 4.95 w% U-235. There are 243 Region 1 storage locations in the storage pool. The geometric representation is as described previously in this report.

As part of the Region 1 criticality analysis, the applicant analyzed various abnormal conditions in addition to the standard loading. These included eccentric fuel assembly positioning, uncertainties due to manufacturing tolerances, water temperature/density, and accident conditions.

The staff reviewed the criticality analyses for Region 1 as presented against the guidance in NUREG-0800 Section 9.1.1. As part of a regulatory audit held May 6-7, 2009, the staff inspected the computer runs used by the applicant to ensure that the methodology was correctly followed. Based on the review of the methodology and its application, the staff finds the Region 1 racks to be acceptable for use as presented in TR-65.

Region 2 Storage Racks

The applicant performed the Region 2 SFP criticality analysis in a manner similar to the Region 1 analysis, except that the analysis included the use of burnup credit in order to meet the requirements of 10 CFR 50.68. The applicant followed the guidelines provided by the Kopp memo as well as the methodology of recently approved license amendment requests in calculating the burnup credit.

In RAI-SRP9.1.1-SRSB-08, NRC staff questioned the applicant's burnup credit assumption that a 5 percent reactivity uncertainty penalty included the effects of missing nuclide data on the computational biases and uncertainties. In response to this RAI, the applicant's September 16, 2009, letter described a loading pattern restriction on the Region 2 racks and its plan to submit a simplified analysis that does not require burnup credit. This plan will not require any changes to the physical rack design as presented in TR-65. Subsequently, after SFP license amendment requests that included similar burnup credit methods were accepted by the agency, the applicant submitted a letter dated July 28, 2010, requesting that the agency return to reviewing the full-capacity SFP criticality design in TR-65 Revision 2.

The staff compared the applicant's burnup credit methodology to the guidance provided in the Kopp memo as well as to SERs for recently approved SFP license amendment requests. The staff notes that the burnup credit guidance does not explicitly state how an applicant should handle biases and bias uncertainties and whether or not the 5 percent reactivity uncertainty covers them when considering potential lack of data regarding specific isotopes.

As a result of reviewing the recent SERs, the staff determined that the submittals had been based on approaches to the implementation of burnup credit similar to that used by the applicant in TR-65, Revision 2. The staff determined that past precedent supports the applicant's position that the use of the 5 percent reactivity uncertainty penalty has been approved previously to cover the depletion bias uncertainty. The staff considers Open Item OI-SRP9.1.1-SRSB-08 closed.

During the regulatory audit held May 6-7, 2009, the staff reviewed the depletion and criticality calculations used by the applicant in the AP1000 Region 2 SFP criticality analysis. The staff determined that the applicant used the codes previously approved in this section and correctly applied the applicable code biases and bias uncertainties while calculating the values for k_{eff}. Based on the staff's review of the methodology and analysis, as detailed in this section, along with the precedent set by the recent approval of applications using burnup credit, the staff concludes that the applicant's analysis of the Region 2 SFP criticality demonstrates compliance with the requirements of 10 CFR 50.68 by following the Kopp memo.

Revised Rack Dimensions

DCD Revision 18 introduced an increased thickness of the Region 1, Region 2 and damaged fuel storage rack cell walls, for the purposes of increased structural stability. DCD Revision 19 includes additional changes related to this change in thickness. In a letter dated March 3, 2011, the applicant submitted TR-65 Revision 3. Revised criticality analyses based on the latest spent fuel pool rack design are located in Supplement 1 of TR-65, Revision 3.

Following the same methodology as found in TR-65 Revision 2, the new analyses provide new k_{eff} calculations based on the burnup loading curve found in TS 3.7.12 (Figure 3.7.12-1) of the DCD. These new calculations demonstrate that the k_{eff} values are lower for all of the analyzed fuel as compared with the previous analysis (TR-65, Revision 2), except for the fuel with 2.5 wt% and 4.0 wt% enrichment. The calculations of k_{eff} for fuel with 2.5 wt% and 4.0 wt% enrichment demonstrate an increase, but this increase is within the administrative margin maintained by the applicant in the previous analysis (TR-65, Revision 2). These increased k_{eff} values do not violate the limits specified in 10 CFR 50.68.

The staff reviewed the calculations included in Supplement 1 of TR-65, Revision 3 and confirmed that the same methodology presented in Revision 2 was used including the

determination and use of biases and uncertainties. The revised k_{eff} values do not violate the regulatory limits. Therefore, staff finds the AP1000 spent fuel pool criticality calculations based on TR-65, Revision 3 acceptable.

Restrictions and Limitations

The AP1000 SFP design as presented in TR-65, Revision 2 is approved for use with the following limitation:

> Limitation #1: Applicability
>
> The AP1000 SFP is approved for storing the fuel types presented in (or bounded by) TR-65, Revision 2. Any fuel not bounded by those used in TR-65, Revision 2 (higher enrichment, different burnable absorbers designs than analyzed, etc.) will require further analysis.

COL Information Item 9.1-4 Evaluation

To assure compliance with GDC 4, NUREG-0800 Section 9.1.2 III.2.G states the reviewer should verify that "the materials wetted in the SFP, (e.g., spent fuel racks, fixed neutron poison, and the SFP liner) and, if applicable, the new fuel vault are chemically compatible and stable. The review also verifies whether there are potential mechanisms to alter the dispersion of any strong fixed neutron absorbers. The secondary reviewer provides input for this review." NUREG-0800 Section 9.1.2 I.11.B further states that "the reactivity of fuel in the SFP is controlled by plates or inserts attached to spent fuel racks containing neutron poison dispersed in a matrix. In some environments, the matrix may degrade and release the neutron poison, resulting in some reduction of neutron absorbing properties of the panels. The licensee should have a program for monitoring the effectiveness of the neutron poison present in the neutron absorbing panels."

The staff has reviewed the information included in TR-65, which identifies the neutron absorber material in the spent fuel storage racks as Metamic®, a metal matrix composite material consisting of a Type 6061 aluminum (Al 6061) alloy matrix reinforced with boron carbide (B_4C). TR-65, Section 2.4.8 describes testing to qualify the Metamic® material for spent fuel rack service, including short and long-term elevated temperature tests, accelerated corrosion and radiation tests, mechanical properties and neutron transmission testing. The staff has previously issued an SER approving a topical report supporting the use of the Metamic® material in spent fuel racks in an operating plant. The operating plant subsequently submitted a license amendment request to use the Metamic® material in the SFP, which was approved via an SER issued by the NRC staff. The staff noted that the same generic vendor report supporting the application to use Metamic® in the operating plant is referenced in TR-65. The SER for the license amendment at the operating plant placed conditions on the use of the Metamic® material: specifically, implementation of a coupon sampling program to ensure performance consistent with the laboratory qualification testing.

The Metamic® absorber material is relied upon in the TR-65 criticality analysis to maintain the required 5 percent subcriticality margin. While TR-65, Section 2.4.8 indicates no significant loss of neutron absorbing capacity is expected for the Metamic® material based on the testing conducted, the Metamic® material is a new material with very little operating experience in the SFP environment. Spent fuel racks with Metamic® have been installed at the operating plant but the time in service for these racks as of March 2008 has been only a few months, and no

coupons have been withdrawn or tested. TR-65 and the DCD include no mention of the coupon surveillance program implemented by the operating plant, nor do they recommend a similar program for the AP1000 plants.

Although the data from the operating plant surveillance program could be used to confirm the laboratory test results and could be extrapolated to the Metamic® in the AP1000, a relatively small amount of data from the operating plant will be available when construction begins for the first AP1000 plants. Further, the service conditions for the Metamic® material in the operating plant may not be identical to the expected service conditions for the Metamic® material in the AP1000 design. Additionally, some qualification tests such as the radiation testing only encompassed a 40 year rather than a 60 year life. Therefore, the staff considered that a coupon sampling plan similar to that implemented in the operating plant should be implemented for the AP1000 plants. Therefore, in RAI-SRP9.1.2-CIB-01, the staff requested that the applicant provide the following information, and include the information in the next revision to the AP1000 DCD:

1) A description of the neutron absorbing material to be used in the spent fuel storage racks. The description should include the material type, chemical composition, and mechanical properties, and a discussion of the suitability of the absorber material for long-term use in the SFP environment. Include a description of any testing performed to qualify the material for 60 years service in the SFP environment, specifically with respect to corrosion and radiation degradation. The description should also address whether the absorber material has an anodized finish, the anodizing process used, and the cleaning process to ensure removal of surface contaminants prior to installation.

2) A description of the recommended program to be implemented by the licensee to confirm that the behavior of the neutron absorbing material is consistent with the behavior of the material in the qualification tests. For example, the DCD may need to identify a COL item requiring the COL applicant to include a description in the COL application of the coupon sampling or monitoring program for the licensee to implement when the plant is placed into commercial operation.

The applicant responded by letter dated April 18, 2008. With regard to question #1, the applicant stated that the material that will be used in the AP1000 fuel storage racks is Metamic, a metal matrix composite material consisting of a Type 6061 aluminum alloy matrix reinforced with boron carbide (B_4C) as described in TR-65. The Metamic® will be in the form of sheets having a nominal thickness of 0.269 cm (0.106 in) and a minimum ^{10}B areal density of 0.0304 gm/cm^2 (minimum 30.5 wt percent B_4C). The panels are not intended to be anodized, but will be cleaned via glass bead blasting and washing with demineralized water to ensure removal of surface contamination prior to installation. The applicant also included the density, yield and ultimate strength, and elongation of the material in its description of the material.

In the April 18, 2008 letter, the applicant also described the testing performed to qualify the Metamic® material for 60 years service in the SFP environment. The applicant referenced proprietary testing by a vendor (HOLTEC) as documented in HOLTEC Report HI-2043215, "Source Book for Metamic® Performance Assessment," Revision 2, HOLTEC International, dated September 2006 (not publicly available) (hereafter referred to as the "HOLTEC report") as its basis for qualification of the material:

- Elevated temperature testing of 31 w% B_4C Metamic® at 398.9 °C (750 °F) in air for nearly a year. There was no reduction in thickness, change in weight, reduction in ^{10}B

content or change in density. The applicant stated that the results of this test demonstrate that exposure to a temperature of 48.9 °C (120 °F) in the SFP will not detrimentally affect the condition of the Metamic® panels.

- Accelerated corrosion testing at 93.30 °C (200 °F) for 90 days. No corrosion was observed and no significant change of ^{10}B areal density. The applicant stated that while these tests were carried out at a temperature only 26.7 °C (80 °F) higher than the typical upper bound, this is sufficient to yield results representative of longer periods. The applicant referenced Department of Energy (DOE) fundamentals Handbook DOE-HDBK-1015/1-93 Module 2 – Corrosion, which states "A temperature rise in the range of -6.7 °C to 10 °C (20 °F to 50 °F) doubles the corrosion rate until the formation of the protective oxide film is complete." The applicant also stated that the aluminum oxide layer that forms on the Metamic® is largely inert and that once the protective oxide film forms the corrosion rate becomes approximately zero. The applicant also referenced the DOE handbook with regard to the effect of pH on corrosion rate. The handbook indicates essentially zero corrosion rate at pH 5.5 and a corrosion rate of nearly zero in a pH range of 4-8. The applicant stated that the normal pH of the AP1000 SFP is within this range. The applicant stated that the complete lack of any chemical changes in these tests, combined with the knowledge of the effects of temperature and pH on corrosion rate, is sufficient to show that the aqueous pool environment, even for 60 years or more, will not detrimentally affect the condition of the Metamic® Panels Radiation testing of 31 w% B_4C with both gamma (1.5×10^9 Gy (1.5×10^{11} rads)) and fast neutron (1.7×10^{18} to 5.8×10^{19} n/cm^2) components. The conclusions of the post irradiation testing were that the Metamic® exhibited excellent dimensional stability after irradiation, and there was no change in Boron-10 areal density.

In response to the second part of the RAI, the applicant stated that an in-situ surveillance program to monitor the condition of the Metamic® in the racks will be implemented for the AP1000 spent fuel racks. The program uses representative material coupons, and is patterned after similar programs used for years at operating plants. The specific Metamic® monitoring program will be developed by the COL applicant. The applicant recommended the following tests to be performed on the coupons:

1) Neutron attenuation measurements (to verify the continued presence of boron)

    a. Acceptance criteria – A decrease of no more than 5 percent in Boron-10 content, as determined by neutron attenuation, is acceptable.

2) Thickness measurement (as a monitor of potential swelling)

    a. Acceptance criteria – An increase of thickness at any point should not exceed 10 percent of the initial thickness at that point

The applicant also included a markup for DCD Section 9.1.2.2.1 including a description of the Metamic® material (including all the information described above) and the qualification testing performed. The applicant also included a markup for a new COL Information Item 9.1.6.7 in Section 9.1.6 of the DCD, which reads as follows:

> The COL holder shall implement a spent fuel rack Metamic® coupon sampling or monitoring program when the plant is placed into commercial operation.

The staff reviewed the response to RAI-SRP9.1.2-CIB-01 and finds that the applicant provided an adequate description of the material.

The topical report SER for the operating plant placed conditions upon the use of the material; specifically, that a coupon surveillance program be implemented. The coupon surveillance program was to include the following attributes:

- size and types of coupons to be used (i.e., similar in fabrication and layout to the proposed insert including welds and proximity to stainless steel);
- technique for measuring the initial B_4C content of the coupons;
- simulation of scratches on the coupons;
- frequency of coupon sampling and its justification; and tests to be performed on coupons (e.g., weight measurement, measurement of dimensions (length, width and thickness), and B_4C content); these tests should also address, at a minimum, any bubbling, blistering, cracking, flaking, or areal density changes of the coupons, any dose changes to the coupons, or the effects of any fluid movement and temperature fluctuations of the pool water.

The coupon surveillance program approved for the operating reactor included visual examination and photography, measurement of weight and density, and measurement of the length and width dimensions in addition to thickness.

The acceptance criterion proposed by the applicant for ^{10}B content is a decrease of no more than 5 percent as determined by neutron attenuation, which is essentially the same as the acceptance criteria approved for the operating plant of any change in ^{10}B content of 5 percent. The staff finds the applicant's proposed criterion for ^{10}B content acceptable since it is consistent with that previously approved by the staff.

The acceptance criterion for thickness for the operating plant is any change in thickness of (+/-) 0.025 cm (0.01 in) for a 0.25 cm (0.1 in) coupon. The applicant's proposed acceptance criterion for thickness is no increase in thickness greater than 10 percent. The staff finds that the applicant's criterion is acceptable because it will detect significant swelling of the material thickness.

As part of the material qualification, the applicant also cited the results of radiation testing of 31 wt percent B_4C with both gamma (1.5×10^9 Gy (1.5×10^{11} rads)) and fast neutron (1.7×10^{18} to 5.8×10^{19} n/cm^2) components. An appendix to the HOLTEC report indicates that the gamma dose of 1.5×10^9 Gy (1.5×10^{11} rads) is roughly equivalent to the exposure Metamic® would receive in 40 years of actual fuel rack service. Although the AP1000 plant design life is 60 years, the staff finds the use of a gamma dose equivalent to 40 years exposure acceptable since the plant license duration will be 40 years, and the recommended monitoring program includes testing to verify the continued capability of the Metamic® materials to provide the required neutron absorption capacity. The HOLTEC report did not compare the fast neutron exposure expected in fuel pool service to the fast neutron exposure in the qualification program. However, the coupon monitoring program recommended by the applicant is likely to detect any degradation associated with fast neutron exposure.

The staff finds the referenced corrosion testing is appropriate. The accelerated corrosion testing resulted in essentially no corrosion of the material. Aluminum and aluminum alloys form a passive oxide film in most air or water environments that limits general corrosion to negligible rates. In the corrosion testing, some Metamic® coupons with a mill finish experienced pitting corrosion. The topical report summarized the results of a corrosion test program performed by Electrical Power Research Institute (EPRI). The duration of the EPRI corrosion tests was slightly over one year. The pitting in the EPRI tests was attributed to impurities present on the coupon surface based on the fact that the coupons cleaned by glass beading (as the AP1000 Metamic® material will be cleaned) or chemically cleaned prior to anodizing did not experience pitting. The corrosion testing performed by EPRI was verified by tests documented in the HOLTEC report performed in similar environments, but for a shorter duration (90 days). The applicant concluded the accelerated test results are sufficient to show that the aqueous SFP environment, even for 60 years, will not detrimentally affect the condition of the Metamic® panels. However, the applicant did not provide a quantitative basis for extrapolating the corrosion test results to 60 years. Although the staff agrees that corrosion appears to have been stopped by the formation of a passive film, due to the limited experience with Metamic® in operating reactors, the staff did not agree that a corrosion concern can be completely precluded for Metamic.

In the response to RAI-SRP9.1.2 CIB1-01, the applicant described the mounting and location of the coupons in the SFP, but did not provide the size. The applicant indicated that the coupons would be precharacterized for weight, dimensions (especially thickness) and ^{10}B loading, but did not provide: the technique for measuring the initial B_4C content; a recommended schedule for withdrawal and testing of the coupons; whether coupons included scratches; recommended tests to address bubbling, blistering, cracking, flaking, or areal density changes of the coupons; any dose changes to the coupons; or the effects of any fluid movement and temperature fluctuations of the pool water.

Since the applicant did not provide recommended criteria for several of the items addressed in the conditions on the use of Metamic® in the SER, and due to the limited experience with Metamic® material in operating reactors, particularly with regard to long-term corrosion behavior, in an April 28, 2008 supplement to RAI-SRP9.1.2-CIB1-01, the staff requested the following additional information:

> Provide a recommendation to the COL applicant for the following aspects of the Metamic® coupon surveillance program, and include the same information in the next revision to the DCD:
>
> - recommended coupon withdrawal schedule
> - size and types of coupons to be used (i.e., similar in fabrication and layout as the proposed insert including welds and proximity to stainless steel);
> - technique for measuring the initial B4C content of the coupons;
> - whether the coupons should include simulated scratches, or explain why simulated scratches are unnecessary.
> - tests to monitor bubbling, blistering, cracking, or flaking.

- test to monitor for corrosion, such as weight loss measurements and/or visual examination.

If any of these items are not recommended, provide a justification for excluding the item from the program.

In response to the supplementary request, the applicant submitted a revised response to RAI-SRP9.1.2-CIB1-01, dated June 20, 2008. The response to the supplementary request stated that the applicant and the COL applicants together are providing a Metamic® coupon surveillance program. The applicant is responsible for the design aspects of the Metamic® coupon surveillance program and the COL applicants are responsible for the programmatic aspects.

The supplemental response also stated the following items should be included in the site surveillance program, and indicated whether the applicant or the COL applicant was responsible for each item as follows:

- Recommended coupon withdrawal schedule-the applicant
- Size and types of coupons to be used (i.e., similar in fabrication and layout as the proposed insert including welds and proximity to stainless steel)- the applicant
- Technique for measuring the initial B4C content of the coupons-COL applicants
- Whether the coupons should include simulated scratches, or explain why simulated scratches are unnecessary-the applicant
- Tests to monitor bubbling, blistering, cracking, or flaking-COL applicants
- Test to monitor for corrosion, such as weight loss measurements and/or visual examination-COL applicants

The supplemental response further stated that this information is described in the COL holder's Metamic® coupon surveillance program, and that the applicant has worked with HOLTEC to design the Metamic® Coupon Tree requiring eight coupons for 60 years of surveillance. Based on this, the applicant has specified a coupon tree with 14 coupons (six additional coupons). The applicant provided a revision of the DCD markup from the original response to RAI-SRP9.1.2-CIB1-01 describing the number and size of the coupons, and showing the recommended coupon withdrawal schedule. The staff finds the information provided on the sizes and types of coupons is acceptable because the coupons are cut from the actual Metamic® absorber panels; therefore, the coupons are representative of the actual absorber panels, including the presence of any scratches. Therefore, scratches will not be deliberately added to the coupons. The staff also finds the proposed coupon surveillance schedule acceptable because it requires more frequent testing of the coupons early in plant life when problems are more likely to be detected, and covers the entire 60-year design life of the plant. The applicant also included a markup of DCD Table 1.8-2 showing COL Information Item 9.1-7 for a Metamic® Monitoring Program (also included in the original RAI response).
Section 9.1.6.7 of the proposed DCD markup included the text of the COL information item:

> The COL holder shall implement a spent fuel rack Metamic® coupon sampling or monitoring program when the plant is placed into commercial operation.

However, the staff found this wording did not include a sufficient level of detail to provide direction to the COL applicant with respect to the Metamic® Monitoring Program elements, as described in the supplemental RAI response.

The staff held a telephone conference with the applicant on July 11, 2008, to clarify whether the DCD would be revised to incorporate all the information provided in the supplemental response, and clarify the requirements for the Metamic® Monitoring Program. During the teleconference, the applicant agreed to provide a revised supplemental response that would include the text for COL Information Item 9.1-7 describing the elements of the Metamic® Monitoring Program for which the COL applicant is responsible. The revised supplemental response was received via letter dated August 21, 2008. In the revised response, the applicant provided a markup of DCD Section 9.1.6.7, which added details of the Metamic® Monitoring Program. The additional information stated that this program would include tests to monitor bubbling, blistering, cracking, or flaking and a test to monitor for corrosion such as weight loss measurements and or visual examination. However, the tests listed did not include the two tests originally proposed by the applicant in its original April 18, 2008, response to RAI-SRP9.1.2-CIB1-01, namely neutron attenuation and thickness tests.

In a response dated April 21, 2009, the applicant provided a revised markup of DCD Section 9.1.6.7 that includes the neutron attenuation and thickness tests in the COL information item text, in addition to those tests previously identified. The staff, therefore, considers 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 portion of COL Information Item 9.1-4 that addresses compatibility of the neutron absorbing materials used in the spent fuel racks with the spent fuel environment meets GDC 4. Therefore, the staff finds that the DCD changes regarding this issue, as proposed by the applicant in TR-65, are acceptable.

9.1.2.2.4.3 Conclusion

The staff has reviewed the AP1000 SFP criticality analysis and methodology as presented in TR-65 Revision 2 and concludes that the AP1000 SFP design is acceptable for spent fuel storage as described in the application and with the limitations as listed in this safety evaluation.

The staff finds the applicant's proposed resolution to AP1000 COL Information Item 9.1-4, which addresses the compatibility of the neutron absorbing materials used in the spent fuel racks with the spent fuel environment, meets GDC 4 and is, therefore, acceptable. Furthermore, the staff finds that the TR-65 conclusions regarding the evaluation for compatibility of the neutron absorbing materials used in the spent fuel racks with the spent fuel environment are generic and are expected to apply to all COL applications referencing the AP1000 DC.

9.1.2.3 Conclusion

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

In its previous evaluations of AP1000 DCD, Section 9.1.2, the staff identified acceptance criteria based on the design's meeting relevant requirements in GDC 2, GDC 4, GDC 5, GDC 61, GDC 62, and in GDC 63. The staff found that the AP1000 spent fuel pool cooling system (SFS) design was in compliance with these requirements of NUREG-0800 Section 9.1.2 and determined that the design of the AP1000 spent fuel storage, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 spent fuel storage as documented in AP1000 DCD, Revision 17 against the relevant acceptance criteria as listed above and in NUREG-0800 Section 9.1.1, and 9.1.2. The staff finds that the applicant's proposed changes do not affect the ability of the spent fuel storage to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. The staff concludes that the AP1000 new fuel storage design continues to meet all applicable acceptance criteria. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. The proposed changes to the spent fuel rack design and criticality analysis contribute to the increased standardization of the certification information in the AP1000 DCD and, thus, meet the requirements of 10 CFR 52.63(a)(1)(vii). Therefore, the staff finds that the proposed changes to AP1000 Section 9.1.2 are acceptable.

9.1.3 SFP Cooling and Purification

9.1.3.1 Summary of Technical Information

Section 9.1.3, "SFP Cooling System," of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In AP1000 DCD, Revision 17, the applicant proposed the following changes to Section 9.1.3 of the certified design:

1. The applicant proposed to increase the SFS pumps' common suction pipe diameter from 15.24 cm (6 in) to 25.4 cm (10 in) from the SFP to the penetration at the SFS pump room and then reduced from 25.4 cm (10 in) to 20.32 cm (8 in). Where the common suction pipe branches off to the individual SFS pumps, the pipe is reduced again, to 15.24 cm (6 in). The applicant documented this change in TR-103, "Fluid System Changes," APP-GW-GLN-019, Revision 2 of October 2007.

2. The applicant proposed to increase the SFS pumps' common discharge pipe diameter from 15.24 cm (6 in) to 20.32 cm (8 in). The applicant documented this change in TR-103.

3. The applicant proposed to increase the number of spent fuel storage locations in the SFP from 619 fuel assemblies to 889 fuel assemblies. The applicant documented this change in TR-103.

In the AP1000 DCD, Revision 17, the applicant submitted changes documented in its response to RAI-TR103-SBPA-01 on November 9, 2007. The additional changes included the following:

a. The applicant modified Section 9.1.3.1.3.1, "Partial Core," for the assumed heat load to be based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the SFP, including 44 percent of a core (69 assemblies) placed in the pool beginning at 120 hours after shutdown.

b. The applicant modified Section 9.1.3.1.3.2, "Full Core Off-Load," for the assumed heat load to be based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the SFP, plus one full core placed in the pool at 120 hours after shutdown.

c. The applicant modified Table 9.1-2, "SFP Cooling and Purification System Design Parameters," to reflect the new SFP storage capacity of 889 fuel assemblies.

d. The applicant modified Section 9.1.3.4.3, "Abnormal Conditions," with regard to the decay heat levels in the SFP and the amount of makeup water required to provide fuel pool cooling in the event of an extended loss of normal SFP cooling. The applicant also reduced the lengths of time when no makeup is needed and when safety-related makeup from the cask washdown area is sufficient to achieve SFP cooling from 7 days to 72 hours.

e. The applicant modified Table 9.1-4, "Station Blackout/Seismic Event Times," with regard to the event descriptions, time to saturation, and height of water above fuel at 72 hours and at 7 days. In addition, the applicant revised note 7 for Table 9.1-4.

f. As a result of design changes to the shield building, the availability of the water in the passive containment cooling water storage tank (PCCWST) for refilling the SFP has changed. This impacted the basis of several scenarios of the SFP thermal analysis. These changes include:

    (i) The applicant modified TS, DCD Chapter 16, Section 3.6.7, "Passive Containment Cooling System (PCS) – Shutdown," to lower the required reactor decay heat limit for air-only containment cooling from 9 megawatts thermal (MWt) (30.7 million British thermal units (MBtu)) to 6 MWt (20.5 MBtu).

    (ii) The applicant modified Section 9.1.3.4.3, "Abnormal Conditions," with regard to the required safety-related makeup water sources. The cask loading pit (CLP) is now credited as a safety-related makeup water source when the PCSWST is not available to provide safety-related makeup water to the SFP and the SFP heat load is higher than 5.6 MWt (19.1 MBtu) and less than 7.2 MWt (24.6 MBtu).

(iii) The applicant modified TS 3.7.9, "Fuel Storage Pool Makeup Water Sources," to verify that the CLP is available and communicated to the SFP before it is needed as a makeup water source.

(iv) The applicant modified Section 9.1.3.4.3 with regard to required makeup water flow from the passive containment cooling ancillary water storage tank (PCCAWST) for post 72 hour cooling. The change allows the system to adjust the makeup flow between 132.5 Lpm (liters per minute (35 gallons per minute (gpm)) and 189.3 Lpm (50 gpm) as needed.

(v) The applicant modified Table 9.1-4, with regard to the event descriptions and time to saturation at 72 hours and at 7 days.

4. The applicant proposed to raise the specified maximum allowable elevation for the bottom of a spent fuel assembly to be within 7.47 m (24 ft, 6 in) of the operating deck. The applicant documented a change to maximum allowable elevation in TR-121 and a subsequent change in AP1000 DCD, Revision 17.

   The applicant proposed to revise AP1000 DCD, Tier 1, Table 2.1.1-1, "Inspections, Test, Analysis, and Acceptance Criteria," and changed the acceptance criteria for design commitment number 5, to say, "the bottom of the dummy fuel assembly cannot be raised to within 24 ft, 6 in of the operating deck floor."

   The applicant also proposed to change the normal water level in the SFP from 0.610 m (2 ft) below the operating deck to 0.381 m (15 in) below the operating deck. This change results in an increase in normal water inventory in the SFP from 685,159 liters (181,000 gallons) to 721,121 liters (190,500 gallons).

5. The applicant proposed to modify the design basis refueling boron concentration to be 2700 parts per million (ppm). The applicant documented this change in TR-18, "AP1000 Core & Fuel Design," APP-GW-GLR-059 (WCAP-16652-NP), Revision 0 of October 2006.

6. The applicant proposed to revise the description of where the main suction line for the SFP cooling system connects to the SFP from "at an elevation 0.61 m (2 ft.) below the normal water level of the pool" to "at an elevation 1.83 m (6 ft.) below the operating deck."

   The applicant also revised the description of SFP alarms in the main control room (MCR) from stating that alarm in the MCR from safety-related instrumentation occurs "when water level reaches either the high level or low level setpoint" to "when level in the SFP reaches the low-low level setpoint."

7. The applicant modified the limiting site interface air temperatures in AP1000 DCD. These changes are evaluated in Section 2.3.1, "Regional Climatology"; Section 12.2, "Ensuring that Occupational Radiation Doses Are as Low as Is Reasonably Achievable"; and Section 12.4, "Radiation Protection Design" of this report. Section 9.1.3.1.3.1 and Section 9.1.3.1.3.2 were modified to reflect these changes. Additionally, Section 9.1.3.1.3.1 and Section 9.1.3.1.3.2 were modified to clearly describe when the different temperature limits are applicable to the SFS.

9.1.3.2 Evaluation

The staff reviewed all changes to the SFS in accordance with NUREG-0800 Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." The staff reviewed all changes identified in AP1000 DCD, Revision 17. The staff did not re-review descriptions and evaluations of the SFS in AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. All technical changes in the DCD are supported by information presented in the applicant's TRs, RAI responses, and the DCD itself.

The regulatory basis for AP1000 DCD, Section 9.1.3, is documented in NUREG-1793. The staff has reviewed the proposed changes to DCD Section 9.1.3 against the applicable acceptance criteria of NUREG-0800 Section 9.1.3. The following evaluations discuss the results of the staff's review.

The specific criteria that apply to the proposed DCD changes are; 10 CFR 52.63(a)(1)(vi), which concerns substantially increasing overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii), which concerns contribution to the increased standardization of the certification information in the AP1000 DCD.

9.1.3.2.1 SFS Pump Common Suction Pipe Diameter Increase

In Revision 17 of the AP1000 DCD the applicant proposed to increase the SFS pumps' suction pipe diameter. The basis for this change is documented in TR-103. In TR-103, Section II.B.6, the applicant states that the previously specified SFS pumps' suction pipe diameter of 15.24 cm (6 in) from the SFP to the individual pumps resulted in large pressure drops, which could cause cavitation in the SFS pump suction lines when the SFP temperature is elevated. The large suction line pressure drop created an unacceptable condition in which SFP cooling with the SFS pumps could have become incapable of restoration following a temporary loss of SFP cooling. The increase in suction line diameter will reduce the pressure drop in the suction line and increase the net positive suction head available (NPSHa) for SFS pump operation.

The staff finds that the safety function of the SFS continues to be met because the change does not affect SFP water level or makeup capability, and capability to keep the spent fuel assemblies cooled and covered with water is not affected by this change. In addition, the staff finds that the operational flexibility of the SFS pumps is increased because NPSHa will be adequate under a wider range of operating conditions. The staff finds that the system continues to comply with GDC 61 with regard to decay heat removal and capability to prevent reduction in fuel storage coolant inventory under accident conditions and that this change is needed because it increases the NPSHa for the SFS pumps and provides increased operational flexibility to support restart of the pumps following a loss of cooling event in which the SFP temperature becomes elevated. Therefore, the staff finds the proposed change to be acceptable.

9.1.3.2.2 SFS Pump Common Discharge Pipe Diameter Increase

In TR-103, Section II.B.6, the applicant states that the increase in the SFS pump common discharge pipe diameter, combined with the increased suction pressure provided by the increased SFS pumps' common suction pipe diameter, reduces the SFS pumps' required head and allows the pumps' horsepower at normal operating conditions to be lowered by

approximately 35 percent. This change provides additional operational flexibility by supporting a decrease in the required pump horsepower without degrading safety-related functions or operating margins in the SFS. The staff finds that the safety function of the SFS continues to be met because the change does not affect SFP water level or makeup capability, and capability to keep the spent fuel assemblies cooled and covered with water is not affected by this change. The staff finds that this change is acceptable because it provides additional operational flexibility for the SFS pumps, and the design continues to comply with GDC 61 with regard to decay heat removal and capability to prevent reduction in fuel storage coolant inventory under accident conditions. The staff noted that there was an AP1000 documentation discrepancy because this change, which is described in TR-103, Revision 2, was not reflected in AP1000 DCD, Revision 16.

In RAI-SRP9.1.3-SBPA-02, dated April 16, 2008, the staff requested that the applicant update the application to reflect the proposed change in SFS pump common discharge pipe diameter. In its response dated May 28, 2008, the applicant stated that the increase in SFS pump common discharge pipe diameter should have been reflected in Revision 16 of the AP1000 DCD and that it would be captured in the next revision of the AP1000 DCD. In DCD Revision 17, Figure 9.1-6 "Spent Fuel Pool Cooling System Piping and Instrumentation Diagram," the applicant proposed to change the SFS common discharge pipe diameter to 20.32 cm (8 in) as identified in the response to RAI-SRP9.1.3-SBPA-02. Therefore, the staff finds the applicant's response to RAI-SRP9.1.3-SBPA-02 to be acceptable.

9.1.3.2.3 Increase in Number of Spent Fuel Storage Locations

In TR-103, Section II.B.17, the applicant states that the storage capacity of the spent fuel storage racks in the SFP has been updated to provide 889 spent fuel storage locations, an increase of 270 from the previous 619 locations. As a result of the increased spent fuel storage capacity, the maximum decay heat input to the SFP is increased under various refueling offload conditions. The applicant updated the SFP thermal analysis to demonstrate that the SFS can maintain the stored fuel, cooled and submerged under water for 72 hours after the initiating event from safety-related sources and up to 7 days from internal sources.

During the June 25, 2009 audit, the staff requested clarification with respect to the AP1000 DCD, Revision 17 change in Figure 9.1-4, "Spent Fuel Storage Pool Layout (889 Storage Locations)." In RAI-SRP9.1.3-SBPA-15, the staff asked the applicant to explain the inconsistency in Figure 9.1-4 between Revision 16 and 17 in that Rack C1 contained an arrangement of 12 x 10 (minus 7 cells) assemblies in Revision 16 and 12 x 10 (minus 2 cells) assemblies in Revision 17.

In its response dated August 25, 2009, the applicant described that the Revision 16 Rack C1 label was incorrect and was corrected in Revision 17 to be arranged in a 12 x 10 module, with 2 cells missing in the North-South direction. The staff determined that this change was an editorial correction and not a design change. The rack description provided in the DCD is consistent with this correction. Based on the above discussion, the staff finds the applicant's response to RAI-SRP9.1.3-SBPA-15 to be acceptable and the issue is resolved.

In Revision 17 of the DCD the applicant updated Tier 2, Table 9.1-4 to reflect the calculated height of water above the fuel at 72 hours and at 7 days after the seismic event, for the three limiting offload scenarios. Table 9.1-4 contained a number of notes. Note 6 stated:

> Alignment of the PCS ancillary water storage tank and initiation of PCS recirculation pumps provide a makeup water supply to maintain this pool level or higher above the top of the fuel.

In Revision 15 of the DCD, this note only applied to SFP cooling for the period of time between 72 hours and 7 days. AP1000 DCD, Revision 17 added this note to the first offload scenario described in Table 9.1-4 for the period of time prior to 72 hours.

The staff determined that this change was inconsistent with the system description provided in the TS Basis for TS 3.7.9, and inconsistent with the staff's position documented in NUREG-1793 for AP600, AP1000 Revision15, SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs"; and SECY-98-161, "The Westinghouse AP600 Standard Design as it Relates to the Fire Protection and the Spent Fuel Pool Cooling Systems."

In Revision 1 to Open Item OI-SRP9.1.3-SBPA-13, the staff requested that the applicant clarify/justify if the AP1000 design is in accordance with the established staff position, or if the design is being changed to introduce a new design basis.

In its response dated August 20, 2010, the applicant stated that PCCAWST was never credited for SFP makeup prior to 72 hours, and the addition of Note 6 to the height of water above the fuel prior to 72 hours was an editorial error. The applicant's response also included a markup of Table 9.1-4 removing Note 6 from the height of water above the fuel prior to 72 hours.

Based on the applicant's response, the staff found the applicant's response acceptable and Revision 1 of Open Item OI-SRP9.1.3-SBPA-13 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.

In Revision 15 of the DCD Tier 2 Section 9.1.3.1.3.1, the applicant stated that the assumed partial core heat load is based on the decay heat generated by the accumulated fuel assemblies stored in the fuel pool for 10 years plus 44 percent of a core (68 assemblies) being placed into the pool. In Revision 17 of the DCD Tier 2 Section 9.1.3.1.3.1, the applicant states that the assumed partial core heat load is based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the fuel pool, which includes 44 percent of a core (69 assemblies) being placed into the pool. Additionally, in Revision 15 of the DCD Tier 2 Section 9.1.3.1.3.2, the applicant stated that the assumed full core offload heat load is based on the decay heat generated by the accumulated fuel assemblies stored in the fuel pool for 10 years, plus one full core placed in the pool. In Revision 17 of the DCD Tier 2 Section 9.1.3.1.3.2, the applicant states that the assumed full core offload heat load is based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the fuel pool, plus one full core placed in the pool.

The staff reviewed these changes and determined that by assuming that all SFP locations are filled, the calculated heat load bounds the worst possible offload scenario. The staff found that these changes were consistent with the increase in SFP capacity, were conservative in nature, and resulted in more limiting conditions. Therefore, the staff finds the proposed change to increase the total number of fuel assemblies used to calculate the heat load in the SFP for the partial core and the full core offload scenarios acceptable.

AP1000 DCD, Revision 15, Section 9.1.3, discusses three offload scenarios that represent the bounding SFP heat loads for all anticipated accident conditions. These decay heat loads are inputs for the thermal analysis of the SFP cooling.

- The first offload scenario postulates that a seismic event (concurrent with a station blackout) occurs while the reactor is operating immediately following a 44 percent core refueling. Since the reactor is operating when the event occurs, the thermal analysis assumes that the decay heat of the reactor is higher, or equal to 9 megawatt-hours MWh (30.7 MBtu); therefore, the PCCWST is reserved for containment cooling and it cannot be credited to provide safety-related makeup water to the SFP.

- The second offload scenario postulates that a seismic event (concurrent with a station blackout) occurs after a refueling is completed, and that this refueling occurred immediately following a previous 44 percent core offload. After a refueling is completed, the decay heat in the reactor is lower than 9 MWh (30.7 MBtu); therefore, the PCCWST can be credited to provide safety-related makeup water to the SFP.

- The third offload scenario postulates that a seismic event (concurrent with a station blackout) occurs after an emergency full core offload has been completed, and that this occurred immediately following a previous 44 percent core offload. This offload scenario represents the highest possible decay heat load in the SFP. Since the reactor is assumed to be empty, the PCCWST is credited for providing safety-related makeup water to the SFP.

The AP1000 DCD Revision 15 SFP is designed to use safety-related water sources to remove the SFP decay heat for the first 72 hours following events when the normal SFP cooling system is unavailable. For all the offload scenarios discussed above, the stored fuel in the SFP remains covered with water using only safety-related makeup water sources for the first 72 hours after the onset of the event. After the first 72 hours and before 7 days, the SFP credits the use of regulatory treatment of nonsafety systems (RTNSS) to provide makeup water to the SFP. The minimum water level necessary to achieve sufficient cooling of the stored fuel is the subcooled, collapsed water level (without vapor voids) required to cover the top of the fuel assemblies.

In AP1000 DCD Revision 15, the thermal analysis credits the water volume in the SFP (below the minimum water inventory level), the fuel transfer canal (including gate volume), the cask wash-down pit, and the PCCWST as the safety-related makeup water sources available for the first 72 hours of the event (depending on the offload scenario evaluated, some sources may not be available). Establishing makeup from the cask wash-down pit and the PCCWST requires operator action to re-position manual valves. The AP1000 DCD Revision 15 credits providing makeup water from the nonsafety-related water source in the PCCAWST to the SFP between 72 hours and 7 days after the event. This water source, piping segments, and the pumps are classified as RTNSS Class B in Westinghouse Commercial Atomic Power (WCAP)-15985, Revision 2, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," dated August 2003, and have availability controls.

In AP1000 DCD Revision 17, the applicant proposed several changes to Section 9.1.3 related to the thermal analysis. The basis for these changes were documented in; TR-103; TR-54; TR-65; TR-105, APP-GW-GLN-105, "Building and Structure Configuration, Layout, and General Arrangement Design Updates" Revision 2 of October 2007; and TR-108, APP-GW-GLN-108, "AP1000 Site Interface Temperature Limits," Revision 2 of September, 2007.

The staff reviewed all of these TRs and the SFP thermal analysis report documented in APP-SFS-M3C-012, "AP1000 Spent Fuel Pool Heatup, Boiloff, and Emergency Makeup on Loss of Cooling," and determined that additional information was needed. The staff submitted RAI-TR103-SBPA-01, RAI-SRP9.1.3-SBPA-04, RAI-SRP9.1.3-SBPA-08, RAI-SRP9.1.3-SBPA-13, and RAI-SRP9.1.3-SBPA-05; requesting additional information related to the SFP thermal analysis inputs, assumptions, methodology and results. The staff's concerns related to the SFP thermal analysis inputs, assumptions, and results were identified as Open Item OI-SRP9.1.3-SBPA-04.

In response to RAI-TR103-SBPA-01 and RAI-SRP9.1.3-SBPA-04, the applicant stated that the conditions assumed for the calculated decay heat levels are the off-load conditions described in AP1000 DCD Sections 9.1.3.1.3.1 and 9.1.3.1.3.2 and that the calculated values are representative of the limiting off-load conditions as described in the applicable AP1000 DCD sections.

During the June 25, 2009 regulatory audit of the SFP thermal analysis report, the staff identified that the thermal analysis credited non-conservative assumptions related to the initial SFP water level. The staff identified this issue as Open Item OI-SRP9.1.3-SBPA-08(b), and requested that the applicant correct or justify these report findings. The applicant responded to the staff's question by revising the thermal analysis report eliminating the non-conservative assumption. On December 8, 2009, the staff performed a regulatory audit of the revised thermal analysis report and confirmed that the initial SFP water level after a seismic event had been reduced. This reduction in water level eliminated the non-conservative assumption identified by the staff in Open Item OI-SRP9.1.3-SBPA-08(b); therefore, the staff considers Open Item OI-SRP9.1.3-SBPA-08(b) closed.

The applicant also confirmed that there is no nonsafety-related piping connections in the SFP below an elevation of 39.27 m (128.83 ft), which is the minimum water level assumed in the SFP thermal analysis report. The SFP piping that extends below an elevation of 39.27 m (128.83 ft) are equipped with anti-siphon devices that prevent draining the SFP below the minimum inventory limit and are designed to be capable of performing their safety function following a design basis seismic event. During the audit of the applicant's SFP thermal analysis report, the staff verified that the analysis assumes that the initial water level is 39.27 m (128.83 ft). Therefore, the staff finds that the SFP thermal analysis was performed using assumptions in accordance with the system design and the system description provided in DCD Section 9.1.3.

The applicant subsequently revised the thermal analysis by introducing a change in methodology. The new methodology assumed that the boiling temperature of the water in the SFP would be affected by the pressure produced by the elevation of the column of water. This assumption allowed the SFP water temperature to rise above 100 °C (212 °F) before boiling. The staff determined that the applicant had not properly justified this assumption and requested that the applicant provide justification for this assumption or revise the thermal analysis calculation.

In response the applicant revised the thermal analysis removing the assumption that the SFP water temperature would rise above 100 °C (212 °F) before boiling. On January 25, 2010, the staff performed a regulatory audit of this revised thermal analysis and confirmed that this assumption had been removed. Therefore, the staff finds this acceptable, since the thermal analysis no longer changes the previously approved methodology.

Since the staff audits confirmed that the revised SFP thermal analysis used an approved methodology and used conservative inputs and assumptions, the staff considers Open Item OI-SRP9.1.3-SBPA-04 resolved.

In Revision 17 of the DCD the applicant updated Tier 2, Table 9.1-4, "Station Blackout/Seismic Event Times," to reflect the results of the revised thermal analysis calculation. The table showed that the time to boil for all three limiting offload scenarios had decreased. For the most limiting scenario (full core offload) the time to boil decreased from 2.5 hours to 1.37 hours. In a later revision to the SFP thermal analysis report, the time to boil for this offload scenario was raised to 2.33 hours. The staff's evaluation of this increase in time to boil is evaluated further below in this SE section.

In RAI-SRP9.1.3-SBPA-13, the staff requested that the applicant update the DCD in order to address the impact of the initial decrease in SFP time to boil on the required operator actions needed to cope with this event. The staff also pointed out that due to changes in the SFP thermal analysis, the information in Note 8 of Table 9.1-4 needed to be updated.

The applicant responded to RAI-SRP9.1.3-SBPA-13 in letters dated August 25, 2009, and February 10, 2010. The applicant's response proposed to modify Note 8 to properly represent the revised thermal analysis. The staff found this acceptable since the DCD table is now consistent with the revised thermal analysis. The applicant's response also stated that under most limiting conditions with the highest SFP decay heat, the operator will have more than 18 hours after boiling has begun to establish safety-related makeup. In addition, the applicant proposed to add a new Note 9 to Tier 2, Table 9.1-4 applicable to all off-loading scenarios analyzed.

The proposed Note 9 stated "operator action to align makeup water to the spent fuel pool must occur within 18 hours of the seismic event." The staff determined that the proposed wording of Note 9 did not clearly reflect the minimum time available for operator action. This was identified as Open Item OI-SRP9.1.3-SBPA-13 in the SER with open items.

In letter dated August 6, 2010, the applicant submitted for staff review APP-GW-GLR-096 Revision 1, (Proprietary) and APP-GW-GLR-097 Revision 1 (Non-Proprietary), "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses." In a letter dated March 10, 2011, the applicant submitted for staff review APP GW GLR 096 Revision 2, (Proprietary) and APP GW GLR 097 Revision 1 (Non Proprietary). These letters also included markups of the DCD sections impacted by this change. These changes to the Shield Building impacted the availability of the safety-related water source in the PCCWST to be used as makeup to the SFP for two of the offload scenarios discussed in the DCD. The evaluation of the shield building design changes are evaluated in Section 6.2 of this report. This section evaluates the impact of these changes on the SFS.

The DCD markups included a proposed revision to Note 9 of Tier 2, Table 9.1 4. The revised note states that "[a] minimum of 18 hours is available for operator action to align makeup water to the spent fuel pool after a seismic event." The revised Note 9 clearly states the minimum time that the operator has to perform the required actions to align safety-related makeup water to the SFP. The staff finds that the applicant's calculation demonstrates that the operator will have sufficient time to take the required actions to align the SFP makeup sources to prevent the SFP boildown to a water level that would uncover the stored fuel. Therefore, the staff concerns identified in Open Item OI-SRP9.1.3-SBPA-13 are considered closed. In a subsequent revision

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

AP1000 DCD, Revision 15, TS 3.6.7, "Passive Containment Cooling System (PCS) - Shutdown," required that while the reactor decay heat is at or higher than 9 MWh (30.7 MBtu) , the PCCWST is reserved for containment cooling. When the decay heat in the reactor is below 9 MWh (30.7 MBtu), the PCCWST is credited for providing SFP safety-related water makeup. As a result of the design changes to the shield building described in the August 6, 2010 letter, the PCCWST is now reserved for containment cooling while the reactor decay heat is at or higher than 6 MWh (20.5 MBtu).

AP1000 DCD, Revision 15, TS 3.7.9, "Fuel Storage Pool Makeup Water Sources," states that the PCCWST is required in order to increase the SFP heat load above 5.4 MWh (18.4 MBtu). The PCCWST is not available for both containment cooling and SFP makeup simultaneously. Once enough fuel has been transferred to the SFP to raise the decay heat in the SFP to 5.4 MWh (18.4 MBtu), the PCCWST needs to be available as a safety-related source for SFP makeup. However, during normal refueling operations, the reactor decay heat would still be greater than 6 MWh (20.5 MBtu); therefore, the PCCWST would not be available for SFP makeup since it is still required as safety-related source for containment cooling. TS 3.7.9 prevents additional fuel offloading until the PCCWST can be credited to provide safety-related makeup water to the SFP. In order to reduce the impact of this change on the refueling schedule, the applicant proposed to credit the CLP as a safety-related makeup water source.

The applicant revised the SFP thermal analysis to take credit for the water stored in the CLP as a safety-related makeup water source. While revising the SFP thermal analysis, the applicant identified an error in the calculation. The calculation erroneously assumed that at the onset of the initiating event, the SFP already had received 1 hour of decay heat. This error affects all of the calculated times to boil. On July 16, 2010, the staff audited the most recent and the previous thermal analysis calculation and confirmed that previous thermal analysis had also included this error. The applicant also identified that even after the PCCWST is required to be available for the SFP makeup, the PCCWST isolation valves (V001A/B/C) can still be automatically actuated. If the PCCWST isolation valves are opened the water would be drained onto containment instead of being sent to the SFP. The operators have 24 hours to take action to close the valves until Surveillance Requirement (SR) 3.7.9.1 is violated (the PCCWST volume is drained to $< 1.514 \times 10^6$ liters (L) (400,000 gallons)). To prevent inadvertent actuation of these valves, the applicant proposed to modify SR 3.7.9.1 to ensure that one motor-operated valve (MOV) isolation valve (gate valve) is closed and secured prior to the PCCWST becoming operable for SFP makeup. The PCCWST air-operated isolation valves (V001A/B) cannot be used because they are fail-open. During a loss of onsite and offsite power the valves will lose compressed air and eventually open.

The staff evaluated the justification and the markups included in APP-GW-GLR-096, Revision 1, and determined that this proposed change to SR 3.7.9.1 provides assurance that the safety-related makeup water volume needed for SFP cooling would be available when needed. The staff evaluated APP GW GLR 096, Revision 2 and determined that this revision did not introduce any new changes to the information discussed above. Therefore, the staff finds the proposed change to SR 3.7.9.1 acceptable.

The revised thermal analysis concluded that the CLP contains sufficient water to allow the SFP decay heat limit to be raised 7.2 MWh (24.6 MBtu). The report also identified that if the accident scenario were to occur during a refueling outage, when the reactor decay heat is higher than 6

MWh (20.5 MBtu) and the SFP heat load is below 7.2 MWh (24.6 MBtu), the PCCASWT flow rate limits will not be sufficient for maintaining the stored fuel covered. DCD Section 6.2.2.4.2, "Preoperational Testing," and Section 9.1.3.4.3, "Abnormal Conditions," states that the PCCASWT has the capability of providing a total of 378 Lpm (100 gpm) to the PCCWST for containment cooling while providing 132 Lpm (35 gpm) to the SFP for SFP cooling. With the higher SFP heat load, the boiloff rate at 72 hours is higher and the rate of water makeup is also higher.

APP-GW-GLR-097, Revision 1 described the scenario in which the SFP heat load is at its highest, and that this peak is not coincident with the peak demand for containment cooling. The PCCAWST provides SFP water makeup and containment cooling, simultaneously, for the period of time between 72 hours and 7 days after the onset of the event. The applicant proposed to revise DCD Sections 6.2 and 9.1.3 to specify that the makeup flow from the PCCAWST will be throttle/adjusted to provide 303 Lpm (80 gpm) to the PCCWST and over 189 Lpm (50 gpm) to the SFP when additional flow is required in the SFP. The applicant stated that these new flow rates provide sufficient cooling for both the containment and the SFP cooling.

The staff confirmed that a flow rate of 189 Lpm (50 gpm) is higher than the anticipated boiloff rate at 72 hours (for this offload scenario). The staff also verified the proposed update to DCD Section 9.1.3.4.3 presented in the DCD markups and determined that the new description of the system operation is in accordance with the new design. Therefore, the change was found 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 applicant also proposed to modify TS 3.7.9 to reflect the new thermal analysis assumptions. The modified TS require the CWP to be operable when the SFP decay heat load is higher than or equal to 4.6 MWh (15.7 MBtu) and less than 7.2 MWh (24.6 MBtu). The change adds the requirement of having the CLP available when the SFP decay heat load is higher than 5.6 MWh (19.1 MBtu) and less than or equal to 7.2 MWh (24.6 MBtu). The PCCWST is required as SFP makeup water source if the SFP decay heat is higher than 7.2 MWh (24.6 MBtu) and the reactor decay heat load is less than 6 MWh (20.5 MBtu). The applicant proposed to modify DCD Section 9.1.3, TS surveillance requirements, and the TS basis to reflect these changes. The staff identified an apparent inconsistency between TS 3.7.9 and DCD Section 9.1.3, related to the maximum decay heat that the spent fuel pool is capable of dissipating in 72 hrs with no additional makeup. In a letter dated February 25, 2011, the applicant proposed to change all references to the maximum decay heat limit for the SFP with no makeup available from 4.6 MWh (15.7 MBtu) to 4.7 MWh (16.0 MBtu). The staff found the new limit is in accordance with the SFP thermal analysis report previously evaluated by the staff and, therefore, the proposed changes were found acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text,which resolve this issue.

Establishing makeup water flow from the CLP following the initiating event requires no operator action. The applicant proposes to establish makeup water flow by opening the gate that separates the SFP and the CLP. New SR 3.7.9.4 requires verification that the CLP water level is at or higher than 4.2 m (13.75 ft) (minimum level) and that the CLP and the SFP are in communication, prior to exceeding the new SFP decay heat load limit of 5.6 MWh (19.1 MBtu). The staff finds this proposed SR 3.7.9.4 provides assurance that the required safety-related makeup water is going to be available when needed; therefore, the staff finds the proposed

SR 3.7.9.4 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 finds that the applicant's thermal analysis has demonstrated that, with the increased spent fuel storage capacity, sufficient water inventory and sufficient makeup capability are available to keep the spent fuel covered with water under all limiting conditions, consistent with the safety functions described in DCD Section 9.1.3.5, and in accordance with NUREG-0800 Section 9.1.3. As a result of the SFP capacity increase, the time to reach saturation and the height of water coverage over stored spent fuel has changed.

For the first offload scenario (seismic event occurred while the reactor is at power immediately following a 44 percent refueling), saturation is reached in 7.38 hours. During the first 72 hours, crediting only safety-related sources, the height of water above the fuel is maintained at 0.42 m (1.4 ft). Between 72 hours and 7 days, the applicant credits the use of the PCCAWST (a RTNSS Class B system) to provide makeup water to the SFP and maintain the height of water above the fuel at 0.42 m (1.4 ft).

For the second limiting offload scenario (seismic event occurred after refueling, immediately following a 44 percent refueling), saturation is reached in 5.59 hours. If the decay heat inside the reactor is at or higher than 6 MWh (20.5 MBtu), the PCCWST is not available to provide safety-related makeup water to the SFP, and TS 3.7.9 limits the SFP heat load to 7.2 MWh (24.6 MBtu). The remaining safety-related water sources have sufficient water inventory to cover the stored fuel for 72 hours. If the decay heat inside the reactor is below 6 MWh (20.5 MBtu), the PCCWST is available to provide safety-related makeup water to the SFP. Therefore, offloading operations can continue and the SFP decay heat load can be higher than 7.2 MWh (24.6 MBtu). During the first 72 hours, crediting only safety-related sources; the height of water above the fuel is maintained at 1.3 m (4.2 ft). Between 72 hours and 7 days, the applicant credits the use of the PCCAWST to provide makeup water to the SFP and maintain the height of water above the fuel at 1.3 m (4.2 ft).

For the third limiting offload scenario (seismic event occurred after completing and emergency full core offload, immediately following a 44 percent refueling), saturation is reached in 2.33 hours. There is no fuel in the reactor; therefore the PCCWST is available to provide safety-related makeup water to the SFP. During the first 72 hours, crediting only safety-related sources, the height of water above the fuel is maintained at 2.43 m (8 ft). Between 72 hours and 7 days, the applicant credits the use of the PCCAWST to provide makeup water to the SFP and maintain the height of water above the fuel at 2.43 m (8 ft).

With the increased number of spent fuel assemblies, the staff finds that the SFP continues to maintain water coverage above the spent fuel assemblies for at least 72 hours following a loss of the nonsafety-related SFP cooling system, using only safety-related makeup water, and that adequate time is available for operators to establish the required makeup water.

The staff also finds that the SFP continues to maintain water coverage above the spent fuel assemblies for at least 7 days following a loss of the nonsafety-related SFP cooling system, using RTNSS "B" makeup water, and that adequate time is available for operators to establish makeup water from on-site sources. Because adequate cooling and coverage of spent fuel bundles is maintained, the staff finds that the SFS continues to comply with requirements of GDC 61 related to provisions for decay heat removal and capability to prevent reduction in fuel storage coolant inventory under accident conditions. Based on the capability of the SFP and the SFS to maintain adequate cooling for spent fuel under limiting conditions and continued

compliance with requirements of GDC 61, the staff finds the proposed change to credit the CLP as a safety-related makeup water source, and the increase in SFP spent fuel locations from 619 to 889 locations, acceptable.

9.1.3.2.4 Increase in Maximum Allowable Elevation of a Spent Fuel Assembly and Increase in Normal SFP Water Level

In AP1000 DCD Revision 17 the applicant proposed to increase the maximum allowable elevation of a spent fuel assembly and increase the normal SFP water level. Both Tier 1 and Tier 2 information are affected by these changes. The justification for these changes is included in TR-121.

To support the increased fuel assembly height, the applicant proposed to increase the specified normal SFP water level from 40.61 m (133.25 ft) to 40.92 m (134.25 ft). In AP1000 DCD, Section 9.1.3.1.4, "Spent Fuel Purification," the applicant revised the discussion of exposure rates to say, "The spent fuel pool cooling system is designed to limit exposure rates to personnel on the SFP fuel handling machine to less than 2.5 millirem per hour." In AP1000 DCD, Table 9.1-2, "Spent Fuel Pool Cooling and Purification System Design Parameters," the applicant updated the SFP water volume to 724,906 L (191,500 gallons), including racks without fuel, at a water level of 30.48 cm (12 in) below the operating deck. Previously, in AP1000 DCD, Revision 15, the SFP water volume was stated as 685,159 L (181,000 gallons), including racks without fuel, at a water level of 76 cm (30 in) below the operating deck.

The staff noted the applicant's statement in TR-121 that SFP water level increases .31 m (1 ft) from 40.61 m (133.25 ft) to 40.92 m (134.25 ft) is inconsistent with the applicant's statement that before the proposed change the normal SFP water level is 76 cm (30 in) below the operating deck, and after the proposed change the normal SFP water level would be 30.48 cm (12 in) below the operating deck. In RAI-SRP9.1.3-SBPA-03, dated April 16, 2008, the staff asked the applicant to clarify the amount of increase in normal SFP water level and to discuss how the water level change impacts previous analyses.

In its response dated May 28, 2008, the applicant stated that the correct SFP water level is 40.92 m (134.25 ft), and that this is consistent with a water level that is 30.48 cm (12 in) below the operating deck. The applicant stated that Revision 15 of the AP1000 DCD was inconsistent in that it stated that the water level was at an elevation 40.61 m (133.25 ft), but it also stated that the water level was 76 cm (30 in) below the operating deck. The applicant stated that TR-121 and Revision 16 of the DCD corrected this inconsistency and that all affected analyses were included in TR-121 and were performed assuming an SFP water level of 40.92 m (134.25 ft). The staff noted that in Revision 17 to the DCD, the applicant had changed the normal SFP water level from 30.48 cm (12 in) below the operating deck to 38.1 cm (15 in) below the operating deck, which correlates to a SFP water volume decrease of approximately 3785 L (1000 gallons). During the June 25, 2009 audit, the staff asked the applicant to verify that the Revision 17 change in normal SFP water volume did not impact the calculations in APP-SFS-M3C-012 and to discuss all assumptions that changed in the decay heat calculations.

The applicant clarified that the decay heat calculations in APP-SFS-M3C-012 are based on a SFP water level that correlates to the lowest non-seismic component connected to the SFP, because the assumption is that all non-seismic components attached to the SFP will fail. The lowest non-seismic component connected to the SFP is the SFS main suction line, which is below the normal operating water level.

Based on its review, the staff finds the applicant's response acceptable because the applicant corrected the inconsistency between SFP water level elevation and SFP surface location below the operating deck. On the basis that the updated values, as stated in TR-121 and Revision 17 of the AP1000 DCD, are consistent and these values have been evaluated against the decay heat analysis calculations in APP-SFS-M3C-012, the staff's concerns described in RAI-SRP9.1.3-SBPA-03 are resolved.

The staff noted that in Revision 17 to the DCD the applicant had increased the minimum combined water volume of the SFP and fuel transfer canal. In RAI-SRP9.1.3-SBPA-07 the staff asked the applicant to justify the increase in minimum combined water volume of the SFP and fuel transfer canal from 176,778 L (46,700 gallons) in Revision 16 of DCD Table 9.1-2 to 490,210 L (129,500 gallons) in Revision 17 of DCD Table 9.1-2, and to identify the effects, if any, on the decay heat calculations. During the June 25, 2009 audit, the applicant clarified that the Revision 16 value was an error and the decay heat calculations assumed the minimum combined water volume of 490,210 L (129,500 gallons) in Revision 17 of DCD Table 9.1-2. The staff verified this information and subsequently withdrew RAI-SRP9.1.3-SBPA-07.

The staff's evaluation in Section 9.1.3 of this report is limited to consideration of the operating and thermal hydraulic performance characteristics of the SFS. The proposed change to increase the maximum allowable elevation of a spent fuel assembly does not affect the operating and thermal hydraulic performance characteristics of the SFS. The heat input to the SFP and heat removal capability of the SFS is not affected by this change. The staff finds the proposed increase in maximum allowable elevation of a spent fuel assembly to be acceptable from the standpoint of effects on operation and thermal hydraulic performance of the SFS. The impact of this change on fuel handling is evaluated by the staff in Section 9.1.4 of this supplement to NUREG-1793. The impact of this change on radiation exposure of operating personnel is evaluated by the staff in Section 12.2 and Section 12.4 of this supplement to NUREG-1793.

The staff reviewed the proposed change to increase the available SFP coolant inventory and the NPSH available at the suction of the SFS pumps. The staff finds that these changes improve or enhance the previously available operating margins for the SFS pumps under normal operating conditions. Under conditions where loss of SFP cooling might occur, the staff finds that the increased water inventory in the SFP allows for a longer time before the SFP would reach saturation and provides longer times for operators to take corrective actions to reestablish SFP cooling. The staff finds that the SFS continues to comply with requirements of GDC 61, related to provisions for decay heat removal and capability to prevent reduction in fuel storage coolant inventory under accident conditions. Because operating margins are improved and the system continues to conform to requirements of GDC 61, the staff finds that the proposed increase in SFP water level is acceptable with regard to effects on the operating and thermal-hydraulic performance characteristics of the SFS.

9.1.3.2.5 Increase in Specified Design Basis Refueling Boron Concentration

In TR-18, Revision 0, the applicant proposed a change in the specified design basis refueling boron concentration, increasing the specified boron concentration from 2500 ppm to 2700 ppm in the SFP water. The applicant stated that this change is for consistency with the accumulator boron concentration value provided in the DCD Chapter 15 accident analysis. The staff noted that the accumulator boron concentration value stated in AP1000 DCD, Chapter 15 accident analysis is 2600 ppm, not 2700 ppm.

In RAI-SRP9.1.3-SBPA-01, dated April 16, 2008, the staff asked the applicant to clarify what the correct value for boron concentration is and to resolve any inconsistencies in the documentation.

In its response dated May 28, 2008, the applicant stated that the correct SFP boron concentration is 2700 ppm and that this is consistent with values stated in Section 9.1.3.2 and Table 9.1-2 of Revision 16 of the AP1000 DCD and the in-containment refueling water storage tank (IRWST) boron concentration. The applicant stated that the accumulator boron concentration in Chapter 15 is stated as 2600 ppm and that this is consistent with TS SR 3.5.1.4, which requires the boron concentration in each accumulator to be between 2600 and 2900 ppm.

In its evaluation, the staff noted that the change in SFP boron concentration from 2500 ppm to 2700 ppm is consistent with the allowable boron concentration values specified for the accumulators credited in the DCD Chapter 15 accident analysis. The staff noted that the accumulator boron concentration value stated in the DCD Chapter 15 accident analysis is 2600 ppm because that is the minimum boron concentration permitted by TS SR 3.5.1.4. The staff finds this provides a satisfactory explanation of what is meant by the statement that "this change is made for consistency with the accumulator boron concentration values." Therefore, the staff finds the applicant's response to be acceptable and the staff's concern in RAI-SRP9.1.3-SBPA-01 is resolved.

The reactivity control effects of this change are evaluated in Section 4.3 of this supplement to NUREG-1793. The effects of this change on the criticality evaluation of the stored spent fuel are evaluated in Section 9.1.2 of this report. With regard to operational and thermal-hydraulic performance of the SFP cooling system, the staff finds this change acceptable because the boron content of the SFP water has no effect on the heat input to the SFP, the heat removal capability of the SFS, the operating margins or the performance characteristics of the SFS.

9.1.3.2.6 Changes to Piping Diagrams for the Spent Fuel Pool Cooling and Safety-Related Instrumentation

The staff noted that, in the AP1000 DCD Revision 17, Figure 9.1-5, "Piping Diagrams for Spent Fuel Pool Cooling (Normal Operation)," the applicant removed branch lines shown to and from the cask pit, changed the SFP connection to chemical and volume control system (CVS) from a separate penetration to a connection shared with the SFS pump suction, deleted the return line from the SFS discharge going to the in-containment refueling water storage tank, and changed several valve types on the drawing.

In RAI-SRP9.1.3-SBPA-10, the staff requested the applicant to provide justification for the changes mentioned above and to discuss whether any of these changes impact the safety conclusions. Also, for the CVCS connection, the staff requested the applicant to clarify whether the previously used SFP penetration has been removed entirely or whether it remains as "unused and capped," which is what the revised drawing appears to indicate. This was identified as Open Item OI-SRP-9.1.3-SBPA-10 in the SER with open items.

During the June 25, 2009 audit, the applicant explained that deleting the cask pit branch lines was not a change in design. The cask pit was originally designed with a common drain line; and the drawing was corrected to represent the actual design. With respect to the CVCS connection, the change also represented a correction in the drawing, and not a design change.

The design of the CVCS never had a penetration in the SFP. It was designed to connect to the SFS pump suction line, as represented in DCD Revision 17, Figure 9.1-5.

In a letter dated September 17, 2009, the applicant stated that none of the clarifications on Figure 9.1-5 represented changes to the safety conclusions for the AP1000 SFS. These changes were introduced to correct discrepancies between Figure 9.1-6 (which represented the correct design of the SFS) and Figure 9.1-5, which included errors.

The staff reviewed the applicant's response and determined that none of the changes in Figure 9.1-5 represents a technical change; they are corrections to represent the actual SFS design as described in Section 9.1.3. The staff considers Open Item OI-SRP9.1.3-SBPA-10 resolved.

In AP1000 DCD Revision 17, Section 9.1.3.7.D, the applicant proposed to change the description of MCR alarms. In DCD Revision 15, the applicant stated that safety-related instrumentation is provided to give an alarm in the MCR when the water level in the SFP reaches either the high level or low level setpoint. In DCD Revision 17, the applicant states that safety-related instrumentation is provided to give an alarm in the MCR when the water level in the SFP reaches the low-low-level setpoint. The staff issued RAI-SRP9.1.3-SBPA-11, which reads as follows:

a) Provide the basis for the safety-related level instrumentation change,

b) Clarify whether any MCR or local alarms are available to give an alarm on high level or on low level setpoints in the SFP, as previously described in AP1000 DCD, Revision 15, and

c) Justify the impact of the change in SFP level alarms to previously performed safety evaluations or operator response evaluations.

During the June 25, 2009 audit, and in an RAI response dated October 2, 2009, the applicant clarified that only the safety-related low-low level alarm is located in the safety-related instrumentation; the high and low level setpoint signals come from nonsafety-related instrumentation and, therefore, are not referred to as safety-related instrumentation in Revision 17 of the DCD. This was identified as Open Item OI-SRP9.1.3-SBPA-11 in the SER with open items.

In the RAI-SRP9.1.3-SBPA-11 response letter dated October 2, 2009, the applicant clarified that the high and low level alarms were not removed from the design; these alarms will be available to alert operators of SFP water level fluctuations. In the RAI response, the applicant proposed to change DCD Tier 2, Section 9.1.3.7 to clarify that the nonsafety-related instrumentation and alarms are available to alert operators. The staff verified that the High and Low level alarms were never intended to be safety-related alarms; therefore, the change in instrumentations does not impact the previous safety conclusion. Thus, Open Item OI-SRP-9.1.3-SBPA-11 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff noted that, in the AP1000 DCD Revision 17, Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves," the applicant lists each containment penetration and provides a summary of the containment isolation characteristics. In DCD Table 6.2.3-1 Sheet 2 of 4, the applicant identifies the containment isolation valves related to the SFP. Valve

SFS-PL-V067 is a pressure release valve located between SFS-PL-V034 and SFS-PL-V035. In DCD Tier 1 Table 2.2.1-1, the applicant also identifies the same pressure release valve but DCD Tier 2 Figure 9.1-6, "Spent Fuel Pool Cooling System Piping and Instrumentation Diagram," Sheet 1 of 2 does not show valve SFS-PL-V067. In RAI-SRP9.1.3-SBPA-12, the staff asked the applicant to update Figure 9.1-6, Sheet 1 to include valve SFS-PL-V067.

In its response dated August 25, 2009, the applicant agreed to revise DCD Figure 9.1-6 in the next DCD revision and illustrated the change in its response. The staff finds the change 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.

9.1.3.2.7 Modification of the limiting site interface air temperatures

The applicant modified the wet bulb air temperature in AP1000 DCD, Section 9.1.3.1.3 and the discussion on exposure rates to personnel in Section 9.1.3.1.4. These changes are evaluated in Section 2.3.1, "Regional Climatology"; Section 12.2, "Ensuring that Occupational Radiation Doses Are as Low as Is Reasonably Achievable"; and Section 12.4, "Radiation Protection Design," of this report. The site temperature (wet and dry bulb) impacts the cooling tower performance, which affects the temperature of the component cooling water system (CCS). The SFS heat exchanger (HX) is cooled by the CCS, and a change in the CCS temperature affects the performance of the SFS.

Westinghouse Impact Document 36 (APP-GW-GLE-036), "Impact of a Revision to the Current Wet Bulb Temperature Identified in Table 5.0-1 (Tier 1) and Table 2-1 (Sheet 1 of 3) of the DCD (Revision 16)," Revision 0 of June 27, 2008, documents the impact of increasing the maximum safety wet bulb non-coincident temperature from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F) and the maximum safety wet bulb coincident temperature from 26.7 °C (80 °F) to 30.1 °C (86.1 °F). The TR indicates that the SFP heat removal capability has been affected by this change. The thermal analysis results presented in the TR demonstrate that the SFS is capable of maintaining the SFP below 48.9 °C (120 °F) following a partial core fuel shuffle refueling, with the wet bulb temperature at the maximum safety limit (most limiting case). For a full core offload scenario, the thermal analysis demonstrated that the SFS is capable of maintaining the SFP at or below 48.9 °C (120 °F) based upon a service water heat sink at a maximum normal ambient wet bulb temperature at or below 60 °C (140 °F).

During the July 16, 2010, regulatory audit of the thermal calculation the staff discussed with the applicant the need to clarify DCD Tier 2 Section 9.1.3.1.3.1 "Partial Core." This section describes that the thermal analysis for the SFP cooling partial core scenario is based on a CCS supply temperature limited by the maximum normal ambient wet bulb temperature. The staff considered that this statement was not entirely correct, as while the reactor is at power, the CCS temperature limit is based on the maximum safety ambient wet bulb temperature.

In Open Item OI-SRP9.1.3-SBPA-13, Additional Question 3, the staff requested that the applicant clarify that DCD Section 9.1.3.1.3.1 properly represents the thermal analysis basis of the SFS. In a response letter dated August 20, 2010, the applicant presented the markups for modifying DCD Section 9.1.3.1.3.1 and Section 9.1.3.1.3.2 to clearly indicate when either the safety limit or the normal limit is credited. The staff found that the applicant's response properly addresses the staff concerns and clarified the DCD statement that could cause confusion. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.3.3 Conclusion

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

In its previous evaluations of AP1000 DCD, Section 9.1.3, "Spent Fuel Pool Cooling System," the staff identified acceptance criteria based on the design meeting relevant requirements in 10 CFR Part 50, Appendix A, GDC 2; GDC 4; GDC 5; GDC 44, "Cooling Water"; GDC 45, "Inspection of Cooling Water Systems"; GDC 46, "Testing of Cooling Water Systems"; GDC 61; GDC 63; and in 10 CFR 20.1101(b), "Radiation protection programs," as it relates to radiation dose being kept as low as is reasonably achievable (ALARA). The staff found that the AP1000 SFS design was in compliance with these requirements, as referenced in NUREG-0800 Section 9.1.3 and determined that the design of the AP1000 SFS, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 SFS as documented in AP1000 DCD, Revision 17 against the relevant acceptance criteria as listed above and in NUREG-0800 Section 9.1.3. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 SFS to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD. The staff concludes that the AP1000 SFS design continues to meet all applicable acceptance criteria and the proposed changes are properly documented in the updated AP1000 DCD. The proposed changes meet the criteria of 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information in the AP1000 DCD. Therefore, the staff finds that the proposed changes to the AP1000 SFS are acceptable.

9.1.4 Light Load Handling System (Related to Refueling)

9.1.4.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.1.4, "Light Load Handling Systems" (LLHS) of the AP1000 DCD, Revision 15. In AP1000 DCD, Revision 17, the applicant has proposed to make changes to Section 9.1.4 of the DCD.

The applicant proposed 5 technical changes in the DCD supported by information presented in TR-106 and TR-121.

The staff reviewed the following changes to the AP1000 DCD, Section 9.1.4:

1. The applicant proposed to change the name, type and crane capacity of the new-fuel handling crane. In the previously approved DCD Revision 15, the new-fuel handling crane was called the fuel handling jib crane with a crane capacity of 907 kg (2000 lb). In DCD Revision 17, the applicant changed the name of the fuel handling jib crane to the new-fuel handling crane since the final crane specified may not be a jib crane. The new-fuel handling crane capacity was increased to lift a new fuel assembly, control rod assembly and handling tool [total weight of 919 kg (2,027 lb]. The basis for this change was documented in TR-106. In its June 26, 2008, response to RAI-SRP9.1.4-SBPB-01,

the applicant stated that the function of moving new fuel will be transferred to the FHM and that the new-fuel handling crane will be eliminated.

The applicant identified this change in AP1000 DCD, Tier 2, Sections 9.1.1.1, 9.1.1.2, 9.1.1.2.1, and 9.1.1.3.

2. The applicant proposed to change the safety and seismic classification of the FHM and SFHT. In the previously approved DCD Revision 15, the FHM and SFHT were classified safety Class C, seismic Category I. The applicant changed the classification for both items to AP1000 Class D nonsafety-related, seismic Category II. The basis for this change is documented in TR-106.

   The applicant identified this change in DCD Tier 2, Sections 3.2.2.5, 3.2.2.6, and Table 3.2-3.

3. The applicant proposed to change the description of the FHM. In the previously approved DCD Revision 15, the FHM was described as "the same design as the refueling machine and includes the same safety features." The applicant changed the description of the FHM to "the fuel handling machine has the same design functions as the refueling machine and includes the same safety features." The basis for this change is documented in TR-106. This section was later changed in response to RAI-SRP-9.1.4-SBPB-03 to provide of the FHM safety features.

   The applicant identified this change in DCD Tier 2, Sections 9.1.4.3.3.

4. The applicant proposed to change Tier 1, Table 2.1.1-1, "ITAAC Acceptance Criteria for Design Commitment 5." The revised acceptance criterion would state that "[t]he bottom of the dummy fuel assembly cannot be raised to within 24 ft, 6 in. [7.46 m] of the operating deck floor." This change is documented in TR-121. This height restriction was later revised as reviewed in Section 9.1.4.2.4 below.

5. The applicant indicated that due to the radius of the FHM manipulator mast and the proximity to the SFP walls, approximately 25 percent of the SFP storage cells cannot be serviced by the mast crane. Also, there are instances where fuel inspection and/or fuel repair require the fuel to be moved from the SFP storage racks to the designated fuel inspection or fuel repair workstation. These non-normal fuel transfer operations are performed using the SFHT. The SFHT is a long handled tool, which latches onto the fuel assembly top nozzle via manually actuated grippers. Lifting of the SFHT and attached fuel assembly is performed using a hoist on the FHM. The applicant later changed the FHM configuration to eliminate the need for the mast crane and replace it with two hoists to handle fuel with the SFHT and new-fuel handling tool. This is reviewed in Section 9.1.4.2.5 below.

6. The applicant proposed to eliminate from the AP1000 design the mounting of the rod cluster control storage station from the reactor cavity wall and, therefore, its intended use. In DCD Section 9.1.4, the applicant has changed the title of paragraph K under the section heading Component Description from "Rod Cluster Control Storage Station" to "Not used." In addition, the description of the rod cluster control storage station in paragraph K has been removed.

7. The applicant proposed to add the spent fuel assembly handling tool to the list of tools itemized under the section heading Fuel Handling Tools and Equipment. In the DCD Section 9.1.4, the applicant added to this section paragraph C with the title "Spent Fuel Assembly Handling Tool." The new paragraph describes some operational aspects of the tool and preoperational testing.

   The applicant identified this change in DCD Tier 2, Section 9.1.4.3.4.

8. In DCD Revision 17, the applicant proposed to change the refueling water and reactor coolant nominal boron concentration. The applicant completed the "Spent Fuel Storage Racks Criticality Analysis," for the new SFP racks and documented its results in TR-65. The applicant proposed the changes to boron concentration in DCD Section 9.1.4 to be consistent with the analyses presented in TR-65. This change is evaluated in Section 9.1.2.2.4 of this report.

9. In DCD Revision 17, Sections 9.1.4.2.3 and 9.1.4.3.7, the applicant changed the distance between the top of the active fuel to the surface of the spent fuel water. In Section 9.1.4.3.7, the applicant proposed to change the dose rate at the surface of the water during spent fuel transfer from 20 millirem/hour or less to an exposure rate for an operator to 2.5 millirem per hour or less. These changes are reviewed in Section 12.3 of this report.

9.1.4.2 Evaluation

The staff reviewed all changes to the LLHS in the AP1000 DCD Revision 17 in accordance with NUREG-0800 Section 9.1.4, "Light Load Handling System (Related to Refueling)." The regulatory basis for Section 9.1.4 of the AP1000 DCD is documented in NUREG-1793, which states that staff acceptance of the design, is contingent on compliance with the following requirements:

- GDC 2, as it relates to the ability of structures, systems, and components (SSCs) to withstand the effects of earthquakes

- GDC 5, as it relates to whether shared SSCs important to safety are capable of performing required safety functions

- GDC 61, as it relates to a radioactivity release resulting from fuel damage and the avoidance of excessive personnel radiation exposure

- GDC 62, as it relates to criticality prevention

The specific criteria that apply to the proposed DCD changes are; 10 CFR 52.63(a)(1)(vi), which concerns substantially increasing overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii) which concerns contribution to the increased standardization of the certification information in the AP1000 DCD.

9.1.4.2.1 Name Change and Crane Capacity Change for New-Fuel Handling Crane

The applicant proposed to change the name, type and crane capacity of the new-fuel handling crane. In Revision 17 to the DCD, the applicant deleted the classification of new-fuel handling crane as a jib crane. Based on TR-106, the applicant made this change because the final crane specified may not be a jib crane. Accordingly, the applicant changed the name of this crane to the new-fuel handling crane (from the fuel handling jib crane). The applicant also changed the capacity of the new-fuel handling crane from 907 kg (2000 lb) to 919 kg (2027 lb), which is the total combined weight of a new fuel assembly, control rod assembly and handling tool. Subsequently, the applicant stated in a letter dated June 26, 2008, that the function of moving new fuel will be transferred to the FHM and that the new-fuel handling crane will be eliminated. This issue is discussed below.

In accordance with NUREG-0800 Section 9.1.4, the LLHS needs to meet the requirements of GDC 2, GDC 5, GDC 61 and GDC 62. The NUREG-0800 Section 9.1.4 acceptance criteria for meeting the requirements of GDC 61 and GDC 62 are based on meeting the guidelines of ANSI/ANS 57.1-1992, "Design Requirements for Light Water Reactor Fuel Handling Systems." The staff finds that by changing the name of the new-fuel handling crane, removing its designation as a jib crane, and increasing its capacity, the applicant will continue to comply with the requirements of GDCs 5, 61 and 62.

The NUREG-0800 Section 9.1.4 acceptance criteria for meeting the requirements of GDC 2 are based on compliance with meeting the Regulatory Positions C.1 and C.2 of RG 1.29. The new-fuel handling crane, which handles new fuel and loads the new fuel into the SFP, was designed to be a seismic Category II component. Regulatory Position C.2 of RG 1.29 and Section 3.2.1.1.2 of DCD Revision 17, describe the guidance for seismic Category II SSCs. These state, in part, that seismic Category II SSC should be designed to preclude their structural failure during a safe shutdown earthquake (SSE) or interaction with seismic Category I items that could degrade the functioning of safety-related SSCs to an unacceptable level. In order to be seismic Category II, DCD Sections 9.1.1.2.1.D and 9.1.2.2.1.E state that the new-fuel handling crane will neither fall into the new fuel storage pit nor collapse into the SFP during a seismic event. Although the new-fuel handling crane will neither fall nor collapse during a seismic event as stated above, DCD Sections 9.1.1.2.1.D and 9.1.2.2.1.E did not state that the new fueling handling crane will continue to hold its maximum load (not drop the load) during the seismic event. Since a load drop could degrade the functioning of safety-related SSCs to an unacceptable level, the staff asked the applicant in RAI-SRP9.1.4-SBPB-01 to explain how this crane will meet seismic Category II criteria considering the maximum load carried by the crane.

The applicant responded in a letter dated June 26, 2008, indicating that the function of moving new fuel will be transferred to the FHM and that the new-fuel handling crane will be eliminated. The applicant stated that the FHM will be changed from a sigma style crane to a bridge style crane. The applicant stated that the FHM will remain a seismic Category II component and will not drop a fuel element under SSE conditions. The staff found the applicant's response satisfactory because the function of the new-fuel handling crane will be performed by the FHM and the FHM will continue to be a seismic Category II component in order to not drop its load during an SSE. Thus, the applicant will need to comply with GDC 2 when the new-fuel handling crane is eliminated and replaced by the FHM.

The applicant provided the staff with Revision 1 to its response to RAI-SRP9.1.4-SBPB-01 in a letter dated April 3, 2009. The applicant's revised response states that the words, "The fuel

handling machine is designed to maintain its load carrying and structural integrity functions during a safe shutdown earthquake," will be inserted into DCD Tier 2, Section 9.1.1.2.1.D, "New Fuel Rack Design," and DCD Tier 2, Section 9.1.2.2.1.E, "Spent Fuel Rack Design." In the response, the applicant provided markups of DCD Tier 2, Sections 9.1.1.2.1 D and 9.1.2.2.1 E, which showed how the additional text will be added in the next DCD revision. However, in Revision 1 to the RAI response the applicant also made the statement that these changes support the statement in DCD Tier 1, Section 2.1.1, "Fuel Handling and Refueling System," Item 6, "The RM (refueling machine) and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake." While the Revision 1 response addressed the FHM seismic capabilities, the RM seismic capabilities were not addressed even though the RAI response implied that they would be addressed.

In a letter dated September 4, 2009, the applicant provided Revision 2 to the response to RAI-SRP9.1.4-SBPB-01. The applicant's revised response stated that the words, "the refueling machine is designed to maintain its load carrying and structural integrity functions during a safe shutdown earthquake" will be included in Tier 2 of the DCD. The staff finds this acceptable because the DCD will address both RM and FHM ability to hold their load during an SSE, and the DCD will provide Tier 2 seismic information to support Tier 1 content. Therefore, RAI-SRP9.1.4-SBPB-01 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.4.2.2 Non Safety-related Classification for AP1000 Fuel Handling Equipment:

The applicant proposed to change the safety and seismic classification of the FHM and SFHT. As documented in TR-106, the applicant reclassified the FHM and the SFHT as nonsafety-related, seismic Category II. As stated in Section 9.1.4.1.1 of this report, NUREG-0800 Section 9.1.4 states that the LLHS needs to meet the requirements of GDC 2, GDC 5, GDC 61 and GDC 62. The staff determined that this change does not affect the LLHS meeting the requirements of GDC 5 because this change will not cause the sharing of equipment important to safety between nuclear power units.

The NUREG-0800 Section 9.1.4 acceptance criteria for meeting the requirements of GDC 61 and GDC 62 are based on meeting the guidelines of ANSI/ANS 57.1-1992. The staff finds that the nonsafety classification of the FHM and the SFHT is consistent with ANSI/ANS 57.1-1992. Therefore, the staff finds the reclassification of FHM and the SFHT as nonsafety-related to be acceptable.

The NUREG-0800 Section 9.1.4 acceptance criteria for meeting the requirements of GDC 2 are based on compliance with meeting the Regulatory Positions C.1 and C.2 of RG 1.29. Since these SSCs have been accepted by the staff as nonsafety-related, Regulatory Position C.2 of RG 1.29 is applicable in meeting the requirements of GDC 2. Regulatory Position C.2 states that seismic Category II SSCs need to be designed to preclude their failure, which could reduce the function of any seismic Category I SSC during an SSE to an unacceptable safety level.

The staff asked the applicant in RAI-SRP9.1.4-SBPB-02, to verify that the FHM and the SFHT will continue to hold their design load during an SSE. The applicant responded in a letter dated June 26, 2008, that the FHM will be designed to maintain its structural integrity and load carrying ability during an SSE. The staff found the applicant's response satisfactory because it complies with Regulatory Position C.2 of RG 1.29 and thus, the requirements of GDC 2 are met. Although the AP1000 DCD, Revision 16 stated that the FHM maintains its structural integrity

during an SSE, the DCD did not state that the FHM maintains its load carrying capability during an SSE.

The applicant provided the staff with Revision 1 to its response to RAI-SRP9.1.4-SBPB-02, dated April 3, 2009. The applicant's revised response referred to the changes made in its Revision 1 response to RAI-SRP9.1.4-SBPB-01. As stated above, while the Revision 1 response to RAI-SRP9.1.4-SBPB-01 addressed the FHM seismic capabilities, the RM seismic capabilities were not addressed as the RAI response implied. The applicant's September 4, 2009, revised response to RAI-SRP9.1.4-SBPB-01 included information related to RAI-SRP9.1.4-SBPB-02 addressing the RM seismic capability. The staff finds this response acceptable because it ensures the RM and FHM seismic capability to hold load during an SSE.

However, in DCD Revision 16, Table 3.2-3, the applicant designated dual seismic classification to the FHM. This was the only SSC in Table 3.2-3 that had a dual seismic classification of "II/NS," and no reason for the dual seismic classification was given by the applicant. Therefore, the staff requested in RAI-SRP9.1.4-SBPB-02 that the applicant provide justification for the dual seismic classification. The applicant responded in a letter dated June 26, 2008, that the FHM was a seismic Category II component and that Table 3.2-3 would be updated in Revision 17 of the DCD. The applicant provided a DCD markup page of Table 3.2-3. The staff has confirmed that Revision 17 of the DCD Tier 2, Table 3.2-3 was revised as committed in the RAI response. Furthermore, the staff finds that the issue concerning the seismic classification of the FHM has been adequately addressed and resolved by the applicant, because the FHM complies with Regulatory Position C.2 of RG 1.29 and thereby the requirements of GDC 2. Therefore, the staff's concern described in RAI-SRP9.1.4-SBPB-02 is resolved.

With the designation of the FHM and RM as seismic Category II, the staff finds that the LLHS will continue to meet the requirements of GDC 2 as it relates to the ability of SSCs to withstand the effects of earthquakes, GDC 61 as it relates to adequate safety from radioactivity resulting from fuel damage, and GDC 62 as it relates to prevention of criticality.

9.1.4.2.3 Fuel Handling Machine Generic Description

The applicant proposed to change the safety evaluation description of the FHM. In accordance with TR-106, the applicant revised the description of the FHM. In DCD Revision 16, Section 9.1.4.3.3, the applicant stated that the FHM "has the same design functions as the RM and includes the same safety features." However, DCD Revision 16, Sections 9.1.4.2.4 and 9.1.4.2.2.3 stated that the RM services the core—including the function to latch and unlatch control rods. No such function was attributed to the FHM. Additionally, DCD Revision 16, Section 9.1.4.2.3 stated that the FHM is used to load spent fuel into the shipping casks. No such function was attributed to the RM. Additionally, the RM operated exclusively in containment while the FHM operated exclusively in the fuel handling area. Therefore, the staff requested, in RAI-SRP9.1.4-SBPB-03, that the applicant explain how the FHM has the same design functions as the RM.

The applicant responded in a letter dated June 26, 2008, that the FHM design will be changed to a bridge/gantry style machine with two 2-ton overhead hoists. The applicant then described the safety interlocks, bridge hold-down devices, hoist braking system, and the fuel assembly support system for the FHM. The staff addresses the adequacy of the applicant's description of the FHM in Section 9.1.4.2.5 of this report.

9.1.4.2.4 Tier 1, Table 2.1.1-1, ITAAC Acceptance Criteria for Design Commitment 5

In the previously approved AP1000 DCD Revision 15, Tier 1, Table 2.1.1-1, ITAAC Acceptance Criteria for Design Commitment 5 stated that "[t]he bottom of the dummy fuel assembly cannot be raised to within 26 ft, 1 in. [7.9 m] of the operating deck floor." In DCD Revision 17, Tier 1, Table 2.1.1-1, the applicant proposed to change the lift height to 7.46 m (24 ft, 6 in). The bases for these changes are included in TR-121, and these changes are reviewed by the staff in Section 9.1.3 and Section 12.3 of this report. Some of the potential effects on DCD Section 9.1.4 are summarized in the applicant's response to RAI-SRP9.1.4-SBPB-04, Revision 1. The staff determined that these changes made to DCD Tier 2, Section 9.1.4 and Tier 1, Table 2.1.1-1 are conforming changes to the changes made in DCD Section 9.1.3 and do not impact the staff's safety evaluation of DCD Section 9.1.4. In addition, the staff's evaluation of the justification for the minimum shielding change is discussed in the applicant response to RAI-SRP12.3-CHPB-02 and is reviewed and documented in Section 12.3 of this report. The new lift height limit of 7.46 m (24 ft, 6 in) continues to demonstrate that the FHM will not raise a fuel assembly above the minimum required depth of water shielding.

9.1.4.2.5 Moving Spent Fuel With the SFHT and Auxiliary Hoist of the FHM

NUREG-0800 Section 9.1.4, "Light Load Handling System (Related to Refueling)," invokes GDC 61 for avoidance of excessive personnel radiation exposure. NUREG-0800 Section 9.1.4 acceptance criteria for meeting the relevant aspects of GDC 61 are based in part on the guidelines of ANSI/ANS 57.1-1992. Section 6.1.1 of this standard states, "Mechanical or electrical safety devices shall be designed into the system to prevent damage to fuel units and conditions that pose a radiation hazard or an unintentional radiation exposure risk to personnel." Section 6.4.1.2 of this standard recommended testing of these safety devices. In RAI-SRP9.1.4-SBPB-04, the staff asked the applicant to explain how they meet ANSI/ANS 57.1-1992 for the auxiliary hoist of the FHM, and the staff asked what ITAAC will test the functionality of the aforementioned safety devices.

The applicant responded to RAI-SRP9.1.4-SBPB-04 in a letter dated June 26, 2008, that the mast and auxiliary hoist was eliminated from the FHM and replaced with two overhead/trolley hoists. Since the applicant eliminated the mast and replaced it with an overhead hoist, the applicant needed to address the effects upon safe movement of fuel without the stability and position accuracy of a mast, in not causing mechanical damage to the new and spent fuel assemblies during movements of fuel assemblies. Movements of fuel within the refueling area include loading spent fuel into the spent fuel racks and moving spent and new fuel between the SFP and the fuel transfer canal.

The applicant provided the staff with Revision 1 to its response to RAI-SRP9.1.4-SBPB-04 in a letter dated May 20, 2009. The applicant's revised response stated that use of an overhead hoist design with an SFHT is common industry practice and has been accepted in previous designs. The applicant further stated: (1) the FHM is unable to raise the fuel to an unsafe radiological height in the SFP at any time during transport, storage, and installation activities; (2) features of the lifting devices' control circuitry and procedural operation prevent inadvertent unsafe movement of the fuel; (3) fuel transit speed, as operating limits or obstructions are approached, is automatically reduced to prevent overtravel or excessive sway of a fuel assembly. The staff finds this method of moving spent fuel with the SFHT acceptable since this type of design is consistent with common industry practice and is currently used in operating plants.

The applicant stated in a letter dated June 26, 2008, that both FHM hoists will be equipped with the safety devices identified in ANSI/ANS 57.1-1992, Section 6.3.1 with the exception of (9) Grapple Release as this feature is the manual operation of the fuel handling tool. A staff review of the applicant's proposed changes to the DCD, as described in its response to RAI-SRP9.1.4-SBPB-04, found that the down-position (interlock), end-travel (hardstop), up-limit (hardstop), translation inhibit (interlock) were not described. Furthermore, the applicant was not clear in its response whether the bridge travel (interlock) and trolley travel (interlock) are part of the design of the FHM. The applicant needed to clarify in the DCD that the FHM hoists have the safety devices identified in ANSI/ANS 57.1-1992, Section 6.3.1.

In its May 20, 2009 revised response to RAI-SRP9.1.4-SBPB-04, the applicant also proposed to include a description that clarifies the safety interlocks for the FHM in Section 9.1.4.3.3. Additional content was proposed in the RAI- SRP9.1.4-SBPB-03 response dated June 4, 2009 to incorporate a description of FHM bridge hold-down devices, hoist braking system, and the fuel assembly support system into Section 9.1.4.3.3. The applicant also stated that portions of the DCD "Light Load Handling System (Related to Refueling)" text will be clarified to indicate that the safety features listed include the safety requirements listed in ANSI/ANS 57.1. The clarifying text, which states that the safety features listed include the safety requirements listed in ANSI/ANS 57.1, will be added to DCD Sections 9.1.4.3.1, "Refueling Machine"; 9.1.4.3.2, "Fuel Transfer System"; and 9.1.4.3.3, "Fuel Handling Machine." In Section 9.1.4.3.3, there is one exception to ANSI/ANS 57.1 for the grapple release, as this feature is the manual operation of a fuel handling tool. In its response, the applicant provided markups of DCD Tier 2 Sections 9.1.4.3.1, 9.1.4.3.2 and 9.1.4.3.3 to be added in the next revision to the DCD. The staff finds this response acceptable since the safety requirements listed in ANSI/ANS 57.1 will be applied to the RM, fuel transfer system, and FHM; this meets GDC 61 and GDC 62. The staff's concern described in RAI-SRP9.1.4-SBPB-04 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

To address the part of the staff's RAI-SRP9.1.4-SBPB-04 request to identify the ITAAC that will test the functionality of the auxiliary hoist of the FHM, the applicant proposed, in Revision 17 of DCD Tier 1, changes to Section 2.1.1, "Fuel Handling and Refueling System," paragraphs 4, 5 and Table 2.1.1-1 (ITAAC) for associated Design Commitments 4 and 5. Based on its review, the staff finds that the changes to the above ITAAC are acceptable since they will adequately verify that the FHM/SFHT gripper assemblies are designed to prevent opening while the weight of the fuel assembly is suspended from the grippers. ITAAC Design Commitment 5 will also verify that the FHM hoists are limited such that the minimum required depth of water is maintained.

Single failure and non-single-failure proof hoists

In the June 26, 2008, response to RAI-SRP9.1.4-SBPB-04, the applicant stated that the FHM design will be changed to a bridge style machine with two 2-ton overhead hoists, one of which is single-failure proof. It was stated that a single-failure proof hoist and the new fuel handling tool will be used to handle new fuel and a nonsingle-failure proof hoist and the SFHT will be used to handle spent fuel. The applicant initially stated that the single-failure proof hoist may also handle spent fuel, but it would not have access to all spent fuel handling/storage locations. In a March 18, 2009, public meeting between the staff and the applicant, the use of the FHM single-failure proof hoist and nonsingle-failure proof hoist was discussed in detail. The applicant stated that the new FHM will handle new fuel and spent fuel.

In the June 26, 2008 response to RAI-SRP9.1.4-SBPB-03, the applicant also stated, "The fuel handling machine is restricted to raising a fuel assembly to a height at which the water provides a safe radiation shield," and in response to RAI-SRP9.1.4-SBPB-04, the applicant stated that "each FHM hoist will have a mechanical limit based on maximum hoist up travel and spent fuel handling tool length." Since the new FHM will be moving both new fuel and spent fuel, and new fuel is handled above deck level when it is transferred to the new fuel racks and transferred from the new fuel storage vault into the SFP, the applicant did not state in the DCD how the same cranes that are restricted in hoist uptravel can handle new fuel above deck level. Use of the FHM hoist for new fuel also apparently conflicted with the revised Tier 1 ITAAC Table 2.1.1-1 Item 5, which stated, "FHM hoists are limited such that the minimum required depth of water shielding is maintained."

The applicant provided Revision 1 to its response to RAI-SRP9.1.4-SBPB-04 in a letter dated May 20, 2009 and Revision 1 to its response to RAI-SRP9.1.4-SBPB-03 in a letter dated June 4, 2009. Both of the applicant's revised RAI responses include the same additional paragraph, which defined some restrictions to the use of the nonsingle-failure proof hoist and the single-failure proof hoist of the FHM for handling new fuel and other loads throughout the fuel handling area. The single-failure proof hoist in conjunction with the SFHT is not capable of raising spent fuel to a height that clears the spent fuel racks, fuel transfer system fuel basket, spent fuel shipping cask, or the new fuel elevator. The staff found that the applicant's Revision 1 responses to RAI-SRP9.1.4-SBPB-03 and RAI-SRP9.1.4-SBPB-04 did not adequately address how the single-failure proof crane of the FHM with hoist uptravel restrictions can handle new fuel above the deck level.

The applicant's Revision 2 to its RAI-SRP9.1.4-SBPB-03 response was issued in a letter dated October 15, 2009. The letter provides a clarifying description of the fuel movement process using the FHM. The design for the movement of spent fuel utilizes the nonsingle-failure proof hoist and its associated SFHT. The new fuel handling tool is used with the single-failure proof hoist to move new fuel. The SFHT and the new fuel handling tool are manually operated tools and differ in length by approximately 9.14 m (30 ft). The single-failure proof hoist does not have the lift height to raise a spent fuel assembly clear of the spent fuel racks, fuel transfer system fuel basket, spent fuel shipping cask, or the new fuel elevator when using the SFHT. When spent fuel is stored in the spent fuel racks, or other interim storage locations, spent fuel movement with either hoist would be physically impossible using the new fuel handling tool, as the operating handle of the tool would be submerged in approximately 6.09 m (20 ft) of water.

Therefore, a description of the fuel movement (new and spent) process for both FHM hoists using their handling tools, and a discussion of their interlocks, needed to be included in the DCD. This was identified as Open Item OI-SRP9.1.4-SBPA-03.

To address Open Item OI-SRP9.1.4-SBPA-03, the applicant provided a response dated March 31, 2010. The response proposed to revise DCD Section 9.1.4.2.4 to incorporate the statement provided in RAI-SRP9.1.4-SBPB-03 and RAI-SRP9.1.4-SBPB-04 regarding the restrictions for use of single and nonsingle-failure proof hoists in the DCD. However, when further requested to clearly define the usage of the single and nonsingle-failure proof hoists, the applicant submitted a revised response to Open Item OI-SRP9.1.4-SBPA-03 dated May 19, 2010. This revised response proposed to revise DCD Section 9.1.4.2.4 to define the use of hoists above deck. This response also described the availability of a selector switch to choose the correct hoist and integrated controls to avoid inadvertent use of the incorrect hoist. The response also defined the intended use of each hoist as follows:

The single-failure proof hoist will be used for;

- primarily handling new fuel
- the movement of loads <1814 kg (4000 lb) in the fuel handling area of the auxiliary building
- a redundant hoist over the SFP for the handling of control components

The nonsingle-failure proof crane will be used for;

- handling fuel and control components in the SFP
- the hoist shall be restricted from handling a load above the operating floor within 4.6 m (15 ft) of the SFP

The proposed additions to Section 9.1.4.2.4 clarify that the nonsingle-failure proof hoist is primarily used for submerged handling activities in the SFP. There are areas in the fuel handling area of the auxiliary building that the single-failure proof hoist is not capable of accessing due to travel limitations. Therefore, the response stated that it is necessary for the nonsingle-failure proof hoist to be used in areas other than the SFP. As a safety precaution, the applicant indicated that the nonsingle-failure proof hoist will be restricted from handling a load above the operating floor within 4.6 m (15 ft) of the SFP. The applicant further stated that the single-failure proof hoist will be capable of handling loads in the new fuel handling area and the spent fuel handling area with operator warnings associated with the handling of spent fuel and proposed corresponding changes to DCD.

Although the proposed DCD changes addressed restriction on the nonsingle-failure proof hoist travelling over SFP, it was unclear what provisions are provided to prevent the single-failure proof hoist from handling new fuel over SFP with the new fuel handling tool. For additional clarification of new fuel movement, the applicant provided a revised response to Open Item OI-SRP9.1.4-SBPA-03 dated July 9, 2010, which further defined the restrictions of new fuel movement above deck while using the new fuel handling tool. The open item response proposed a revision to the DCD to indicate that the nonsingle-failure proof hoist is restricted from handling new fuel above the operating floor. The applicant also clarified that the new fuel elevator fuel carrier is located in the tool storage area of the SFP. The open item response revised DCD Figure 9.1-4 to indicate the location of the new fuel elevator and defined the safety interlocks to prevent the transporting of new fuel above the operating floor over the spent fuel racks by the single-failure proof hoist in Section 9.1.4.2.4.

Based on the above, the staff finds that there is reasonable assurance that the use of the single and nonsingle-failure proof hoist configuration on the FHM will minimize the potential for damage to fuel, fuel assemblies, and to storage or transport containers. The restrictions on fuel movement and safety interlocks provide safe movement of new and spent fuel in the auxiliary building using these hoists. Therefore, Open Item OI-SRP9.1.4-SBPA-03 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

To address the section part of the staff's RAI-SRP9.1.4-SBPB-04 to identify the ITAAC that will test the functionality of the auxiliary hoist of the FHM, the applicant proposed, in Revision 17, Tier 1, changes to DCD Tier 1, Section 2.1.1, "Fuel Handling and Refueling System"

paragraphs 4 and 5 and DCD Tier 1 Table 2.1.1-1 (ITAAC) for associated Design Commitments 4 and 5. Based on its review, the staff finds that the changes to the above ITAAC are acceptable.

Table 9.1-1 in Section 9.1.4.2.8 of this report provides a summary of the fuel handling operations of the FHM discussed above.

9.1.4.2.6 Elimination of Rod Cluster Control Storage Station from Reactor Cavity Wall

In DCD Tier 2, Revision 17, the applicant proposed to eliminate from the AP1000 design the mounting of the rod cluster control storage station from the reactor cavity wall and, therefore, its intended use. The applicant proposed that it will instead perform rod cluster control assembly inspections in the auxiliary building. As stated in Section 9.1.4.1.1 of this report, NUREG-0800 Section 9.1.4 states that the LLHS needs to meet the requirements of GDC 2, GDC 5, GDC 61 and GDC 62. The staff determined that this change of eliminating the rod cluster control storage station from the reactor cavity wall will have no effect on the remaining LLHS components from meeting the applicable GDC.

9.1.4.2.7 Addition of Spent Fuel Assembly Handling Tool under Fuel Handling Tools and Equipment Section

In DCD Tier 2, Revision 17, the applicant proposed to add the spent fuel assembly handling tool to the list of tools itemized under the Section 9.1.4.3.4. This change is associated with the comprehensive changes made to Section 9.1.4 by the applicant due to the elimination of the new-fuel handling crane as discussed in the applicant's responses to RAI-SRP9.1.4-SBPB-01 through RAI-SRP9.1.4-SBPB-04. The staff determined that the addition of the spent fuel assembly handling tool to the list of tools itemized under the section heading fuel handling tools and equipment is acceptable, because it represents additional changes needed to Section 9.1.4, which were not previously presented in the responses to RAI-SRP9.1.4-SBPB-01 through RAI-SRP9.1.4-SBPB-04.

9.1.4.2.8 Fuel Handling Summary Table

Table 9.1-14. Fuel Handling Machine (FHM) Hoist Operations

Fuel Handling Scenario	Fuel Handling Machine (FHM) Hoist Operations			
	Non-Single-Failure-Proof Hoist		Single-Failure-Proof Hoist	
	New Fuel Assembly Handling Tool	Spent Fuel Assembly Handling Tool	New Fuel Assembly Handling Tool	Spent Fuel Assembly Handling Tool
Moves new fuel (NF) above the fuel handling area (FHA) operating floor/deck over spent fuel (SF) racks.	No	No	No	No
Moves NF above the FHA operating floor/deck.	No	No	**Yes**	No

Table 9.1-14. Fuel Handling Machine (FHM) Hoist Operations

Fuel Handling Scenario	Fuel Handling Machine (FHM) Hoist Operations			
	Non-Single-Failure-Proof Hoist		Single-Failure-Proof Hoist	
	New Fuel Assembly Handling Tool	Spent Fuel Assembly Handling Tool	New Fuel Assembly Handling Tool	Spent Fuel Assembly Handling Tool
Moves NF over the NF racks. (Same as Scenario 2)	No	No	**Yes**	No
Moves NF over the NF Elevator (elevator up).	No	No	**Yes**	No
Moves NF over the NF Elevator (elevator down).	No	**Yes**	No	No
Moves NF within the SFP over the SF racks.	No	**Yes**	No	No
Moves SF within the SFP over the SF racks.	No	**Yes**	No	No

9.1.4.3 Conclusion

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

The staff reviewed the applicant's proposed changes to the AP1000 LLHS as documented in DCD, Revision 17 against the relevant acceptance criteria as listed above and in NUREG-0800 Section 9.1.4. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 LLHS to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. The staff concludes that the AP1000 LLHS design continues to meet all applicable acceptance criteria and the proposed changes are properly documented in the updated AP1000 DCD. The proposed changes meet the criteria of: 10 CFR 52.63(a)(1)(vi), on the basis that they substantially increase overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information in the AP1000 DCD. Therefore, the staff finds that the proposed changes to the AP1000 LLHS are acceptable.

9.1.5 Overhead Heavy Load Handling Systems

9.1.5.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.1.5, "Overhead Heavy Load Handling Systems" (OHLHS) of the AP1000 DCD, Revision 15. In AP1000 DCD, Revision 17, the applicant has proposed to make changes to Section 9.1.5 of the DCD.

The applicant proposed changes in the DCD, which are supported by information presented in TR-106.

The staff reviewed the proposed changes to Section 9.1.5 of the AP1000 DCD, Revision 15.

1. The applicant proposed to rename the cask handling crane, upgrade its seismic classification, and make the cask handling crane single-failure proof. In the previously approved AP1000 DCD Revision 15, the applicant named this crane the spent fuel shipping cask crane. In DCD Revision 17, the applicant has changed the name of this crane to the cask handling crane, and changed the design basis to single-failure proof and seismic Category I.

   The applicant has also proposed to classify the maintenance hatch hoist as single-failure proof. In the previously approved AP1000 DCD Revision 15, this hoist was seismic Category I, but not single-failure proof. In DCD Revision 17, the applicant has added single-failure proof design criteria for the maintenance hatch hoist.

   In DCD Revision 17, the applicant also added two new safety-related functions for the mechanical handling system (MHS), those being the prevention of the uncontrolled lowering of a heavy load by both the cask handling crane and the maintenance hatch hoist. The applicant also added the cask handling crane and the maintenance hatch hoist to Tables 2.3.5-1 and 2.3.5-3 in Tier 1 to add these components to the tables that list seismic Category I equipment and component locations for the MHS. The applicant also added the cask handling crane and the maintenance hatch hoist to ITAAC in Table 2.3.5-2 of Tier 1.

   The bases for these changes are documented in the applicant's TR-106.

   The applicant identified these changes in AP1000 DCD Revision 17, Tier 1, Section 2.3.5, "Mechanical Handling System," and Tables 2.3.5-1, 2.3.5-2, and 2.3.5-3; Tier 2, Sections 9.1.5.1.1, "Safety Design Basis," 9.1.5.2, "System Description," and 9.1.5.3, "Safety Evaluation," Table 9.1-5, "Nuclear Island Heavy Load Handling Systems," and Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment."

2. The applicant proposed to modify the codes and standards applicable to the polar crane, cask handling crane and other overhead cranes and hoists. In the previously approved AP1000 DCD Revision 15, overhead cranes were designed to ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)." Other cranes and hoists that handle heavy loads were designed according to the applicable ANSI standard. AP1000 cranes were designed to ASME NOG-1 and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The spent fuel shipping cask crane was designed to ASME NOG-1 for a Type III crane and

ANSI/ANS-57.1 and ANSI/ANS-57.2, "Design Requirements Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants." In DCD Revision 17, the applicant has changed the polar crane and cask handling crane to be designed according to NUREG-0554 supplemented by ASME NOG-1 for a Type I single-failure proof crane. Other cranes and hoists which handle heavy loads will be designed according to ASME NOG-1 and to the applicable ANSI standard. In addition, the design of AP1000 cranes will comply with the guidance in NUREG-0612. The bases for all these changes are documented in TR-106.

The applicant identified these changes in DCD Revision 17, Tier 2, Sections 9.1.5.1.2, "Codes and Standards," and 9.1.5.2.1.2, "Component Descriptions," and Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment."

3. The applicant proposed to change one method by which the movements of the bridge, trolley, main, and auxiliary hoists of the polar crane can be controlled. In DCD Revision 15, the polar crane was controlled from either the operator's cab or from a pendant suspended from the crane. In DCD Revision 17, the applicant has changed control from a pendant to remote control. The remote control of the crane will have the same control functions, push button functions, and interlocks as the previous pendant design. The basis for this change is documented in TR-106.

   The applicant has identified this change in DCD Revision 17, Tier 2, Section 9.1.5.2.1.1, "System Operation."

4. The applicant proposed a new description for the cask handling crane. In the previously approved DCD Revision 15, the applicant did not describe the cask handling crane, formerly called the spent fuel shipping cask crane. In DCD Revision 17, with the upgrade of the crane to single-failure proof and seismic Category I, the applicant added the following sections and titles to describe the cask handling crane and its operation:

   9.1.5.2.2 Cask Handling Crane General Description
   9.1.5.2.2.1 System Operation
   9.1.5.2.2.2 Component Descriptions.
   9.1.5.2.2.3 Instrumentation Applications

   The bases for these changes are documented in TR-106.

5. The applicant proposed to change the maximum load rating of the containment polar crane. In the previously approved DCD Revision 15, the maximum load rating of the containment polar crane was 249.48 metric tons (275 tons). In DCD Revision 17, the applicant proposed to change the maximum load rating for the containment polar crane to 272.16 metric tons (300 tons). The basis for this change is documented in TR-106.

   The applicant identified this change in DCD Revision 17 in Table 9.1-5, "Nuclear Island Heavy Load Handling Systems."

6. The applicant proposed to change the tabular information in Table 9.1.5-1 for the cask handling crane (previously spent fuel shipping cask crane in DCD Revision 15). In the previously approved DCD Revision 15, the table was for a nonsingle-failure proof spent fuel cask crane and listed exceptions and clarifications to ANSI/ANS 57.1 and ANSI/ANS 57.2. In DCD Revision 16, the applicant removes the exceptions table and

replaces it with design information for the bridge, trolley, main hoist and auxiliary hoist of the cask handling crane.

The applicant identified this change in DCD Revision 17 in Table 9.1.5-1, "Cask Handling Crane Component Data."

7. The applicant proposed to specify the main hook of the polar crane to handle the RCP pump/motor shell. In the previously approved DCD Revision 15, the auxiliary hook of the polar crane was used to handle the RCP pump/motor shell. In DCD Revision 17, the applicant proposed to change Table 9.1.5-2, "Special Lifting Devices Used for the Handling of Critical Loads," to designate the main hook of the polar crane to be used with the RCP special lifting device to lift the RCP pump/motor shell. The basis for this change is documented in TR-106.

   The applicant identified this change in Table 9.1.5-2, "Special Lifting Devices Used for the Handling of Critical Loads."

8. The applicant proposed to change the approximate capacity of the polar crane auxiliary hoist from 68.04 metric tons (75 tons) to 22.68 metric tons (25 tons). In the previously approved DCD Revision 15 in Table 9.1.5-3, "Polar Crane Component Data," 68.04 metric tons (75 tons) was listed as the approximate capacity of the auxiliary hoist of the polar crane. In DCD Revision 17, the applicant proposed to change the approximate capacity of the polar crane auxiliary hoist to 22.68 metric tons (25 tons). The basis for this change is documented in TR-106.

   The applicant identified this change in Table 9.1.5-3, "Polar Crane Component Data."

9.1.5.2 Evaluation

The staff reviewed all changes to the OHLHS in the AP1000 DCD Revision 17 in accordance with NUREG-0800 Section 9.1.5, "Overhead Heavy Load Handling Systems." The staff did not re-review descriptions and evaluations of Section 9.1.5 of DCD Revision 15 that were previously approved and are not affected by the new changes. The regulatory basis for Section 9.1.5 of the DCD is documented in NUREG-1793.

The specific criteria that applies to the proposed DCD changes are; 10 CFR 52.63(a)(1)(vi), which concerns substantially increasing overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

9.1.5.2.1 Cask Handling Crane and Containment Maintenance Hatch Hoist Design Basis Changes (Single-failure Proof and Seismic Category I)

TR-106 states that the AP1000 cask handling crane design was neither seismically qualified, nor single-failure proof to protect against dropping a cask. Since a cask drop could cause significant plant damage, a design change was initiated to upgrade the cask handling crane to single-failure proof and seismic Category I. This change was made to reduce the possibility of a serious plant event. With this design change the crane name was changed from the spent fuel shipping cask crane to the cask handling crane. The cask handling crane will thus be designed

to support a critical load during and after a safe shutdown earthquake and is classified as seismic Category I.

TR-106 further states that the maintenance hatch hoist [9.07 metric tons (10 ton capacity)] is not specified in the DCD as single-failure proof. For personnel and equipment safety reasons, the maintenance hatch hoist design has also been changed to be single-failure proof. The equipment hatch hoist which was already designated a single-failure proof crane had its name changed to containment equipment hatch hoist to be specific in its plant location.

On the basis that upgrading of the cask handling crane and maintenance hatch hoist to single-failure proof and upgrading the cask handling crane to seismic Category I will result in greater safety during the handling of critical loads within the nuclear island, the staff finds the design changes acceptable.

However, the applicant, in DCD Tier 1 Section 2.3.5, "Mechanical Handling System," did not list single-failure proof as certified design information with ITAAC for the polar crane, the cask handling crane, the containment equipment hatch hoist or the containment maintenance hatch hoist. One design criterion for Tier 1 information is that it should include features and functions that could have a significant effect on the safety of a nuclear plant or are important in preventing or mitigating severe accidents. A drop of the reactor vessel head or a spent fuel cask could affect plant safety. Without heavy load drop analyses that proves the safety of a load drop, single-failure proof design criteria for the associated crane/hoist is necessary for plant safety and should be part of the Tier 1 safety significant design criteria for the polar crane and the cask handling crane. Therefore, the staff requested the applicant in RAI-SRP9.1.5-SBPB-09 to explain why they did not include single-failure proof design criteria and ITAAC in Tier 1 of the DCD for the polar crane, the cask handling crane, the containment equipment hatch hoist and the containment maintenance hatch hoist.

In its response dated September 3, 2008, the applicant stated that the single-failure proof criteria within ITAAC Table 2.3.5-2 for components 3a (polar crane), 3b (cask handling crane), 3c (equipment hatch hoist), and 3d (maintenance hatch hoist) will be incorporated.

The applicant also provided proposed revisions to DCD Revision 16, Tier 1, Section 2.3.5, Paragraph 3 under the heading "Design Description," to state the containment polar crane, cask handling crane, equipment hatch hoist and maintenance hatch hoist are single-failure proof. Also, in Tier 1 Table 2.3.5-2, the applicant proposed changes to ITAAC for the polar crane, the cask handling crane, the equipment hatch hoist and the maintenance hatch hoist.

Based on its evaluation, the staff finds that the applicant demonstrated in the proposed revisions that single-failure proof design criteria and ITAAC are identified in Tier 1 of the DCD for the polar crane, cask handling crane, containment equipment hatch hoist and the containment maintenance hatch hoist. The staff finds the applicant's response acceptable in that the applicant has now included single-failure proof design criteria and ITAAC in Tier 1 of the DCD for the four load handling systems. The staff confirmed that the applicant made the changes in DCD Tier 1 Revision 17, as described above.

However, the staff finds that the acceptance criteria for the proposed ITAAC should include, not only a report that concludes the acceptability of the proposed inspections, tests, and analyses, but also a certificate of conformance from the vendor stating that the crane/hoist is single-failure proof.

To address the staff's concern regarding the certification of the cranes and hoists as single-failure proof, the applicant provided the staff with Revision 1 to its response to RAI-SRP9.1.5-SBPB-09 in a letter dated June 4, 2009. In the revised response, the applicant stated that single-failure proof criteria will be updated in DCD Tier 1 ITAAC Table 2.3.5-2 for Design Commitments 3a, 3b, 3c and 3d, which are for the two cranes and two hoists. The proposed acceptance criteria for the ITAAC to be incorporated into the next DCD revision will include a certificate of conformance from the crane or hoist vendor stating that the crane or hoist is single-failure proof. The staff finds the response acceptable since the requirement for the crane or hoist vendor to provide a certificate of conformance for being single-failure proof will be added to the acceptance criteria for the crane or hoist ITAAC.

The staff's concern described in RAI-SRP9.1.5-SBPB-09 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.5.2.2 Codes and Standards for Design of OHLHS Cranes and Hoists

This design change corrects wording to state that the polar crane and cask handling crane are designed in accordance with the guidelines of NUREG-0554 supplemented by ASME NOG-1 for a Type I, single-failure proof crane. This change made the DCD consistent with NUREG-0800 Section 9.1.5. Therefore, the staff finds this design change acceptable.

However, for the single-failure proof equipment hatch hoist and maintenance hatch hoist, the applicant only specified ASME NOG-1 and the applicable ANSI standard. ASME NOG-1 can specify non single-failure proof components in the Type II and Type III designs. DCD Revision 16 also stated under Section 9.1.5.2, that the containment equipment hatch hoist and maintenance hatch hoist incorporate single-failure proof features based on NUREG-0612 guidelines. Unlike the polar crane and cask handling crane, there were no detailed descriptions and no specific sections of ASME NOG-1 specified for the containment equipment hatch hoist and maintenance hatch hoist in DCD Section 9.1.5 to make them single-failure proof. Therefore, the staff found the classification for the single-failure proof hoists did not state all necessary design specifications for a single-failure proof component and requested the applicant in RAI-SRP9.1.5-SBPB-01 to provide more design specifications for single-failure proof hoists and a description of the hoists in the DCD.

In its response dated June 26, 2008, the applicant stated that the design specification for the maintenance hatch hoist system and equipment hatch hoist system would follow the guidelines of NUREG-0554, supplemented by ASME NOG-1. The applicant further stated that Table 3.2-3 of the DCD would be revised to reflect these guidelines.

In DCD Revision 17, Tier 2, Table 3.2-3 (Sheet 8 of 65), the applicant revised the table to show the principal construction code for the maintenance hatch hoist and equipment hatch hoist as NUREG-0554, supplemented by ASME NOG-1. However, although the applicant revised DCD Table 3.2-3 to reflect the principal construction code for the two hoists as NUREG-0554, supplemented by ASME NOG-1, the staff noted that the text in DCD Sections 9.1.5.1.2 and 9.1.5.2 has not been changed to reflect the change in the two hoists' construction code, which are NUREG-0554, supplemented by ASME NOG-1 <u>for a Type 1 single-failure proof hoist</u>.

To address the staff's concern, the applicant provided Revision 1 to its response to RAI-SRP9.1.5-SBPB-01 in a letter dated April 13, 2009. The applicant stated that ASME NOG-1, Type 1 designation is not applicable for equipment hatch hoists and maintenance hatch

hoists, as it applies to the design of overhead and gantry cranes from the rails to the load hook. The applicant further proposed that the single-failure proof hatch hoists are designed, fabricated, examined and tested in accordance with cmAA 70 and the guidelines of NUREG-0554, supplemented by provision of ASME NOG-1 as it relates to single-failure proof hoists. The staff found the applicant's Revision 1 response to RAI-SRP9.1.5-SBPB-01 unacceptable. In the response, the applicant stated that the requirements of ASME NOG-1, Type 1 do not apply to equipment hatch hosts and maintenance hatch hoists, but the applicant proposed to apply portions of the NOG-1 requirements anyway. Since the applicant proposed to apply only portions of the applicable single-failure proof criteria to the hatch hoists, the staff asked the applicant to clearly define which portions of NRC guidance (e.g., NUREG-0554 and NUREG-0612) and ASME NOG-1 are applied to the heavy load handling cranes in order to classify them as single-failure proof cranes. The staff also asked the applicant to clarify the ambiguous use of the term "NUREG-0554 supplemented by ASME NOG-1," for all of the single-failure proof cranes. It is unclear whether all of ASME NOG-1, Type I, is applied to the polar and cask handling cranes. Revision 15 of the DCD clearly defined the polar crane as designed to ASME NOG-1 for a Type I, single-failure proof crane. However, DCD Revision 17 indicates "NUREG-0554 supplemented by ASME NOG-1." In RAI-SRP9.1.5-SBPB-01, the applicant was asked to provide justification and clearly define how the proposed polar crane, cask handling crane and hatch hoists designs would satisfy the single-failure proof criteria from the NRC guidance and industry standards with references to the specific paragraphs. In RAI-SRP9.1.5-SBPB-01, the applicant was also requested to provide an evaluation of the selected hoist standard to applicable portions of NUREG-0554 to avoid consideration of potential load drops by classification of the hoist as single-failure proof. This issue was tracked as Open Item OI-SRP9.1.5-SBPB-01 in the SER with open items. To address the staff's concern, the applicant provided a Revision 2 to its response to RAI-SRP9.1.5-SBPB-01 in a letter dated November 11, 2009. The applicant's response provided a detailed description of the hatch hoists to include in DCD Sections 9.1.5.2.3 and 9.1.5.2.4. These sections provide specific design criteria of the foot-mounted equipment hatch hoist, with content similar to what is specified for the polar crane and cask handling cranes. A revision to Section 9.1.5.2.3.2, "Component Descriptions," was proposed to incorporate the hoists description and elaborate on how the code requirements are implemented in the design of key safety-related components. The staff found finds the incorporation of the hatch hoist information in the DCD to be acceptable.

However, based on the applicant's clarification that the hatch hoists are foot-mounted on a platform supported by the containment structure, the applicant needed to justify how the structural loads on containment from the hoist are evaluated. For the proposed single-failure proof hatch hoists, there was no assurance that the evaluation considered the lifted load in conjunction with the seismic accelerations because the previous non-single-failure proof hoist was not required to hold a load during and following a seismic event. This became Open Item OI-SRP9.1.5-SBPB-01 in the SER with open items.

The applicant provided a response to Open Item OI-SRP9.1.5-SBPB-01 on March 31, 2010 describing the methodology for its seismic evaluation of the equipment hatch hoist. The response defined the acceptance criteria as "after a seismic event occurs while the hoist is holding the critical load, the containment vessel will continue to perform its intended safety functions." On May 19, 2010, the applicant provided an additional response to Open Item OI-SRP9.1.5-SBPB-01 indicating that the final loads were expected to be bounded by the current CV Design Specification. As a follow-up, the applicant provided an additional Open Item OI-SRP9.1.5-SBPB-01 response dated June 30, 2010, which indicated that the calculation

number APP-MH40-S2C-002 includes the hatch hoist platform structural analysis and that the applicant has included the hatch hoist configuration in its structural analysis.

On August 16, 2010, the staff conducted a regulatory audit review of the AP1000, "Hatch Hoist Platform Structural Qualification," (APP-MH40-S2C-002). In this design analysis, the applicant constructed a 3D finite element model for the platform structural analysis using a commercially available general purpose code (ANSYS). The platform model consists of a 3D beam (including rigid beams), plate/shell elements and lumped masses to represent both the platform and the hatch hoist system. The methodology, input/output and boundary conditions used were reviewed and found acceptable. Three loading cases were considered: (1) dead weight (1 g); (2) dead weight plus seismic; and (3) dead weight minus seismic. The maximum stresses for each structural component are obtained from the equivalent static analysis and used for design purposes. It was shown that use of 8x8x 5/16 beam and 10x10x5/16 beam with an E70xx fillet weld of 0.635 cm (0.25 in) thickness at the connection to the reactor vessel wall is in compliance with ANSI/American Institute of Steel Construction (AISC) N690-1994, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities (SC-I)," the calculation is acceptable as it provides assurance of an adequate design to support the heavy hatch hoist system under design-basis seismic loading conditions.

Based on the above review, the staff considers Open Item OI-SRP9.1.5-SBPB-01 resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.5.2.3 Remote Control Operation Change for Polar Crane and Cask Handling Crane

In TR-106 Revision 1, the applicant states that allowing the cask handling crane to be operated by radio remote control instead of the operator's cab will allow for an unobstructed view of the load at all times. Special consideration was given to loads being lifted and lowered out of and into the truck/rail bay. The cask handling crane radio remote will meet ASME NOG-1 paragraph 6110 guidelines. TR-106 Revision 1 also states that wording for the polar crane operation is changed from "pendant controls" to "remote control" as a secondary means of control. This would ensure consistency in design and operations of the two single-failure proof cranes (polar and cask handling). The staff reviewed TR-106 Revision 1 specifically for the addition of the radio remote control for the cask handling crane and remote control for the polar crane, and found the additions acceptable. The design change is in compliance with GDCs 4, 5, 13, and 24.

However, the licensing basis for the remote control features of the polar crane and the cask handling crane needs to be established in the DCD along with an appropriate ITAAC to verify that the plant meets the licensing basis. The DCD should include information indicating that: a) the remote control features of the polar crane and cask handling crane will not interfere with any SSC important to safety in accordance with GDC 4; b) the remote control systems will not be shared with multiple unit sites or interfere with other units at the site in accordance with GDC 5; c) remote control systems will maintain variables and systems within prescribed operating ranges in accordance with GDC 13; and d) remote control systems will be separate from protection systems, such that failure of the control system leaves the protection systems intact satisfying all reliability, redundancy, and independence requirements in accordance with GDC 24.

In RAI-SRP9.1.5-SBPB-12, the staff requested that the applicant specify the licensing basis for the remote control features of the polar crane and the cask handling crane in the DCD and establish ITAAC to verify the plant meets the licensing basis.

In its response to RAI-SRP9.1.5-SBPB-12, dated January 12, 2009, the applicant addressed each of the licensing basis questions by stating: a) the remote control system will comply with Section 8, Electromagnetic Compatibility (EMC Qualifications) of the Equipment Methodology (EQ) plan, APP-GW-G1-002, Revision 1. The transmitter power of the remote control system for the two cranes will be set to a level that will allow continuous communications with the crane receivers throughout their areas of operation. The signal strength will be adjusted to minimize signal propagation to areas outside of the two cranes normal operating areas; b and c) the two cranes' remote control systems will be designed so that they are integrated into the operating site's frequency plan; and d) the remote control systems will be designed to fail into a safe mode of operation in the event of a loss of communications or failure of the portable remote unit. Should a system failure occur, all crane movements are halted.

The staff finds the applicant's response acceptable since the addition of the remote control systems to the polar and cask handling cranes does not impact the conclusions made in this report about the heavy load handling systems. The design and operation of the actual crane portions of the polar and cask handling cranes does not change with the use of radio remote controls as the controls are the same as the pendant (cask handling crane) and cab controls (polar crane). Therefore, the applicant does not need to provide additional licensing basis and ITAAC verification.

The applicant updated DCD Sections 9.1.5.2.1 and 9.1.5.2.2.1 in Revision 16 to include the use of radio remote control for the polar and cask handling cranes. No further DCD updates are necessary for remote control design changes/additions, and the staff's concern described in RAI-SRP9.1.5-SBPB-12 is resolved.

9.1.5.2.4 Upgrade of Cask Handling Crane to Single-failure Proof and Detailed Description Addition

TR-106 Revision 1 states the AP1000 cask handling crane design was neither seismically qualified nor single-failure proof to protect against dropping a cask. Since a cask drop could cause significant plant damage, a design change was initiated to upgrade the cask handling crane to single-failure proof. This change was made to reduce the possibility of a serious plant event.

With this design change, the applicant revised the name of the crane from the spent fuel shipping cask crane to the cask handling crane and upgraded it to single-failure proof. TR-106 Revision 1 provided a detailed description of the new single-failure proof cask handling crane which was added to DCD Revision 17. Because both the polar crane and cask handling crane are single-failure proof, the added detailed description of the cask handling crane is very similar to the polar crane detailed description. Because the design and system operation of the cask handling crane is described almost identically to that of the polar crane, which was found to be in compliance with GDC 2, 4, 5 and 61, the staff finds this design change of adding a detailed description of the cask handling crane to the AP1000 OHLHS acceptable.

NUREG-0800 Section 9.1.5 and NUREG-0612 provide guidance that states that safe load paths should be defined for movement of heavy loads. However, the applicant had not provided procedures and equipment layout drawings that show safe load paths for movement of heavy

loads to minimize the potential to impact irradiated fuel in the reactor vessel and in the SFP and safe shutdown equipment. Therefore, the staff requested the applicant in RAI-SRP9.1.5-SBPB-05 to provide equipment layout drawings that show safe load paths and to provide a COL information item for COL applicants to provide procedures that define safe load paths.

In its response dated June 26, 2008, the applicant stated that the equipment layout drawings that show safe load paths are not provided in the DCD. This information is part of the operational programs and is covered by Section 13.4 of the DCD. The staff finds that Section 13.4 does not necessarily state that the COL applicant will develop procedures that show safe load paths. In evaluating the response, the staff determined that the applicant had not complied with the guidelines specified in NUREG-0800 Section 9.1.5 and NUREG-0612 as stated in that safe load paths had not been identified. Since the AP1000 is a standard design nuclear power plant where the location of all stationary equipment in non-site specific structures has been determined, generic safe load path figures should be developed for known heavy load lifts, such as the reactor vessel head and the spent fuel cask by the applicant for the AP1000 design.

To address the staff's concern, the applicant provided Revision 1 to its response to RAI-SRP9.1.5-SBPB-05, dated April 13, 2009. The applicant stated that a COL information item would be incorporated into the DCD requiring COL applicants to provide a heavy load handling program. The applicant proposed to modify DCD Section 9.1.5 to include a statement pointing out that DCD Section 13.5.1 addresses the development of heavy lift safe load paths. Following the guidance in NUREG-0800 Section 9.1.5 and NUREG-0612, the applicant proposed to modify DCD Section 13.5.1 to include the information that COL applicants referencing the AP1000 certified design would provide a heavy load handling program, which would include safe load paths for movement of heavy loads, to be referenced in procedures and shown on equipment layout drawings. DCD Section 13.5.1 includes the statement that the program and associated procedures will minimize the potential to impact irradiated fuel in the reactor vessel and in the SFP, and safe shutdown equipment from movement of heavy loads. The applicant also stated that it is currently developing drawings identifying safe load paths for the handling of heavy loads, which will then be provided to COL applicants for incorporation into their heavy load handling programs.

The applicant indicated that the revisions to DCD Tier 2 Sections 9.1.5 and 13.5.1 requiring that a COL applicant referencing the AP1000 certified design would provide a heavy load handling program, including safe load paths, would be added to the next DCD revision.

The staff finds the response acceptable since the applicant is developing standard safe load path drawings for heavy loads, which would be provided to the COL applicants. The applicant also added a COL information item for COL applicants to provide a heavy load handling program, which includes safe load paths to be referenced in procedures and shown on equipment layout drawings. It should be noted that the applicant has not created a completely new COL information item but added the requirement for a COL applicant to provide a heavy loads program to existing COL Information Item 13.5.1, which requires COL applicants to provide plant procedures.

The staff's concern described in RAI-SRP9.1.5-SBPB-05 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

NUREG-0800 Section 9.1.5 and NUREG-0612 provide guidance for applicants to describe a heavy load handling program for design, operation, testing, maintenance and inspection of heavy load handling systems. The applicant had not provided a heavy load handling program. Therefore, the staff requested the applicant in RAI-SRP9.1.5-SBPB-06 to provide a COL information item to ensure that the COL applicant will provide such a heavy load handling program.

In its response dated June 26, 2008, the applicant stated that it would provide the COL holders with the Operations and Maintenance Manuals for heavy load handling systems so that the manuals could be used when they created their programs. The applicant further stated that operations programs and procedures were discussed in Sections 13.4 and 13.5 of the DCD to include existing COL information items and no further COL Information Items were necessary. The staff determined that Sections 13.4 and 13.5 did not specify that the COL applicant would provide the heavy load handling program elements specified in NUREG-0800 Section 9.1.5 Section III.3 and NUREG-0612 Section 5.1.1. Therefore, the staff asked the applicant to revise the DCD to include a COL information item similar to RG 1.206, "Combined License Applications for Nuclear Power Plants," Regulatory Position Part III, Section C.I.9.1.5.

To address the staff's concern, the applicant provided Revision 1 to its response to RAI-SRP9.1.5-SBPB-06, dated April 13, 2009. The applicant's revised response states that this concern is addressed in the applicant's RAI-SRP9.1.5-SBPB-05, Revision 1 response.

The staff finds the applicant's RAI-SRP9.1.5-SBPB-05, Revision 1 response, as discussed above, adequately addresses the addition of a DCD COL information item requiring a COL applicant to develop a heavy loads handling program. Therefore, RAI-SRP9.1.5-SBPB-06 is resolved.

In DCD Revision 17, Section 9.1.5.2.2.2, "Component Descriptions," under the heading "Lifting Devices Not Specially Designed," (for the cask handling crane), the applicant states that for the handling of critical loads, dual or redundant slings are used, or a sling having a load rating twice that required for a non-critical load is used. Therefore, the staff requested in RAI-SRP9.1.5-SBPB-02 that the applicant explain how the statements are in conformance with the NUREG-0612 criteria, when a sling having a load rating twice that required for a non-critical load is used instead of a load rating twice that required for a critical load. This same issue is in Section 9.1.5.2.1.2 of DCD Revision 16.

In its response dated June 26, 2008, the applicant stated that in selecting the proper sling for rigging, the load capacity of the sling should be greater than the sum of the maximum static and dynamic load to be lifted. The applicant then stated that no matter what type of load is being lifted "critical lift or non-critical lift," the sling rating needs to take into account both static and dynamic loading. The response stated:

> But when you are making a "critical lift" then you must either:
>
> 1) Use two (2) of the properly rated slings per the formula above [the summation of the static and dynamic loads], OR
>
> 2) Use one (1) sling with twice (2x) the proper rating per the formula above.

In evaluating the response, the staff finds the applicant has clearly stated the application of the requirements of NUREG-0612 for selecting slings for lifting critical loads. The applicant has

properly stated that for critical loads, redundant slings must be used or the selected sling must have a load rating of twice the minimum required capacity. In DCD Revision 17, the applicant elected to use the term "non-critical load" in making a comparison between the normal selection of a sling for lifting a load and the selection of a sling or redundant slings under NUREG-0612 for lifting a load that has been designated a critical load. A "non-critical" load would require only a single sling rated for the summation of the static and dynamic load required to be lifted. However, if the same load was declared a "critical" load, the capacity of the single sling would have to be doubled, or twice the capacity required for the "non-critical" load. Based on its review, the staff finds the applicant's clarification of the DCD text acceptable and RAI-SRP9.1.5-SBPB-02 is resolved.

In DCD Revision 16, Sections 9.1.5.2.1.2 and 9.1.5.2.2.2, both titled "Component Descriptions," the applicant stated under the headings "Lifting Devices Not Specially Designed" that slings or other lifting devices not specially designed are selected in accordance with ANSI B30.9, "Slings," except that the load rating is based on the combined maximum static and dynamic loads that could be imparted to the sling. The two separate headings are associated with the polar crane and cask handling crane, respectively. NUREG-0800 Section 9.1.5 Revision 1, "Overhead Heavy Load Handling Systems," states in Paragraph III.4.C.ii (2), that slings should satisfy the criteria of ASME B30.9 and be constructed of metallic material (chain or wire rope). Therefore, the staff requested the applicant, in RAI-SRP9.1.5-SBPB-03, to explain how the criterion that slings be constructed of metallic material (chain or wire rope) for single-failure proof cranes and hoists that are handling critical loads is satisfied.

In its response dated June 26, 2008, the applicant stated that in addition to the polar crane and cask handling crane, the maintenance hatch hoist and equipment hatch hoist use lifting devices not specially designed that meet the safety factor requirements of ASME B30.26 2004, "Rigging Hardware." The applicant also stated the design specifications for the polar crane and cask handling crane reference ASME B30.9, "Slings," and the NRC Regulatory Issue Summary (RIS) 2005-25, Supplement 1, "Clarification of NRC Guidelines for Control of Heavy Loads," and that these references were added to the DCD. The supplement states that: Slings should satisfy the criteria of ASME B30.9-2003, "Slings," and be constructed of metallic material (chain or wire rope). Revision 17 of DCD Sections 9.1.5.2.1.2 and 9.1.5.2.2.2 was revised to indicate that the slings shall be constructed of metallic material (chain or wire rope) per NRC RIS 2005-25, Supplement 1.

The staff finds the applicant committed to using slings made from metallic material when lifting critical loads. The applicant demonstrated that it is in compliance with NUREG-0800 Section 9.1.5, Revision 1, Paragraph III.4.C.ii (2), which specifies that slings should satisfy the criteria of ASME B30.9 and be constructed of metallic material (chain or wire rope). Therefore, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP9.1.5-SBPB-03 is resolved.

In DCD Revision 16, Section 9.1.5.2.2.2, "Component Descriptions," under the heading "Special Lifting Devices" (for the cask handling crane), the applicant stated that the special lifting devices used for the handling of critical loads are listed in Table 9.1.5-2. The staff's review of Table 9.1.5-2 finds only special lifting devices for the polar crane and none for the cask handling crane. Existing plant operating experience demonstrates that a special lifting device is normally used between a cask and the cask handling crane hook due to the shape and size of the cask. Therefore, the staff requested the applicant in RAI-SRP9.1.5-SBPB-04 to explain if a special lifting device will be used between the cask and cask handling crane hook and if so, why it is not

listed in Table 9.1.5-2. If a special lifting device is not used, the staff asked the applicant to explain the anticipated rigging of the cask to the cask handling crane hook.

In its response dated June 26, 2008, the applicant stated that special lifting devices will be used with the cask handling crane and will be added to the DCD in Table 9.1.5-2. The applicant, in its response, provided a markup of DCD Table 9.1.5-2, "Special Lifting Devices Used for the Handling of Critical Loads." The applicant added the following special lifting devices to Table 9.1.5-2 for the cask handling crane: cask lift yoke, cask lift yoke extension, and loaded canister handling equipment. The following description was also added to the table for these devices: These devices are used for the handling of the casks and loaded canisters, which provide the interface between the cask handling crane and the shipping cask or loaded canister.

In DCD Tier 2, Table 9.1.5-2 Revision 17, the applicant revised the table to show the special lifting devices to be used with the cask handling crane and also provided a description of the use of these devices. With the addition of the cask handling crane special lifting devices to DCD Table 9.1.5-2, the staff finds the applicant's response acceptable and the staff's concern described in RAI-SRP9.1.5-SBPB-04 is resolved.

9.1.5.2.5 Increase in Polar Crane Maximum Load Rating

TR-106 Revision 1 states the critical lift for the polar crane is the lifting of the Integrated Head Package from the Reactor Vessel to the in-containment storage stand during a refueling outage. The critical lift weight for the polar crane has been increased from 249.48 metric tons (275 tons) to 272.16 metric tons (300 tons) to ensure adequate lifting margin. Because the weight of the integrated head package is the critical lift for the polar crane that requires the polar crane to have its maximum load rating increased to 272.16 metric tons (300 tons), while still being a single-failure proof crane in compliance with GDC 2, 4, 5 and 61, the staff finds this design change to be acceptable.

9.1.5.2.6 Addition of Cask Handling Crane Component Data

TR-106 Revision 1 states the AP1000 cask handling crane design was neither seismically qualified, nor single-failure proof to protect against dropping a cask. Since a cask drop could cause significant plant damage; a design change was initiated to upgrade the cask handling crane to single-failure proof. This change was made to reduce the possibility of a serious plant event.

With this design change the crane name was changed from the spent fuel shipping cask crane to the cask handling crane. With the changing of the design of the cask handling crane (previously spent fuel shipping cask crane in DCD Revision 15) to single-failure proof in DCD Revision 17, Table 9.1.5-1 has been completely changed to provide cask handling crane component data. The previous information in Table 9.1.5-1 under Revision 15 was for a non-single-failure proof spent fuel cask crane. DCD Revision 17 Table 9.1.5-1 provides design information for the cask handling crane bridge, trolley, main hoist and auxiliary hoist. Because the addition of the component data for the cask handling crane in Table 9.1.5-1 is similar to the data provided in Table 9.1.5-3 for the polar crane and does not affect the single-failure proof cask handling crane's compliance with GDC 2, 4, 5 and 61, the staff finds this design change to be acceptable.

9.1.5.2.7 Polar Crane Main Hook Used to Lift RCP

TR-106 Revision 1 states the main hook on the polar crane, instead of the auxiliary hook of the polar crane, will be used to install and remove the RCPs. Since the main hook has a larger load capacity than the auxiliary hook and is single-failure proof, the applicant remains in compliance with GDC 4. Therefore the staff finds this design change to be acceptable.

9.1.5.2.8 Decrease in Polar Crane Auxiliary Hook Capacity

TR-106 Revision 1 states the auxiliary hook on the polar crane has been reduced from 68.04 metric tons (75 tons) to 22.68 metric tons (25 tons) now that it is no longer being used to install and remove the RCPs. Because the polar crane main hook will now be used with the RCP special lifting device instead of the auxiliary hook, the capacity of the auxiliary hook can be reduced from 68.04 metric tons (75 tons) to 22.68 metric tons (25 tons). The staff finds that with the change in lift capacity of the auxiliary hook, the polar crane remains a single-failure proof crane and in compliance with GDC 2, 4, 5 and 61.

9.1.5.2.9 Additional Staff Inquiries Associated With Overhead Heavy Load Handling

In RAI-SRP9.1.5-SBPB-07, the staff asked the applicant the following six clarifying questions to provide a better understanding of the applicant's submitted technical information:

a) In RAI-SRP9.1.5-SBPB-07, Subpart a, the staff asked the applicant to clarify if the cask handling crane was to be operated from either a radio remote control, operator's cab or a pendant suspended from the crane since it was not totally clear after a review of the information in TR-106 Revision 1.

In its response dated June 26, 2008, the applicant stated that the cask handling crane will be operated by radio remote control or from a pendant suspended from the crane and that the crane does not have an operator's cab.

Based on its review, the staff finds the applicant's response clarifies the wording in TR-106 and is consistent with DCD Revision 17, in that the applicant clarifies that the cask handling crane will be operated by either radio remote control or from a pendant suspended from the crane. The applicant also updated DCD Sections 9.1.5.2.1 and 9.1.5.2.2.1 to include the use of radio remote control for the polar and cask handling cranes. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart a is resolved.

b) In RAI-SRP9.1.5-SBPB-07, Subpart b, the staff asked the applicant to clarify why in TR-106 Revision 1 for Design Change 256 the title of the design change is for the maintenance hatch hoist design while the cask handling crane is mentioned twice in the text explaining the design change.

In its response dated June 26, 2008, the applicant provided an excerpt from TR-106 Revision 1 of the text from Design Change 256 with the cask handling crane words crossed out in two places and replaced with the words Maintenance Hatch Hoist. No technical information that would affect the DCD was changed.

Based on its review, the staff finds the applicant's response acceptable because the applicant proposed corrections to the identification of the crane in TR-106 Revision 1, Design Change 256 from the cask handling crane to the maintenance hatch hoist; making the design change text

agree with the design change title in TR-106. Therefore, no change to the DCD was needed and the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart b is resolved.

c) In RAI-SRP9.1.5-SBPB-07, Subpart c, the staff asked the applicant to provide the correct section to which a reader is referred in the paragraph under Section 9.1.5.2, System Description, of AP1000 DCD Revision 16, since the currently referenced section does not exist.

In its response dated June 26, 2008, the applicant stated the correct referenced section is 9.1.5.3, not 9.1.5.2.3, and provided a markup of paragraph 9.1.5.2 from DCD Revision 16 showing the correction.

In DCD Tier 2, DCD Section 9.1.5.2 Revision 17, the applicant revised the referenced subsection to show 9.1.5.3. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart c is resolved.

d) In RAI-SRP9.1.5-SBPB-07, Subpart d, the staff provided an excerpt from TR-106, Revision 1, Design Change Description for Design Change 170 (polar crane Design) which states: The critical lift for the polar crane is the lifting of the integrated head package (IHP) from the reactor vessel head to the in-containment storage stand during a refueling outage. The staff asked the applicant to clarify if the IHP is lifted by the polar crane from the reactor vessel head or from the reactor vessel, which components comprise the IHP, and if the IHP and reactor vessel head are two separate lifts by the polar crane.

In its response dated June 26, 2008, the applicant stated that the IHP is lifted by the polar crane from the reactor vessel. The IHP includes the reactor vessel head, the shield shroud, the control rod drive mechanism cooling fans, and the lifting rig. The IHP and the reactor vessel head are not two separate lifts.

Based on its review, the staff finds the applicant's response acceptable because the applicant clarified that the IHP is not lifted from the reactor vessel head as stated in TR-106, Revision 1, Design Change Description for Design Change 170. The IHP is lifted from the reactor vessel by the polar crane. The IHP and the reactor vessel head are not two separate lifts and the IHP is properly defined in Section 3.9.7 of the DCD. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart d is resolved.

e) In RAI-SRP9.1.5-SBPB-07, Subpart e, the staff asked the applicant why the title for Section 9.1.6.5 of the AP1000 DCD Revision 16 was "Inservice Inspection Light Load Handling System," and did not include the OHLHS.

In its response dated June 26, 2008, the applicant stated the title for Section 9.1.6.5 should be changed to "Inservice Inspection Load Handling System" from "Inservice Inspection Light Load Handling System." Based on its review, the staff finds the applicant's response acceptable because the applicant proposes to revise the title of DCD Section 9.1.6.5 to cover both light and heavy load handling systems.

In DCD Tier 2, DCD Section 9.1.6.5 Revision 17, the applicant changed the name of this section from Inservice Inspection Light Load Handling Systems to Inservice Inspection Load Handling Systems. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart e is resolved.

f) In RAI-SRP9.1.5-SBPB-07, Subpart f, the staff stated that the cask handling crane was still referred to as the spent fuel shipping cask crane in Tier 1 Section 2.3.5, Item Number 4 of AP1000 DCD Revision 16 and the section needed correction.

In its response dated June 26, 2008, the applicant states the cask handling crane should not be referred to as the spent fuel shipping cask crane and the DCD needs to be changed in four sections to make the correction. The applicant provides two excerpts from Tier 1 and two excerpts from Tier 2 of the DCD showing where the words spent fuel shipping cask had been crossed out and replaced with cask handling. Based on its review, the staff finds the applicant's response acceptable because the applicant corrected the name of the cask handling crane from spent fuel shipping cask crane in four sections of DCD Revision 17. In DCD Tier 1, Revision 17, Section 2.3.5 and Table 2.3.5-2, the applicant changed the name of the previously shown spent fuel shipping cask crane to cask handling crane. In DCD Tier 2, Revision 17, Sections 9.1.5.3 and 14.2.9.4.14, the applicant also changed the name of the previously shown spent fuel shipping cask crane to cask handling crane. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart f is resolved.

In RAI-SRP9.1.5-SBPB-10, the staff requested additional information related to heavy load drop analysis. In the AP1000 DCD Tier 2, Revision 16, Section 9.1.5.3, "Safety Evaluation," the applicant states: "Postulated load drops are evaluated in the heavy load analysis." In the last sentence of Section 9.1.5.3 the applicant states, "The heavy load analysis is to confirm that a postulated load drop analysis does not cause unacceptable damage to reactor fuel elements, or loss of safe shutdown or decay heat removal capability." The staff asked the applicant to describe what heavy load drop analyses were performed and to describe the results of the analyses.

In its response dated September 3, 2008, the applicant stated that the polar crane, cask handling crane, equipment hatch hoist, and maintenance hatch hoist are single-failure proof, which satisfies the requirements for moving heavy loads, and no heavy load drop analysis was performed. The applicant added single-failure proof criteria to ITAAC Table 2.3.5-2 in DCD Revision 17 for all four of these load handling systems, as identified in its response to RAI-SRP9.1.5-SBPB-09. Additionally, the applicant stated that the main steam isolation valve (MSIV) monorail hoists A and B are used to perform maintenance on the MSIVs. However, the hoists will not be used during plant operation and, therefore, failure of the hoists (while lifting loads) will not prevent the plant from shutting down safely because the plant will already be shut down.

Based on its review, the staff finds acceptable the part of the applicant's response, that states that no heavy load drop analyses are required for the polar crane, cask handling crane, equipment hatch hoist, and maintenance hatch hoist because they are designed single-failure proof. However, the staff determined that the response was unacceptable regarding the MSIV monorail hoists A and B, because the applicant did not address the effect of a load drop on equipment needed for decay heat removal.

To address the staff's concern, the applicant provided Revision 1 to its RAI-SRP9.1.5-SBPB-10 response, dated January 28, 2009. The applicant stated the plant modes in which the main steam isolation values must be operable. Because the MSIVs and main steam safety valves (MSSVs) have to be operable or closed during plant modes 1, 2, 3, and 4, the MSIV monorail hoists shall not be used to service the MSIVs or MSSVs during Modes 1, 2, 3, or 4. The applicant further stated that during Modes 5 and 6 the steam generators are not utilized for

nonsafety-related decay heat removal and the MSIVs could be taken out of service. Therefore, a load drop by the MSIV monorail hoists during modes 5 or 6 would not affect decay heat removal capability of the AP1000.

Based upon its review, the staff determined that the response was unacceptable regarding the MSIV monorail hoists A and B, because the applicant did not address the effect of a load drop on equipment needed for decay heat removal.

To address the staff's concern, the applicant provided Revision 2 to its RAI-SRP9.1.5-SBPB-10 response, dated April 13, 2009. The applicant's revised response states that equipment and components required for decay heat removal during Modes 5 or 6 are not located in the load path for the MSIV monorail hoists.

In the revised response, the applicant proposed revisions to DCD Tier 2 Section 9.1.5.3 that explicitly state that the equipment and components required for decay heat removal during Modes 5 or 6 are not located in the load path of the MSIV monorail hoists.

The staff finds the response acceptable since the applicant clarified that equipment and components required for decay heat removal during Modes 5 or 6 are not located in the load path for the MSIV monorail hoists. The staff's concern described in RAI-SRP9.1.5-SBPB-10 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.5.2.10 Staff Inquiry Regarding Design Changes in TR-106 not reflected in DCD Revision 16

In RAI-SRP9.1.5-SBPB-08, the staff stated that TR-106, Revision 1 in Section V described post AP1000 Revision 16 changes. The staff provided paragraphs a through d of TR-106, Section V showing the post DCD Revision 16 changes and asked the applicant to verify if the changes would be documented in the DCD.

In its response dated June 26, 2008, the applicant stated that all the post AP1000 Revision 16 design changes described in TR-106, Revision 1, Section V would be incorporated in Revision 17 of the DCD.

In Revision 17 of the DCD Tier 2, Sections 9.1.5.2.1.3, 9.1.5.2.2.3 (both sections titled "Instrumentation Applications"); 9.1.5.3, "Safety Evaluation": Table 9.1.5-1, "Cask Handling Crane Component Data"; and Table 9.1.5-3, "Polar Crane Component Data," the applicant has incorporated the changes shown in TR-106 Revision 1 Section V.

In the three DCD Tier 2 sections and two DCD Tier 2 Tables listed above, the applicant proposed changes to align DCD Section 9.1.5 and the associated tables with ASME NOG-1 and NUREG-0554 for single-failure proof cranes.

Specific DCD Revision 17 changes include:

1. In DCD Section 9.1.5.3 the applicant stated that either redundancy or double design factor for load bearing components such as the hoisting ropes, sheaves, equalizer assembly, hooks and holding brakes of single-failure proof cranes is permitted.

The staff determined that the proposed change for potentially allowing redundancy for all load bearing components was unacceptable because it describes a non-redundant equalizer device

as acceptable and because the new statement regarding which components must be redundant was ambiguous. Every component of the reeving system must be redundant except the rope drum, the upper block, the load block, and the hook (which is part of the load block). Among the components listed in TR-106, only design of the hook to twice the normal design load is acceptable in place of redundant hooks. The staff determined that the reference to NUREG-0554, Paragraph 4.3 in TR-106 does not provide the criteria for alternatives to redundancy in load attachment points; those criteria are in Appendix C to NUREG-0612 and ASME NOG-1.

In a letter dated September 8, 2009, the applicant provided a revised response to RAI-SRP9.1.5-SBPB-08. The applicant provided additional changes to DCD Section 9.1.5.3 to provide additional clarification to comply with ASME NOG-1. The staff finds the proposed changes comply with NOG-1 and are acceptable. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-08 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

2. In DCD Tables 9.1.5-1 and 9.1.5-3, the applicant changed wording for the Bridge and Trolley sections to reflect that service, emergency, and parking functions may be performed by a single friction brake.

3. In DCD Sections 9.1.5.2.1.3 and 9.1.5.2.2.3, the applicant stated that the hoist in the raising direction for the polar crane and cask handling crane has block-actuated limit switches, which directly interrupt power to the hoist motor and cause the hoist brakes to set.

4. In DCD Tables 9.1.5-1 and 9.1.5-3, the applicant added information on control, holding and emergency brakes for the main hoist and auxiliary hoist.

TR-106 Revision 1 states that changes 2, 3 and 4 were made to better align the DCD design data for the single-failure proof polar crane and cask handling crane. The staff finds changes 2, 3, and 4 acceptable, since the guidance in NUREG-0554, as supplemented by ASME NOG-1 for single-failure proof cranes, is followed. In accordance with NUREG-0800 Section 9.1.5, cranes designed to the criteria of NOG-1 for a Type 1 crane are acceptable under the guidelines of NUREG-0554 for construction of a single-failure proof crane.

9.1.5.3 Conclusion

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

The staff reviewed the applicant's proposed changes to the AP1000 OHLHS as documented in AP1000 DCD, Revision 17, and in TR-106, Revision 1, which include a design description for the cask handling crane, an upgrade designation of the cask handling crane to seismic Category I and single-failure proof, an upgrade of the maintenance hatch hoist to single-failure proof, a designation of ASME NOG-1 for a Type 1 crane for single-failure proof cranes and hoists, the replacement of pendant control to remote control for the polar crane, and the change in design capacity of the main and auxiliary hooks of the polar crane. The staff concludes that the AP1000 OHLHS design continues to meet all applicable acceptance criteria and the proposed changes are properly documented in the updated AP1000 DCD. The proposed

changes meet the criteria of; 10 CFR 52.63(a)(1)(vi), on the basis that they substantially increase overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information in the AP1000 DCD. Therefore, the staff finds that the proposed changes to the AP1000 OHLHS are acceptable.

9.2 Water Systems

9.2.1 Service Water System

9.2.1.1 Summary of Technical Information

Revision 17 of the AP1000 DCD includes proposed changes to the service water system (SWS) in order to accommodate increased heat loads, establish increased wet and dry bulb temperature limits for the site, and to provide additional design flexibility for COL applicants. The details of these proposed changes are discussed in TR-111, "Component Cooling System and Service Water System Changes Required for Increased Heat Loads," APP-GW-GLN-111, Revision 0, dated May 2007; TR-108, and TR-103.

In AP1000 DCD Revision 17, the applicant proposed Tier 2 changes associated with SWS Section 9.2.1.2.2 "Component Description – Piping Requirements." The applicant corrected the referenced code to use for nonmetallic piping from "ASME B31.1" to "ANSI B31.1, Appendix III."

The applicant proposed a change to Tier 1 Table 5.0-1, "Site Parameters," to increase the maximum safety coincident wet bulb temperature from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F) and non-coincident wet bulb temperature from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F). The details of this change are discussed in APP-GW-GLE-036, Revision 0. Because this change could impact the SWS performance, it is evaluated below. In Revision 17, the applicant made no changes to DCD Tier 1, Section 2.3.8, "Service Water System."

9.2.1.2 Evaluation

The regulatory basis for evaluating the SWS is documented in Section 9.2.1, "Station Service Water System," of NUREG-1793. While the SWS (including heat sink) is nonsafety-related, it is considered to be important to safety because it supports the normal defense-in-depth (DID) 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 SWS makes it subject to RTNSS in accordance with the Commission's RTNSS policy in SECY-94-084 for passive reactor plant designs. The staff's evaluation of the changes that are proposed focuses primarily on confirming that the changes will not adversely affect safety-related SSCs, the capability of the SWS to perform its DID and RTNSS functions, and the adequacy of ITAAC, test program specifications, and RTNSS availability controls that have been established for the SWS. The proposed changes were evaluated using the guidance provided by NUREG-0800 Section 9.2.1, "Station Service Water System," as it pertains to these considerations. Acceptability is judged based upon conformance with the existing AP1000 licensing basis, the guidance specified by NUREG-0800 Section 9.2.1 (as applicable), and SECY-94-084.

The specific criteria that apply to the changes referred to above include 10 CFR 52.63(a)(1)(iii), which concerns the proposed changes reducing unnecessary regulatory burden while maintaining protection to public health and safety and the common defense and security; and 10 CFR 52.63(a)(1)(vii), which concerns the proposed changes contribution to increased standardization of the certification information.

9.2.1.2.1 SWS Design Changes Required for Increased Heat Loads

As discussed in TR-111, increased CCS heat loads and flow rates have resulted in the need for corresponding increases in SWS flow and pump capacity, pipe and component size, and cooling tower heat dissipation capability and makeup rate. The proposed SWS changes are reflected in the AP1000 DCD, Tier 1, Table 2.3.8-2, ITAAC Design Commitment Item 2 relative to the capability to support plant shutdown and spent fuel cooling; Tier 2, Table 9.2.1-1, "Nominal Service Water Flows and Heat Loads at Different Operating Modes;" and Tier 2, Table 16.3-2, "Investment Protection Short-Term Availability Controls (IPSAC)," Section 2.4, "Service Water System (SWS) – RCS Open," SR 2.4.1 with respect to the minimum required SWS flow rate. Section 9.2.2 provides an evaluation of the increased CCS heat loads.

Proposed Increases in SWS Flow Rate and Heat Dissipation Capability

The applicant proposed to increase the minimum required SWS flow rate and heat dissipation capability in order to accommodate the higher heat loads that are proposed for the AP1000 plant design. The values listed in Tier 2 of the DCD, Table 9.2.1-1, and the ITAAC specified by Tier 1 of the DCD, Table 2.3.8-2, reflect these changes. The ITAAC requires COL applicants to demonstrate that the SWS design is capable of supporting plant shutdown and spent fuel cooling. The applicant proposed to change the flow rate value in the ITAAC that demonstrates, by testing, that each SWS pump can deliver at least 37,854 Lpm (10,000 gpm) flow to each CCS HX. The applicant also proposed to increase the required heat transfer rate that is specified for each cooling tower cell to be greater than or equal to 1.8×10^8 kilojoules per hour (kJ/h) (1.7×10^8 Btu/h) at a cold water temperature of 32.2 °C (90 °F). The proposed changes in the ITAAC acceptance criteria relative to SWS flow rate and heat load are consistent with the revised values that are reflected in Tier 2, Table 9.2.1-1, and they appear to be acceptable from this perspective. However, the applicant did not specifically identify how the revised values were determined and on what basis they are considered to be appropriate. For example, the applicant did not explain how the maximum heat load was determined and how much margin is afforded by the ITAAC acceptance criterion. The applicant did not compare the available margin with the amount of margin that is needed based on industry experience to accommodate degradation that is anticipated to occur over time and provide necessary flexibility. Also, the applicant did not explain why the bases for these values and the industry experience that is credited applies to all COL applicants.

The applicant was asked, in RAI-SRP9.2.1-SBPA-01, to provide a more detailed description of the basis for the proposed changes relative to SWS flow rate and cooling tower performance in Tier 2 of the DCD. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant referred to information in the application that was considered by the staff when preparing this question, but a more detailed description was not provided. The applicant also provided information to explain how cooling tower performance and SWS flow rates would be maintained over time. While this information is useful, no provisions were established to assure implementation by COL applicants. The RAI response did not adequately address RAI-SRP9.2.1-SBPA-01 because more information was needed regarding RTNSS and systems supporting the ability to achieve cold shutdown operations.

During the June 25, 2009 audit, the staff explained that in order to resolve the RAI, the applicant should provide more detail explaining how the design margins were established and the maintenance and testing activities that ensure that the margins are adequately preserved over the life of the plant.

In a letter dated August 31, 2009, the applicant provided additional information specific to the design margins, appended to its original response to RAI-SRP9.2.1-SBPA-01. Additionally, the applicant provided a proposed mark up of DCD Tier 2, Section 9.2.1.2.3.4 that supports the information provided in the additional information.

The response identified that the SWS is designed with sufficient margin to ensure that system flow rates and cooling tower performance will be maintained such that RTNSS and DID functions can be performed over the life of the plant. The response described the following functions that meet criteria for DID functions of the CCS as they relate to the SWS:

- For normal residual heat removal system (RNS) cooling, the SWS and CCS are needed to cool the RNS HXs and pumps during RCS cooldown and cold shutdown operation, in order to avoid actuation of the passive residual heat removal (PRHR) HX. The SWS and CCS also provide cooling to the RNS during refueling operation, to avoid heat up of the water in the refueling cavity.

- For SFS cooling, the SWS and CCS provide cooling to the SFS HXs during all modes of plant operation to prevent heat up and boiloff of water in the SFP.

- For CVS miniflow HX cooling, the SWS and CCS provide cooling to the miniflow HXs of the CVS injection pumps. This allows proper operation of the CVS injection pumps, in order to avoid core makeup tank (CMT) actuation.

- For reduced inventory cooling, the SWS and CCS provide cooling to the RNS HXs and pumps during reduced reactor coolant inventory operation.

The response specified that the SWS also supports a function of the CCS that is important for equipment protection by providing necessary cooling to the RCP external HX to avoid reaching high bearing water temperatures and its resulting RCP trip.

The response also identified the SWS heat loads as requested by the staff. The heat loads include the RCP external HX, the SFS HX, and the CVS miniflow HX. The response stated that the cooling functions for the CCS and SWS are dependent on the temperature supplied by the CCS HX. The response also explained that, of these three functions, the RCP external HX cooling function has the lowest design temperature requirements and, therefore, provides the high CCS design temperature.

The ability of the CCS to meet the RCP cooling requirement was evaluated by considering the design operating mode for the CCS HX, which is normal power operation. The CCS HX overall coefficient of heat transfer (U) and required heat transfer area (A) was computed by ensuring that the high CCS design temperature limit was not exceeded during this design case. The total normal power operating heat duty included the maximum SFS heat duty (which is immediately after refueling), maximum heat dissipation by the RCPs, as well as a 20 percent margin above the maximum central chilled water system (VWS) chiller heat load. The SWS temperature

supplied to the CCS HX also included additional margin, since the maximum cooling tower approach temperature was added to the maximum safety wet bulb.

The response explained that the SWS cooling towers are sized for both plant cooldown (higher heat loads) and normal power operation (lower heat loads). The CCS HX design is able to supply cooling water that meets all of the DID and investment protection cooling requirements under the most limiting conditions. The CCS HX specification also includes an additional 10 percent of heat transfer area above the design value to account for fouling and degradation over the HX's operating life. Additional frame length is also included so that 20 percent more than the nominal plate number required to provide the design heat transfer capability can be added to the HX if additional performance is needed.

SWS cooling tower performance is maintained by providing substantial margin in the sizing of the cooling tower. The SWS cooling towers were sized using a peak cooldown heat duty that includes significant conservatism. The cooling towers must be able to remove sufficient heat to cool down the RCS via the RNS, starting 4 hours after reactor shutdown, to cold shutdown conditions 96 hours after shutdown, assuming the persistence of the ambient wet bulb temperature at the maximum normal value of 26.7 °C (80.1 °F). This assumption itself provides substantial margin since the definition of the AP1000 maximum normal temperature value is based on the 1 percent seasonal exceedance wet bulb temperature, which can be experienced for fewer than 30 hours per year. The sizing case assumes that all four RCPs and variable frequency drives (VFDs) are operating at maximum allowable speed, though procedurally only two RCPs and RCP VFDs should run at 50 percent speed (or less) during this condition. Significant margins in the SFS and VWS chiller heat loads were included, as they were for the normal power operating design case. The margin included in these major heat loads, as well as several others, results in approximately 30 percent margin in the SWS cooling tower design heat duty, with respect to the expected heat duty during a realistic plant cooldown. Since the SWS tower cells are sized to meet cooldown time requirements under this extremely conservative operating case, any long term degradation in cooling ability under more realistic heat duties would not prevent the tower from meeting its heat transfer performance requirements.

It should also be noted that the DID and RTNSS functions of RNS cooling during RCS cooldown and RNS cooling during reduced coolant inventory operation require that the CCS provide cooling during these operating modes, and does not require specific temperature limitations or impose defined time to temperature requirements on the CCS and SWS. The CCS and SWS need to be designed to prevent heat up of the RCS if one train of CCS and SWS is not operable. Heat-up needs to be prevented even under maximum normal ambient wet bulb temperature conditions, though the time to cool down can be extended. This capability of the CCS and SWS also ensures that the DID function of the CCS and SWS can be fulfilled even assuming significant degradation in the CCS HX and SWS cooling tower performance, when both trains are operable. Similarly, ambient conditions above the maximum normal wet bulb temperature would not prohibit the CCS and SWS from performing this function, though cooldown times would be extended. The ability of the CCS and SWS to cool the RNS HXs during reduced coolant inventory operation (Modes 5 and 6) is further assured since the heat duty in this mode of operation is significantly reduced. The overall heat duty of the CCS and SWS in this mode is approximately 33 percent of the heat duty at the beginning of normal RNS cooldown,

The SWS pumps are required to provide a nominal flow rate of 39,750 Lpm (10,500 gpm) to cool the CCS HX for all normal operating modes, though a degraded minimum flow rate of

(10,000 gpm) can support decay heat removal from the RNS and SFS systems. In the RAI response, the applicant stated that the SWS flow analysis indicates that the selected SWS pump delivers 39,750 Lpm (10,500 gpm) to the CCS HX for all normal operating modes with all flow resistances (k-values) in the system increased by 10 percent. An additional 7 percent margin in pump-developed head at the design point is added to the system pump curve, specifically to offset any long-term degradation of pump performance during the life of the plant.

There is also a SR in the SWS investment protection short-term availability controls to verify that each SWS pump provides a flow rate of 37,850 Lpm (10,000 gpm) one day prior to entering Mode 5 and reduced-inventory operation.

The staff's review of the applicant's response to RAI-SRP-9.2.1-SBPA-01 finds that it adequately explains how the margins in SWS system design were developed. Also the staff finds that the response adequately explained how the flow margins would be tested and maintained for the life of the plant since testing/surveillance is performed before entering Mode 5. The staff also reviewed the proposed changes to DCD Section 9.2.1.2.3.4 and finds that they adequately provide a high level description of the SWS margin. Therefore, RAI-SRP9.2.1-SBPA-01 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Design Considerations

In addition to accommodating the higher CCS heat load while maintaining appropriate SWS flow velocities and pressure drops, the SWS return temperature to the cooling tower is not allowed to exceed 48.9 °C (120 °F) under design heat load conditions. As discussed in Section 3.4.1 of TR-111, this limit is set to prevent long-term degradation of the cooling tower fill material. In order to assure that this temperature limit will not be exceeded upon initiation of shutdown cooling when the SWS heat load is maximized at 3.65×10^8 kJoules/hr (3.46×10^8 Btu/hr), the applicant proposes to use larger capacity SWS pumps with a flow rate of 37,747 Lpm (10,500 gpm) per train. The use of larger capacity pumps will avoid delays when cooling down the plant in preparation for refueling and this is considered to be acceptable by the staff since the shutdown timeframe is enhanced. However, this design capability does not ensure that plant operators will adhere to this temperature limit. The staff requested, in RAI-SRP9.2.1-SBPA-02, that the applicant identify and describe in Tier 2 of the DCD those SWS design limitations that should be adhered to and explain how adherence to these limitations is assured. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant provided additional information about the system design margins, alarms, and capabilities, and discussed cooling tower fill material options that are available for higher temperature situations. However, the response did not adequately address the staff's request in that SWS design limitations and provisions to ensure adherence were not discussed. More information was requested in RAI-SRP9.2.1-SBPA-02 regarding RTNSS and systems supporting the ability to achieve cold shutdown operations.

During the June 25, 2009 audit, the staff explained that in order to resolve this RAI, the applicant should provide more detail identifying in Tier 2 of the DCD those SWS design limitations that should be adhered to and to explain how these limitations are assured including the addition of instrumentation if required. In a letter dated August 31, 2009, the applicant provided additional information specific to the design limitations appended to its original response to RAI-SRP9.2.1-SBPA-02.

The applicant agreed that SWS cooling tower basin water level Instrumentation was necessary and should be included in ITAAC. The response stated that this parameter is needed to verify that SWS pumps will be supplied with adequate net positive suction head (NPSH). Testing to verify adequate NPSH is also discussed in Tier 2 Section 14.2.9.2.6, "Service Water System Testing," in the item (d) of the 'General Test Acceptance Criteria and Methods. The applicant proposed to revise DCD Tier 1 Table 2.3.8-2 to identify the service water cooling tower basin level instrument, SWS-009.

The staff reviewed the applicant's additional information in its response to RAI-SRP9.2.1-SBPA-02 and finds that it provides an adequate means to measure the service water cooling tower basin level and ensures adequate SWS NPSH that can be verified by testing. The staff also reviewed the proposed changes to the DCD and finds that they adequately provide the instrumentation required to ensure adequate SWS NPSH. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The proposed increase in SWS capacity and flow rate is expected to result in an increase in the minimum water inventory that must be maintained in the cooling tower basin. The applicant did not describe specifically what water inventory should be maintained in the cooling tower basin in order to support SWS operation and, in particular, to assure adequate net positive suction head for the SWS pumps. In RAI-SRP9.2.1-SBPA-03 the staff asked the applicant to identify and describe in Tier 2 of the DCD the cooling tower basin water inventory requirements, the basis for this determination, and how this inventory is assured to be maintained. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant provided information to explain the design capability of the cooling tower basin inventory and while this information is useful, specific quantitative information is needed to verify the adequacy of design. For example, the specific NPSH requirements for the SWS pumps was not provided and design details of the cooling tower basin were not provided to show that the NPSH requirement was satisfied; other considerations such as vortexing were not addressed. Furthermore, this information was not included in the Tier 1 and Tier 2 descriptions as appropriate, and there was no discussion about how implementation of operational assumptions is assured. Additionally, the response made reference to use of a raw water system (RWS) for providing makeup to the SWS cooling tower basin, which has not been described for the AP1000 design in accordance with 10 CFR 52.47(24), "Contents of applications; technical information" requirements. Consequently, the response did not adequately address RAI-SRP9.2.1-SBPA-03.

During the June 25, 2009 audit, the staff explained that in order to resolve this RAI, the applicant should provide more information on its evaluation of whether it is reasonable to have a cooling tower level requirement prior to going to mid-loop operations for RTNSS considerations, and entering Modes 5 and 6. The staff also asked the applicant to evaluate and describe the differences in water level versus usable volume (in the SWS cooling tower basins) between normal operation and RTNSS.

In a letter dated August 31, 2009, the applicant provided additional information specific to the design limitations appended to its original response to RAI-SRP9.2.1-SBPA-03. In the response, the applicant agrees with the need to add SWS cooling tower basin usable volume to the ITAAC for SWS. A minimum SWS cooling tower basin reserve volume is required to provide water inventory for up to 12 hours when normal makeup capability from the RWS is lost. The basin is sized to contain 870,600 L (230,000 gallons) between the low level alarm setpoint (elevation 30.2 m (99 ft)) and the lowest usable level (28.5 m(93 ft, 6 in)), which coincides with

the low-low level alarm setpoint. This is the minimum cooling tower usable volume needed to support a plant cooldown for 12 hours without makeup. The response states that a criterion will be added to the SWS ITAAC to ensure that the SWS cooling tower basin is constructed to provide this minimum reserve volume with water level at the low level alarm setpoint. Additionally, the applicant proposes to revise Table 2.3.8-2, ITAAC Item 2 to add testing requirements confirming that the SWS cooling tower basin has adequate reserve volume of at least 870,600 L (230,000 gallons) corresponding to its low level alarm setpoint. Additionally, the applicant proposed to revise Tier 2 DCD Section 9.2.1.2.2 for the cooling tower to identify that a minimum usable volume of 870,600 L (230,000 gallons) exists.

The staff reviewed the applicant's additional information in its response to RAI-SRP9.2.1-SBPA-03 and finds that it adequately identifies that the service water cooling tower basin level will have a usable minimum of 870,600 L (230,000 gallons) corresponding to its low level alarm setpoint. The staff also reviewed the proposed changes to the DCD and finds that they adequately provide ITAAC criteria ensuring the testing to confirm the usable volume. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The increased water inventory and flow rates in the SWS piping and cooling towers could result in more severe flooding consequences than previously analyzed. The information that was provided did not describe the impact of the proposed SWS modifications on the consequences of flooding and whether or not safety-related equipment could be adversely affected. The applicant was asked, in RAI-SRP9.2.1-SBPA-04, to address the potential impact of the proposed SWS modifications on safety-related equipment and on the consequences of flooding.

In a letter dated June 26, 2008, the applicant provided information specific to flooding. In the response the applicant stated that there is no safety-related equipment in the turbine building. The component cooling water and service water components on elevation 30.5 m (100 ft), which provide the RTNSS support for the normal RNS are expected to remain functional following a flooding event in the turbine building since the pump motors and valve operators are above the expected flood level. Flooding caused by an SWS piping break in the compartment at the southern end of the turbine building will flow out the open doorway to the turbine hall, and openings in the base of the wall between the compartment and the turbine hall. The increased flooding rate and volume associated with the increase in SWS pump size and basin inventory will not challenge the operability of the CCS pumps nor any other DID or Investment Protection equipment located in this area. Flooding from the circulating water system (CWS) (or from any other turbine hall water source) will flow out of the turbine building onto the ground through access doors without affecting equipment at the southern end of the building.

The staff reviewed the RAI response and concluded that the RTNSS components (CCW and SWS) would remain functional since the pump motors and valve operators are above the expected flood levels and there are no safety-related components in the turbine building. Therefore, RAI-SRP9.2.1-SBPA-04 is considered closed.

9.2.1.2.2 Proposed Increase in the Maximum SWS Supply Temperature

Proposed Revision 16 of the DCD, Tier 2, Sections 9.2.1.1.2 and 9.2.1.2.3.3, reflects an increase in the maximum allowed SWS cooling water temperature being supplied to the CCS HXs during normal power operation. The proposed temperature increase is from a value of 31.7 °C (89 °F) to 34.2 °C (93.5 °F). This is not the same as the maximum SWS supply temperature of 31.4 °C (88.5 °F) that is specified in Tier 2 of the DCD, Section 9.2.1.2.3.4, for

plant cooldown/shutdown. This change is not discussed or explained in the information that the applicant provided, and it is not clear how the SWS can perform its DID and RTNSS functions if the SWS supply temperature exceeds the limit that is assumed for shutdown cooling. In RAI-SRP9.2.1-SBPA-05 the staff requested that the applicant describe in Tier 2 of the DCD the basis and justification for the proposed increase in the maximum allowed normal operating SWS supply temperature.

The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant discussed differences in assumptions that were used for dissipating heat during normal power operation and during plant cooldown. The difference in the cooling water temperature values noted above stems from the use of a less conservative wet bulb temperature for the most limiting SWS DID function (plant cooldown with concurrent spent fuel cooling). However, the applicant did not explain how the SWS DID and RTNSS functional capabilities are assured for those periods when humidity is at its maximum. There is an increased chance of plant trip during hot, humid conditions due to increased electrical demand and it is illogical to assume less humid conditions for the plant cooldown case than what can be experienced during normal power operation. In accordance with SECY-94-084, the SWS should be capable of performing its DID and RTNSS functions over the full range of postulated operating conditions, and the applicant has not demonstrated this to be the case. The applicant response also referred to another increase that is proposed for the maximum safety wet bulb temperature in order to accommodate the Levy site. Because the RAI response did not adequately address the DID and RTNSS functional capabilities of the SWS over the full range of plant operating conditions, satisfactory resolution of RAI-SRP9.2.1-SBPA-05 was not achieved.

On March 18, 2009, the staff conducted a public meeting with the applicant to discuss this RAI. After this meeting, the applicant provided a revised RAI response on May 13, 2009, which addressed wet bulb temperature values related to the SWS to accommodate the Levy site environmental parameters. The applicant stated that the maximum normal non-coincident wet bulb temperature remains at 26.7 °C (80.1 °F) for RTNSS and DID SWS functions and that there was sufficient margin in the system and component design.

In a letter dated May 13, 2009, the applicant stated that higher ambient temperatures (30.0 °C (86.1 °F) vs. 26.7 °C (80.1 °F)) will not impact safety or investment protection and would only result in an extended time to achieve cooldown. During the June 25, 2009 audit, the applicant clarified its original response and supplemental responses. The staff agreed that the responses were acceptable because safety was not impacted while cooldown time was extended; therefore, RAI-SRP9.2.1-SBPA-05 is resolved.

9.2.1.2.3 Impact of Revised Site Interface Temperature Limits on Cooling Tower Performance

As discussed in TR-108, the applicant proposes to change the site interface temperature limits to encompass a broader range of potential sites for AP1000 plants. In particular, Tier 1 of the DCD, Table 5.0-1, "Site Parameters," is revised to specify a maximum (noncoincident) wet bulb temperature of 29.7 °C (85.5 °F) instead of 27.2 °C (81 °F). Tier 2 of the DCD, Table 2-1, "Site Parameters," is revised to reflect this higher (noncoincident) wet bulb temperature as the Maximum Safety (or 0 percent exceedance) value, and the Maximum Normal (or 1 percent exceedance) wet bulb temperature is revised from 25 °C (77 °F) to 26.7 °C (80.1 °F) for the coincident value and from 26.7 °C (80 °F) to 26.7 °C (80.1 °F) for the noncoincident value. The proposed change to the maximum normal noncoincident value is reflected in the DCD Tier 1 ITAAC that are specified in Table 2.3.8-2 and is referred to in DCD Tier 2, Section 9.2.1.2.3.4, "Plant Cooldown/ Shutdown."

The ITAAC requires COL applicants to demonstrate that each cooling tower cell is capable of dissipating the specified shutdown and spent fuel heat loads at the maximum normal (non-coincident) wet bulb temperature. The proposed change in the ITAAC relative to the wet bulb temperature assumption is consistent with the proposed changes in the site interface temperature limits, and it is acceptable from this perspective. However, the difference between the maximum normal and maximum safety non-coincident wet bulb temperatures was rather trivial, only 0.5 °C (1 °F) before the proposed change; but after the proposed change the gap is widened to 3 °C (5.4 °F). This larger delta between the maximum normal and maximum safety non-coincident wet bulb temperature values warrants further consideration to assure that cooling tower performance is adequate for accomplishing its DID and RTNSS functions. Also, because Tier 2 of the DCD, Sections 5.4.7.1.2.3 and 9.2.2.1.2.1 indicate that cooling tower performance is based upon the maximum safety non-coincident wet bulb temperature as a limiting assumption, it is not clear how this capability is assured. Furthermore, the cooling tower cold water temperature that is specified by the ITAAC is revised from 37.8 °C (100 °F) to 32.2 °C (90 °F) without explanation or justification. This does not appear to be consistent with the supply temperature of 34.2 °C (93.5 °F) that is assumed in Tier 2 of the DCD, Section 9.2.1.1.2. Given these observations, the staff requested In RAI-SRP9.2.1-SBPA-06 that the applicant identify and explain in Tier 2 of the DCD the limiting assumptions and bounding conditions that are important relative to cooling tower design, performance, and operation for assuring that the SWS is capable of and can be relied upon to perform its DID and RTNSS functions, what provisions exist to ensure that these limiting assumptions and bounding conditions will be satisfied by COL applicants over the life of the plant, and what the potential consequences are of exceeding the maximum normal non-coincident wet bulb temperature during operating and shutdown conditions.

The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant explained that the DID functions for the SWS are based on the maximum normal non-coincident wet bulb temperature for the site. The response also indicated that the SWS cooling function is not needed for maintaining the plant in a long-term safe condition. The applicant further indicated that SWS flow and cooling tower performance would be monitored on a continuous basis and that licensees will perform testing at regular intervals to determine cooling tower heat transfer capability, but a COL action item does not exist and one was not established to specify this action by COL applicants. The RAI response did not adequately address RAI-SRP9.2.1-SBPA-06 because more information was needed regarding the DID and RTNSS functional capability of the SWS over the full range of plant operating conditions.

During the June 25, 2009 audit, the staff expressed a concern regarding system reliability for RTNSS and DID over full range of operating conditions. The staff referred to similarities to RAI-SRP9.2.1 SBPA-01 and RAI-SRP9.2.1 SBPA-02 above, discussing what instruments are available to the control room for monitoring system performance.

The staff determined that the revised response to RAI-SRP9.2.1-SBPA-01 above resolves this issue because adequate explanation was provided with respect to RTNSS, cooldown, and DID considerations. Further, the discussion identified that the answer to the instrumentation portion of this concern can be found as part of the response to RAI-SRP9.2.1-SBPA-02. Therefore, RAI-SRP9.2.1-SBPA-06 is resolved.

In APP-GW-GLE-036 , the applicant described the impact of changing the current Maximum Safety wet bulb non-coincident temperature from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F) and the maximum safety wet bulb coincident temperature from 26.7 °C (80 °F) to 30.1 °C (86.1 °F) to

encompass more sites in the eastern United States. In APP-GW-GLE-036, the applicant performed an evaluation of the effect on CCS and SWS to determine if sufficient margin exists to accommodate these changes in Maximum Safety wet bulb temperatures.

The CCS and SWS system provide heat removal from numerous plant loads for normal and abnormal modes of operation. During power operation, the systems are designed to accommodate the Maximum Safety temperature conditions (0 percent exceedance) with a single train in service whereas during shutdown operations the system is designed for the Maximum Normal conditions (1 percent exceedance) with both trains in service.

The applicant concluded that the SWS cooling tower is not expected to require changes to accommodate the higher Maximum Safety wet bulb temperature since the cooling tower sizing case is for plant cooldown at 4 hours after reactor shutdown, and is based on the Maximum Normal 1 percent exceedance value of 26.7 °C (80.1 °F), which is unchanged. The SWS cooling water supply temperature for the Maximum Safety case will be 33.1 °C (91.6 °F), resulting in a maximum CCS supply temperature of 36.1 °C (97.0 °F).

Based on its review of the changes to maximum safety wet bulb temperatures as evaluated by the applicant in APP-GW-GLE-036, the staff finds that the applicant adequately explained the increase in wet bulb temperature limits identified in DCD Tier 1 Table 5.0-1 and Tier 2 Table 2-1 and the applicant demonstrated that the changes in wet bulb temperature limits would not adversely impact performance of the CCS.

9.2.1.2.4 Miscellaneous Changes

DCD Tier 2, Revision 17, incorporates two changes in Section 9.2.1 that are described in TR-103. These changes include provisions for using nonmetallic piping and the removal of "smart valves."

Use of Nonmetallic Piping

The applicant proposed a change to Tier 2 of the DCD, Section 9.2.1.2.2, "Component Description," to allow COL applicants the option of using black polyethylene piping (High Density Polyethylene or HDPE) for SWS applications in accordance with the ASME B31.1 Power Piping Code if deemed appropriate by evaluation. In particular, HDPE could be used in areas of low pressure and low temperature, up to 1,034 kilopascal (kPa) (150 psi) and 60 °C (140 °F). Although the SWS is subject to RTNSS, it is not relied upon for post-72 hour cooling following an accident and the design provisions that pertain to seismic, flooding, and hurricane conditions do not apply. Therefore, from this perspective, the proposed use of HDPE is acceptable. However, since the SWS function is considered to be risk important during shutdown conditions when the reactor is open, the impact of using HDPE on SWS reliability and availability assumptions should be considered and addressed. Also, the review criteria specified by NUREG-0800 Section 3.6.1 relative to pipe failure evaluations is based on the use of metal pipe. Unless otherwise justified by the applicant, the potential consequences of pipe failure (including flooding) should be evaluated assuming the complete failure of all HDPE piping during seismic events coincident with metallic pipe failures that are postulated and other considerations that are specified by NUREG-0800. Finally, the specific criteria for allowing the use of HDPE should be specified in the DCD to ensure clarity of the plant licensing basis. The applicant was asked, in RAI-SRP9.2.1-SBPA-07, to revise the DCD (Tier 1 and Tier 2 as appropriate) to address these considerations.

The applicant responded to the staff's request in a letter dated June 26, 2008, and referred to its earlier response to RAI-TR103-EMB2-02 dated February 22, 2008. Also, additional clarifying information was provided to specify that HDPE will be used for the underground portions of the auxiliary makeup line from the secondary fire water tank and for the underground portions of the SWS blowdown to the circulating water system cooling tower. However, the applicant did not address the specific question that was asked by the staff in RAI-SRP9.2.1-SBPA-07. As a separate matter, the staff also requested the applicant to describe how the requirements specified by 10 CFR 20.1406, "Minimization of Contamination" are satisfied with respect to SWS considerations, including provisions that have been established for buried SWS pipe. Consequently, the RAI response did not adequately address RAI-SRP9.2.1-SBPA-07 because more information was needed regarding the use of nonmetallic piping and its potential effects on RTNSS and systems supporting the ability to achieve cold shutdown operations.

In a letter dated August 31, 2009, the applicant provided additional information specific to the use of HDPE nonmetallic piping in the SWS by proposing to revise DCD Tier 2 Section 9.2.1.2.2, "Piping," to specify that instead of nonmetallic piping, only high density polyethylene piping is used for the underground portions of the auxiliary makeup line from the Secondary Fire Water tank, and for the underground portions of the SWS blowdown line to the CWS cooling tower.

The staff reviewed the applicant's additional information in its response to RAI-SRP-9.2.1-SBPA-07 and finds that it adequately explains the use of nonmetallic piping in the SWS system design. The staff also reviewed the proposed changes to DCD Section 9.2.1.2.2 and finds that they adequately specify that only HDPE is used in the SWS. Therefore, RAI-SRP9.2.1-SBPA-07 was resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Related to 10 CFR 20.1406, in the applicant's submittal of TR-98, "Compliance with 10 CFR 20.1406," APP-GW-GLN-098, dated April 10, 2007, and the applicant's response to RAI-SRP- 12.1-CHPB-01, dated September 9, 2008, all radioactive piping is located inside the auxiliary building which minimizes the potential for leakage to the groundwater from piping and fittings. In addition, no piping containing radioactive fluid is directly buried in the ground. Based on the staff's review, the staff determined 10 CFR 20.1406 has been adequately addressed since the buried SWS does not normally contain radioactive fluids. In the event the SWS contains radioactive fluid, a radiation monitor with a high alarm is provided to monitor the service water blowdown flow. The component cooling water HXs and tower blowdown flow can be isolated by remote manual control. 10 CFR 20.1406 design considerations are further discussed in Chapter 12 of this report.

Removal of Smart Valves

The AP1000 design specifies the use of "smart valves" (i.e., valves that contain instrumentation such as temperature, flow and pressure that is used for control or indication) for some system applications. In the case of the SWS, smart valves (V009 and V011) are specified for the cooling tower makeup and blowdown control valves. The applicant proposed to remove the requirement for using smart valves for these functions to provide flexibility in the design for COL applicants. The proposed change replaces the instrumentation that is included in the smart valve design with standard inline instrumentation as illustrated in Figure 9.2.1-1 of the DCD, Tier 2, and references to valves with internal instrumentation are removed from the description provided in Section 9.2.1.5. This proposed change does not eliminate or alter the functional capabilities of any SWS valves or instruments, and should not degrade the capability or

reliability of the SWS to perform its function. If anything, this less complicated arrangement is expected to improve the capability to service and maintain the affected instrumentation, which would tend to improve SWS availability and reliability consistent with SECY-94-084. Therefore, the staff considers the proposed changes to use standard inline instrumentation to be acceptable.

9.2.1.2.5 Investment Protection Short-Term Availability Controls (IPSAC)

The applicant proposed to change the minimum required flow rate specified by SR 2.4.1 in DCD Tier 2, Table 16.3-2, "Investment Protection Short-Term Availability Controls," to be consistent with the SWS design changes discussed above in Section 9.2.1.1.1. The minimum required flow rate for each SWS pump would be changed from more than 32,555 Lpm (8600 gpm) to more than 37,854 Lpm (10,000 gpm). The proposed change is consistent with the minimum required SWS flow rate discussed in Section 9.2.1.1.1 and from this perspective, it is appropriate. On this basis, the staff finds this acceptable.

9.2.1.2.6 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Considerations

The applicant proposed to change SWS flow rate, heat dissipation capability, and temperature conditions that are specified in ITAAC Table 2.3.8-2 to reflect proposed changes that are discussed above in Sections 9.2.1.1.1 and 9.2.1.1.3. The changes that are proposed for the SWS ITAAC are consistent with the specific changes that are proposed in these sections. On this basis the staff finds this acceptable.

9.2.1.2.7 Initial Test Program Considerations

The initial test program for the SWS is discussed in Tier 2 of the DCD, Section 14.2.9.2.6, "Service Water System Testing." The stated purpose of the SWS test program is to verify the capability of the as-installed system to transfer heat from the CCS HXs to the environment. Prerequisites are specified to assure that the SWS is properly configured and ready for testing, that plant conditions are appropriate, and provisions for collecting data have been established as necessary. SWS performance is observed and recorded during a series of individual component and integrated system tests in order to demonstrate that the SWS properly performs its DID functions. While most of the testing appeared to be appropriate and adequate for demonstrating the SWS DID capabilities, the staff noted that adequate performance for the most limiting situations was not specifically addressed, such as confirming adequate NPSH for the most limiting level, temperature, and flow rate situations and likewise for confirming adequate cooling tower performance; and adequate cooling tower makeup capability does not appear to be verified by the testing that is completed.

In RAI-SRP9.2.1-SBPA-08, the applicant was asked to address these considerations and in particular, to explain how the DID capability of the SWS is assured for the most limiting situations such that system reliability and availability assumptions are valid. The applicant responded to the staff's request in a letter dated June 26, 2008, referring the staff to Tier 2 Section 14.2.9.2.6 of the DCD. The applicant also provided additional information concerning the nature and extent of testing that would be performed. However, much of the information was not reflected in Tier 2 of the DCD, and the stated purpose of the test continues to be very narrowly focused on demonstrating the capability of the CCW HXs to transfer heat to the environment. For example, testing is not specified to confirm that hydraulic transients will not occur; especially following a loss of power and potential drain down of the system. Testing is not specified to demonstrate satisfactory performance during limiting conditions; the cooling

tower makeup capability is not specified and confirmed, etc. Furthermore, the applicant credits COL applicants for performing ongoing surveillance and testing, but there are no COL action items to this effect.

In a letter dated August 31, 2009, the applicant provided additional information specific to the design margins, appended to its original response to RAI-SRP-SBPA-08 to clarify that Section 9.2.1.2.3.6 in Tier 2 of the DCD, actions are taken to prevent drain down and water hammer in the SWS during a loss of normal alternating current (ac) power. The motor-operated SWS tower inlet valves, which are loaded onto the diesel generators (DGs), are automatically closed when power is lost.

The applicant further identified that the diesel-backed SWS pumps undergo their normal automatic start procedure, which is described in detail in Section 9.2.1.2.2 in the last paragraph of the 'Service Water Pumps' section. Since the motor-operated SWS pump discharge valve and the tower inlet valve are both powered by the DG, these valves can stroke open to allow partial flow during pump start, thereby maintaining a water solid system.

Further the applicant stated that Tier 2 Section 14.2.9.2.6 includes testing for the instrumentation, controls, actuation signals and interlocks, as described in item (b) of the 'General Test Acceptance Criteria and Methods' section. As indicated in the first bullet, automatic pump actuation is verified in case an operating pump stops. The testing of the valve interlocks will ensure that they are able to perform their automatic start functions during a loss of power transient.

Service water blowdown is also isolated during a loss of normal AC power condition, in order to reduce liquid loss from the system. The SWS blowdown flow control valve is designed to automatically close when power is lost, which is an interlock that will also be tested in accordance with Section 14.2.9.2.6. This flow control valve is isolated using an ac-powered solenoid, which is fed from a protected, inverter-backed bus.

The staff reviewed the applicant's additional information in its response to RAI-SRP-9.2.1-SBPA-01 and RAI-SRP-9.2.1-SBPA-08 and finds that it adequately explains SWS system operation during normal operation, testing, and operation during loss of normal ac power. Therefore RAI-SRP9.2.1-SBPA-08 is resolved.

9.2.1.3 Conclusions

The staff evaluated proposed changes to Revision 15 of the AP1000 DCD that pertain to the SWS. The proposed changes are documented in Revision 17 of the AP1000 DCD, Tier 2, Section 9.2.1, and are reflected in the Tier 1 ITAAC specified in Table 2.3.8-2 and the Tier 2 IPSAC specified in DCD Table 16.3-2, Section 2.4. The staff's evaluation, using the guidance provided by NUREG-0800 Section 9.2.1, confirmed that: a) the proposed changes will not adversely affect safety-related SSCs, b) the SWS is capable of performing its DID and RTNSS functions, and c) ITAAC, IPSAC, and initial test program considerations are adequate and appropriate. The proposed changes conform to the existing AP1000 licensing basis as documented in Revision 15 of the approved DCD. The proposed changes meet the criteria of; 10 CFR 52.63(a)(1)(iii), on the basis that they reduce unnecessary regulatory burden while maintaining protection to public health and safety and the common defense and security; and 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information in the AP1000 DCD.

9.2.2 Component Cooling Water System

9.2.2.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.2.2, "Component Cooling Water System," of the AP1000 DCD, Revision 15. In the AP1000, Revision 17, the applicant proposed changes to Section 9.2.1.

The applicant proposed the following technical changes to Revision 15 of the AP1000 DCD, which are supported by information in the TRs:

1. Proposed changes to the CCS in order to accommodate increased heat loads. This includes the CCS pump design capacity which was changed from 33900 Lpm (8960 gpm) to 35900 Lpm (9500 gpm).

2. The applicant changed the maximum CCS supply temperature to plant components from 35 °C (95 °F) to 37.8 °C (100 °F). Additionally, the applicant raised the wet bulb temperature for service water cooling at normal operations (maximum normal temperature per Table 2-1 for normal shutdown). The ambient design wet bulb temperature was raised from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F).

3. The applicant changed the RCPs component cooling water discharge isolation valves from MOV to air operated valves (AOV). The applicant changed the closing signal input to these valves to an AOV close signal generated by the plant control system. The close signal will use simultaneous CCS flow deviations in the RCP's CCS supply and return lines. The applicant deleted reference to a tube rupture in the RCP motor cooling coil or thermal barrier as a source of reactor coolant leakage into the CCS and replaced this source of potential leakage with the RCP external HX. These changes conform to the applicant's RCPs design change to canned RCPs, which includes a new RCP external HX that replaces the RCP motor cooling coils and thermal barriers what were cooled by CCS.

4. The applicant described changes to the flow sensors in the CCS inlet and outlet lines associated with each RCP external HX. The proposed change clarified that isolation valves used to isolate the component cooling water outlet line associated with its RCP are nonsafety-related. The changes associated with a canned RCP include the installation of a RCP external HX to replace the pump motor cooling coil and thermal barrier. The applicant identified that other plant alarms and indications can be used by the operator to manually initiate a RCP component cooling water system isolation.

5. The applicant provided additional detail for instrumentation for high-level and low-level alarms on the CCS surge tank, automatic actuation of the CCS surge tank makeup water valve for makeup flow from the demineralized water transfer and storage system into the CCS, and flow alarms in the MCR to indicate that a leak exists on the RCP external HX. The RCP external HX replaces the RCP motor cooling coils and thermal barriers and is cooled by CCS. In addition, the applicant identified that flow measuring instrumentation on the RCP component cooling water inlet and outlet lines provide an isolation signal to close an AOV and isolate the affected RCP external HX from the rest of the CCS.

6. The applicant corrected the referenced code to use for nonmetallic piping from "ASME B31.1" to "ANSI B31.1." The proposed change also clarified that ANSI B31.1 Appendix III also may be used for outside containment piping.

Details of these proposed changes are discussed in TR-111, TR-108 and TR-103, respectively.

9.2.2.2 Evaluation

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 DID capability of removing reactor and spent fuel decay heat. It is also 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 SECY-94-084. 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 DID 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 by NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was judged based upon conformance with the existing AP1000 licensing basis, the guidance specified by NUREG-0800 Section 9.2.2 (as applicable), and SECY-94-084.

Modifications to approved standard plant DCs can be proposed provided (among other things) that the changes are deemed to be necessary in accordance with 10 CFR 52.63(a)(1). The proposed changes will allow additional flexibility for COL applicants, thereby reducing the need for departure requests. The specific criteria that apply to the proposed changes referred to above include: 10 CFR 52.63(a)(1)(iii), which concerns reducing unnecessary regulatory burden while maintaining protection to public health and safety and the common defense and security; and 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

During its evaluation, the staff noted that the description that is provided in Tier 2 of the AP1000 DCD, Section 9.2.2, "Component Cooling Water System," does not describe the DID, investment protection, and RTNSS design basis for the CCS. However, it is clear from the ITAAC specified in Tier 1 of the DCD, Section 2.3.1, "Component Cooling Water System;"," the initial test program described in Tier 2 of the DCD, Section 14.2.9.2.5, "Component Cooling Water System Testing," Table 16.3-2, "Investment Protection Short-Term Availability Controls," as it pertains to CCS, and Table 17.4-1, "Risk-Significant SSCs Within the Scope of D-RAP," that the CCS is important for both DID, investment protection, and RTNSS considerations. However, this information has not been adequately reflected in the description that is provided for the CCS in Tier 2 of the DCD, Section 9.2.2. In RAI-SRP9.2.2-SBPA-03 the applicant was asked to include additional information in Section 9.2.2 to better explain the DID, investment protection, and RTNSS design basis for the CCS (also see test considerations referred to below in Section 9.2.2.2.7). The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant indicated that the information provided for the CCS is similar to what was provided for the SWS and other DID systems. The staff confirmed that the Tier 2 information for the CCS is similar to what was provided for other systems of this nature, with no description of the DID, investment protection, or RTNSS design basis for these systems.

During the June 25, 2009 audit, the staff explained that, in order to resolve the RAI, the applicant should provide more detail regarding the CCS RTNSS, Design Reliability Assurance Program (D-RAP), and IPSAC functions. In a letter dated August 31, 2009, the applicant provided additional information about the CCS functions by proposing to revise DCD Tier 2 Section 9.2.2 referring to DCD Tier 2 Section 17.4-1.

The staff reviewed the applicant's additional information in its response to RAI-SRP-9.2.2-SBPA-03 and finds that it adequately explains the CCS D-RAP functions by referencing DCD Tier 2 Section 17.4. The staff also reviewed the proposed changes to DCD Section 9.2.2.3.1 and finds that they adequately referenced the basis for including the CCS components within the scope of D-RAP. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.2.2.2.1 CCS Design Changes Required to Accommodate Increased Heat Loads

As discussed in TR-111, CCS modifications are necessary as a result of higher heat loads and flow rates for cooling the RCPs and the condensate pump oil coolers, adding the VFD for the RCPs as new CCS heat loads and relocating them from the northwest to the southwest side of the turbine building, and excessive CCS flow velocities at the CCS pump suction and discharge headers and inside containment due to the increased CCS flow demand that is necessary to satisfy increased component heat loads. Consequently, the applicant implemented CCS design changes to add new components, reconfigure the CCS piping layout and resize pipe as necessary, revise CCS pump and HX parameters, and increase the CCS design pressure. The CCS changes are reflected in the AP1000 DCD, Tier 1, Table 2.3.1-2, ITAAC Design Commitment, Item 3, relative to the capability to support plant shutdown and spent fuel cooling; Tier 2, Section 9.2.2, "Component Cooling Water System," including Table 9.2.2-1, "Nominal Component Data – Component Cooling Water System," Table 16.3-2, "Investment Protection Short-Term Availability Controls," Section 2.3, " Component Cooling Water System (CCS) – RCS Open," SR 2.3.1 with respect to the minimum required CCS flow rate.

Relocating the VFDs to the southern end of the turbine building places them in close proximity to the CCS pumps and HXs. Failures associated with the VFDs could affect the capability of the CCS to perform its RTNSS function and additional information is needed to address this consideration. In RAI-SRP9.2.2-SBPA-04, additional information was requested to address relocating the VFDs. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant stated that typical failures expected for high power electronic equipment include fires and in this case loss of cooling water from the dedicated cooling system or from the CCS which supplies cooling water to the VFD internal cooling system HXs. Fires in the turbine building, caused specifically by a failure of VFD equipment, that disable both CCS pumps have been addressed by the inclusion of a means to provide 2271 Lpm (600 gpm) of cooling water to normal RNS HX 'A' from the FPS to provide continued capability to remove decay heat from the RCS following suppression of the fire. During suppression activities, the plant passive safety systems ensure that decay heat is removed from the core and therefore cooling of the RNS HXs with CCS is not required. SFS pool cooling is also provided by other means during this period of time. These provisions are described in DCD Revision 16, Tier 2, Sections 9.1.3.4.3, "Abnormal Conditions," and 9.2.2.4.5.5. In addition, a break in the VFD internal cooling water lines or in the CCS lines supplying the HXs does not increase the risk of a flooding event, as a break of this size is enveloped by the bounding flooding case of breaks in larger CCS and SWS lines in the southern end of the turbine building.

The staff reviewed this response and determined that it did not address RTNSS considerations. In a revised RAI response dated April 13, 2009, the applicant stated that the variable frequency drive is used only during heatup and cooldown when the reactor trip breakers are open. During power operations, the drive is isolated and the RCP is run at constant speed; therefore, the VFDs are de-energized during suppression activities.

The staff reviewed the revised RAI response and determined that RTNSS considerations were adequately addressed since the flooding event is bounded by the larger CCS and SWS line breaks in the turbine building and the VFDs would not be energized during the period of time the CCS would be performing their RTNSS function in Mode 5 and 6. Therefore, RAI-SRP9.2.2-SBPA-04 is considered resolved.

In DCD Revision 16, the applicant reanalyzed the fluid pressures throughout the redesigned CCS and determined that the design pressure of the system should be increased from 1,034 kPa (150 pounds per square inch gauge (psig)) to 1,379 kPa (200 psig). However, the total design differential head of the CCS pumps is actually reduced substantially and it was not clear why the system pressure was increased. The staff asked the applicant, in RAI-SRP9.2.2-SBPA-05, to address this inconsistency. In response to RAI-SRP9.2.2-SBPA-05, the applicant referred to industry operating experience showing that relief valve actuations occur frequently during routine realignments (e.g., pump swaps) in CCS that are designed for 1,034 kPa (150 psig). Also, based upon the results of a hydraulic analysis that was performed, the applicant determined that the CCS operating pressure for AP1000 is just below the relief valve setpoint for a system design pressure of 1,034 kPa (150 psig). Consequently, the applicant increased the CCS design pressure to 1,379 kPa (200 psig) in order to minimize the occurrence of relief valve actuations and the likelihood of valve leakage. Based on the information that was provided, the staff finds that the proposed increase in the CCS design pressure will increase the available margin and make the system more robust. Additionally, because the higher relief valve set point will tend to minimize spurious actuations and valve leakage, the proposed increase in CCS design pressure reduces the likelihood of spreading radioactive contamination consistent with 10 CFR 20.1406 requirements. Therefore, the staff considers the proposed increase in CCS design pressure to be acceptable and RAI-SRP9.2.2-SBPA-05 was resolved.

Proposed Increases in CCS Flow Rate and Heat Removal Capability

The applicant proposed to increase the minimum required CCS flow rate and heat removal capability in order to accommodate the design changes referred to above. The values listed in Tier 2 of the DCD, Section 9.2.2 and Table 9.2.2-1, and the ITAAC specified by Tier 1 of the DCD, Table 2.3.1-2, reflect these changes.

The ITAAC specified in Tier 1 of the DCD, Table 2.3.1-2, require COL applicants to demonstrate that the CCS design is capable of supporting plant shutdown and spent fuel cooling. The applicant proposes to change the ITAAC acceptance criteria to demonstrate a flow rate for each CCS pump of at least 10,164 Lpm (2685 gpm) to one normal shutdown cooling HX (this is unchanged), plus 4542 Lpm (1200 gpm) to one SFP HX (this is increased by 284 Lpm (75 gpm) from the previous amount), and at least 16,713 Lpm (4415 gpm) to other CCS heat loads (this is increased by 12,397 Lpm (3275 gpm) from the previous amount), for a total required flow rate for each CCS pump of 31,419 Lpm (8300 gpm). This represents an increase in the total required flow rate for each CCS pump of 12,681 Lpm (3350 gpm). The total CCS pump flow rate that is specified for each pump is consistent with the proposed value that is listed in Tier 2 Table 9.2.2-1, and it is acceptable from this perspective. However, Tier 2 of the DCD,

Section 9.2.2, does not identify what the minimum CCS flow requirements are for these three categories of heat loads that are listed in the ITAAC, how much excess margin is available for each one, the basis for this determination, and how the specified flow balance will be maintained over time. In RAI-SRP9.2.2-SBPA-06, the staff requested that the applicant address this missing information.

The applicant responded to RAI-SRP9.2.2-SBPA-06 in a letter dated June 26, 2008. The applicant provided additional information primarily related to CCS HX design and made reference to TR-111 for additional discussion. However, the applicant's response did not address the specific question that was asked by the staff.

During the June 25, 2009 audit, the staff explained that, in order to resolve the RAI, the applicant should provide more detail concerning how the ITAAC demonstrate adequate flow for RTNSS.

In a letter dated August 31, 2009, the applicant stated that CCS system flow analysis is performed to demonstrate that the selected CCS pump head and flow characteristics ensure delivery of the required flow to all CCS users and also verify that the flow balancing orifices are sized with margin to be adjusted in the field. Also, an additional 7 percent margin in head is added to the CCS pump curve developed from the flow analysis, specifically to offset the effects of any degradation of pump performance occurring during the life of the plant. However, changes in CCS flow performance over time are expected to be minimal, since the CCS is a closed-loop, chemically-treated system with orifices used for flow balancing.

The applicant also stated that the CCS ITAAC requires a minimum flow rate of 10,164 Lpm (2685 gpm) to transfer heat from the RNS during shutdown. This flow rate, which assumes 10 percent degradation from the normal RNS HX flow requirement, is the minimum flow rate needed to remove decay heat from the RNS when it is aligned 4 hours after reactor shutdown (Mode 4). This flow rate must also be verified one day before entering Modes 5 and 6, as a SR included in CCS IPSAC, Table 16.3-2. Since a flow rate of 10,164 Lpm (2685 gpm) is sufficient to remove decay heat during Mode 4, it is also bounding for decay heat removal during Modes 5 and 6, when the RCS decay heat level has been further reduced. As a result, this SR will ensure that the CCS will be able to adequately perform its RTNSS function. This CCS minimum flow rate of 10,164 Lpm (2685 gpm) to the RNS HX is added to Tier 2 Table 9.2.2 1 in the DCD markup, as well as a similar 10 percent degraded value of 4543 Lpm (1200 gpm) to the SFS HX.

The staff reviewed the applicant's additional information in its response to RAI-SRP9.2.2-SBPA-06 and finds that it adequately explains the CCS basis for flow required for its RTNSS functions with design adequate margins at 10 percent degraded values. The staff also reviewed the proposed changes to DCD Table 9.2.2-1 and finds that the table, as changed, adequately describes the basis for CCS flow rates. 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 proposes to increase the heat transfer capability of each CCS HX as specified in ITAAC Table 2.3.1-2. The current acceptance criterion specifies a UA value of 740×10^6 W/°C (12.1×10^6 Btu/hr-°F), and the applicant proposes to change this UA value to 856×10^6 W/°C (14.0×10^6 Btu/hr-°F). The value that is proposed as the ITAAC acceptance criterion is consistent with the revised value that is proposed in Tier 2, Table 9.2.2-1, and it is acceptable from this perspective. However, the applicant did not identify how the proposed CCS HX UA value was determined and how much margin is available to address operational

considerations, on what basis this determination is appropriate and justified, and how the specified CCS heat transfer capability will be maintained over time. In RAI-SRP9.2.2-SBPA-07 the staff asked the applicant to address these heat transfer issues.

The applicant responded to RAI-SRP9.2.2-SBPA-07 in a letter dated June 26, 2008. The applicant provided additional information primarily related to CCS HX design and made reference to TR-111 for additional discussion. However, the response did not address the specific question that was asked.

During the June 25, 2009 audit, the staff explained that in order to resolve the RAI the applicant should provide more detail explaining how the proposed CCS HX coefficient of heat transfer (U) and required heat-transfer area (A) values were determined and how much margin is available to address operational considerations.

In a letter dated August 31, 2009, the applicant stated that the CCS HX UA was established to ensure that supply temperature did not exceed the RCP external HX cooling requirements, under maximum safety wet bulb temperatures. The design UA also bounds the UA value needed to meet the CCS temperature requirements for cooling down the RCS to cold shutdown conditions within 96 hours of reactor shutdown. Selecting a CCS HX UA, which meets temperature requirements during plant cooldown also ensures that the CCS HXs will be able to perform their DID and RTNSS functions of providing cooling to the RNS HXs during RCS cooldown and reduced reactor coolant inventory operation.

The staff reviewed the applicant's additional information in its response to RAI-SRP9.2.2-SBPA-07 and finds that it adequately explains how the CCS HX UA values were determined and how much margin is available to address operational considerations. The CCS HX specification requires the inclusion of additional heat transfer area above the design value to account for fouling and degradation over the HX's operating life. Additional frame length is also included so that additional plates can be added to the HX if additional performance is required. The HX margins are also discussed in Section 9.2.1 of this report and have been adequately addressed. Therefore, RAI-SRP9.2.2-SBPA-07 is resolved.

CCS Pump Design Considerations

Tier 2 Table 9.2.2-1 included additional proposed changes that had not been explained and justified. In particular, the bases for the proposed changes to the CCS pump design capacity and total developed head had not been addressed. Also, the bases for the proposed changes to the CCS HX design duty, design UA, and design flow rate (CCS side) had not been addressed. In RAI-SRP9.2.2-SBPA-08, the staff asked the applicant to address these heat transfer issues.

The applicant responded to RAI-SRP9.2.2-SBPA-08 in a letter dated June 26, 2008. The applicant indicated that the increased CCS pump design capacity is primarily due to increased cooling water flow requirements for the RCPs. The total developed head requirement for the CCS pumps was reduced substantially by increasing the diameter of several of the CCS main supply and return headers to minimize dynamic losses in the system that would otherwise result from the increase in CCS flow rate. The staff agrees that these particular changes are appropriate and justified for the reasons stated. However, the applicant's response did not adequately address and justify the proposed changes to the CCS HX design parameters.

During the June 25, 2009 audit, the applicant stated that the bases of the proposed CCS changes were discussed in TR-111. Further, the applicant clarified that other CCS changes (including lower CCS pump TDH, increasing piping sizes, and the reduction in flow velocities), were provided in TR-111.

The staff finds, based on the review of TR-111, that the UA of the CCS HX has been increased to meet all the CCS performance requirements with the increased heat loads for cooled components. The increase in CCS HX size is associated with an increased SWS flow rate as well as an increased CCS flow rate. On this basis the staff finds that RAI-SRP9.2.2-SBPA-08 is resolved.

CCS Cooling for RCPs, Instrumentation and Controls

In DCD Section 9.2.2.3.4, the applicant proposed to change the RCP component cooling water discharge isolation valves from MOVs to AOVs. The applicant changed the closing signal input to these valves to an AOV close signal generated by the plant control system. The close signal will use simultaneous CCS flow deviations in the RCP's component cooling water supply and return lines. The applicant deleted reference to a tube rupture in the RCP motor cooling coil or thermal barrier as a source of reactor coolant leakage into the CCS and replaced this source of potential leakage with the RCP external HX. The staff finds that the proposed change does not eliminate the requirement that the RCP's CCS outlet line be protected from overpressure by relief valves.

In DCD Section 9.2.2.4.5.2, the applicant further described the flow sensors in the CCS inlet and outlet lines associated with each RCP external HX and added that the cooling water outlet line isolation valves on each RCP are nonsafety-related. With the design change to a canned RCP, the applicant replaced the pump motor cooling coil and thermal barrier with an RCP external HX. The applicant also modified how the RCP component cooling water isolation signal is developed and further clarified the alarms an operator would receive.

In DCD Section 9.2.2.7, the applicant provided detail for the high-level and low-level alarms instrumentation on the CCS surge tank. There are two redundant level channels in the design to reduce the likelihood of reactor trip caused by a single downscale failure of a surge tank level channel. Such redundancy could preclude unnecessary tripping of CCS pumps which would subsequently cause loss of cooling flow to the RCPs and other cooled components. The CCS surge tank makeup water valve is automatically actuated by one of the two level channels, in order to provide makeup flow from the demineralized water transfer and storage system into the CCS. The applicant clarified that flow alarms in the MCR, produced by the two flow channels located on the CCS RCP cooling water inlet and outlet lines, will be used to alert the operator that a leak exists on the RCP external HX. The applicant identified that flow-measuring instrumentation on the RCP component cooling water inlet and outlet lines provides an isolation signal to close an AOV and isolates the leaking RCP external HX from the rest of the CCS.

The staff finds that changes to CCS valves, flow sensors, isolation signals and instrumentation provide an additional level of system reliability and do not result in negative or adverse system interactions. Based on its evaluation, the staff concludes that these valve and instrumentation changes are acceptable and do not change the NUREG-1793 Section 9.2.2 findings or conclusions.

9.2.2.2.2 Proposed Increase in the Maximum CCS Supply Temperature

Tier 2 of the AP1000 DCD, Section 9.2.2.1.2.1, "Normal Operation," proposes to increase the maximum allowed CCS supply temperature to plant components from 35 °C (95 °F) to 37.2 °C (99 °F) during normal plant operations, but the basis for this proposed change was not explained and justified. In RAI-SRP9.2.2-SBPA-09 the staff requested that the applicant justify this change.

The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant explained that the increased CCS supply temperature was due to the proposed increase in the maximum safety non-coincident wet bulb temperature from 27.2 °C (81 °F) to 29.7 °C (85.5 °F) using a cooling tower that is designed for an approach to wet bulb temperature of 13.3 °C (8 °F). The applicant referenced page 11 of TR-108 for additional explanation. The applicant also provided information regarding a further increase in the maximum safety non-coincident wet bulb temperature that was being made (but not yet submitted) to accommodate the Levy site parameters. This information was not included within the scope of this evaluation. While the information that was provided by the applicant explains to some extent how the maximum service water supply temperature is achieved, it did not explain how the maximum CCS supply temperature was determined and justified. The staff also noted that the use of non-conservative temperature assumptions for the plant shutdown and refueling heat transfer analyses was not explained and justified. Furthermore, this approach was not consistent with the information provided in Tier 2 Section 5.4.7.1.2.3, "In-Containment Refueling Water Storage Tank Cooling," which indicates that the maximum safety non-coincident wet bulb temperature is assumed for normal conditions and transients that start at normal conditions.

During the June 25, 2009 audit, the applicant clarified that the normal wet bulb is a realistic value for evaluating DID on investment protection as stated in its May 13, 2009, supplemental response to RAI-SRP9.2.1-SBPA-05. The maximum safety wet bulb temperature of 30.0 °C (86.1 °F) is applicable for full power operations as stated earlier in the May 7, 2009, and May 13, 2009, supplemental RAI responses. The higher ambient temperatures 30.0 °C vs. 26.7 °C (86.1 °F vs. 80.1 °F) will not impact safety or investment protection and would only result in an extended time to achieve cooldown. The staff agreed that the information was previously adequately presented; therefore, RAI-SRP9.2.2-SBPA-09 is resolved. Wet bulb temperature considerations are also discussed in Section 9.2.1 of this report and have been resolved.

In DCD Section 9.2.2.1.2.1 Revision 17, the applicant proposed a change in the maximum CCS supply temperature to plant components from 37.2 °C (99 °F) to 37.8 °C (100 °F). Additionally, the applicant proposed to raise the wet bulb temperature for service water cooling at normal operations (maximum normal temperature per Tier 2 Table 2-1 for normal shutdown). Although the assumption of a 0 percent exceedance was not changed, the ambient design wet bulb temperature would be raised from 29.7 °C (85.5 °F) to 30.0 °C (86.1 °F).

In APP-GW-GLE-036, the applicant described the impact of changing the current maximum wet bulb non-coincident temperature from 29.7 °C (85.5 °F) to 30.0 °C (86.1 °F) and the maximum wet bulb coincident temperature from 26.7 °C (80 °F) to 30.0 °C (86.1 °F) to encompass more sites in the eastern United States. In APP-GW-GLE-036, the applicant performed an evaluation of the effect on CCS to determine if sufficient margin exists to accommodate a 3.3 °C (6.1 °F) change and a 0.3 °C (0.6 °F) for both coincident and non-coincident wet bulb temperatures:

The applicant performed a design assessment and identified the following areas associated with CCS that are affected by the increased maximum wet bulb temperature:

- Safety system design basis – additional cases for containment analysis were included in the safety analysis to support the revised coincident and non-coincident wet bulb temperature.

- Normal, decay and SFP heat removal (cases relying on use of the 0 percent exceedance wet bulb temperature only)

- Component cooling and service water design

Safety System Design Basis

The applicant stated that no changes to the AP1000 design are needed to accommodate any safety issues because evaluations have demonstrated that the current AP1000 accident analyses will bound the revised coincident 46.1°C/30.0 °C (115 °F/86.1 °F) and non-coincident 30.0 °C (86.1 °F) wet bulb temperatures. The applicant stated that the maximum containment peak pressure performance of the passive containment cooling system at the higher wet bulb temperature is bounded by the current analysis for which a bounding sensitivity was documented in the Nuclear Safety Containment Analysis for AP1000.

Normal, Decay and Spent Fuel Pool Heat Removal

The applicant evaluated the impact of wet bulb temperature change on the performance of the normal RNS, SFP cooling, CCS and SWSs. The applicant stated that the performance evaluations considered normal operating modes, the ability to meet post shutdown cooldown times, full core offloads, loss of all ac power, and heat up of the IRWST.

The applicant stated that all design criteria were met including cooldown times and temperature limits with the exception of normal plant power operation with maximum heat loads, one CCS train in service, and at the maximum safety temperature limit of 30.0 °C (86.1 °F) wet bulb. The applicant identified that the exception exists for less than 30 hours per year and that with only one train of CCS, the CCS temperature would rise above 35.0 °C (95 °F) and then return to less than 35.0 °C (95 °F) by the time the 1 percent exceedance temperature was reached. The applicant identified the RCP motor cooling system as the most limiting component, which was designed to operate for at least 6 hours duration with a temperature up to 37.8 °C (100 °F). The applicant's evaluation was that with the maximum allowable cooling water temperature of 36.1 °C (97 °F) for the RCPs (the most limiting component), this change was acceptable.

Component Cooling (CCS) and Service Water System (SWS) design

The applicant stated that CCS will accommodate the heat loads from operations without impacting performance or sizing of the CCS. The SWS cooling tower is not expected to require changes to accommodate the higher wet bulb temperatures since sizing is based on plant cooldown at 4 hours after reactor shutdown, and is based on the unchanged 1 percent exceedance value of 26.7 °C (80.1 °F). The applicant explained that the SWS cooling water supply temperature for the maximum safety case will be 33.1 °C (91.6 °F), which will result in a maximum CCS supply temperature of 36.1 °C (97.0 °F).

In APP-GW-GLE-036, the applicant states that the most limiting CCS component is the RCP motor cooling system and that temperatures of up to 37.8 °C (100 °F) for a duration of 6 hours are acceptable. In DCD Revision 17 Table 5.4-1, the RCP maximum continuous component cooling water inlet temperature is given as 35 °C (95 °F) with a 6 hour elevated temperature of up to 43.3 °C (110 °F). Additionally the applicant identified in the DCD that an input to a reactor trip is RCP "Hi Bearing Temperature" but there was no mention of high motor temperature. As a result of the design change to canned RCPs, the staff asked the applicant, in RAI-SRP9.2.2-SBPA-14, to verify that the RCP motor cooling system is still the most limiting CCS supply temperature: if the RCP motor cooling system is no longer the most limiting CCS cooled component, identify and provide the evaluation of the impacts of the revised wet bulb temperature limit on the plant for the new limiting component. The staff also asked the applicant, in RAI-SRP9.2.2-SBPA-14, to clarify a statement in APP-GW-GLE-036.

The applicant responded to RAI-SRP9.2.2-SBPA-14 on August 31, 2009 and stated that the RCP motor cooling system is still the most limiting component served by the CCS with respect to maximum temperature of the supplied cooling water. The limiting CCS supply temperature for RCP cooling is 37.7 °C (100 °F). The RCP can operate at full speed with CCS supply temperature at this level for up to 6 hours continuously. Since the CCS and SWS are both designed with significant thermal margin, the actual CCS supply temperature to the RCPs and to other CCS components with the plant at power will always be lower than the limiting value of 37.7 °C (100 °F), which assumes maximum operating heat load on the CCS, 4 °C (8 °F) cooling tower approach to wet bulb, and local ambient wet bulb temperature at the maximum safety (0 percent exeedence) level. During plant cooldown with RCPs operating, the CCS temperature may approach 37.7 °C (100 °F) for a few hours at the highest plant cooldown rate of 27.8 °C/h (50 °F/h), but the RCPs are operating at reduced speed in this mode and their cooling requirements are therefore less stringent than for full power, full speed operation.

The staff review determined that this response is acceptable since the design conditions of 37.7 °C (100 °F) have been established in DCD Section 9.2.2.1.2.1, "Normal Operations". During the cooldown period, the component cooling water inlet temperature to the various components does not exceed 43.33 °C (110 °F) as described in DCD Section 9.2.2.4.3, "Plant Shutdown" which is consistent with Table 5.4-1. For this reason, the staff determined RAI-SRP9.2.2-SBPA-14 is resolved.

9.2.2.2.3 Revised Site Interface Temperature Limits

As discussed in TR-108, the applicant proposes to change the site interface temperature limits to encompass a broader range of potential sites for AP1000 plants. In particular, Tier 1 of the DCD, Table 5.0-1, "Site Parameters," would be revised to specify a maximum (noncoincident) wet bulb temperature of 29.7 °C (85.5 °F) instead of 27.2 °C (81 °F). Tier 2 of the DCD, Table 2-1, "Site Parameters," would be revised to reflect this higher (noncoincident) wet bulb temperature as the Maximum Safety (or 0 percent exceedance) value, and the Maximum Normal (or 1 percent exceedance) wet bulb temperature would be revised from 25 °C (77 °F) to 26.7 °C (80.1 °F) for the coincident value and from 26.7 °C (80 °F) to 26.7 °C (80.1 °F) for the noncoincident value.

The proposed changes to the site interface temperature limits are reflected in Tier 2 of the AP1000 DCD, Section 9.2.2, in place of the values that were previously listed. Although the values correspond to how they were used previously (i.e., the replaced values are "like-for-like"), the Tier 2 description does not explain why the maximum safety (noncoincident) wet bulb temperature is specified for normal operation and the maximum normal wet bulb temperature is

specified for other cases. It is not clear why the maximum safety limit does not apply for the CCS DID and RTNSS functions. The staff asked the applicant, in RAI-SRP9.2.2-SBPA-10, to explain and justify this approach and to revise the Tier 2 information to clearly describe the plant design basis in this regard.

The applicant responded to RAI-SRP9.2.2-SPBA-10 in a letter dated June 26, 2008. The applicant explained that the maximum safety non-coincident wet bulb temperature does not apply to RTNSS and Investment Protection functions because they are not functions required to guarantee the safety of the plant. However, contrary to this logic, the applicant also explained that the maximum safety non-coincident wet bulb temperature is used in determining CCS and SWS performance for power operation since the peak ambient wet bulb temperature has a relatively high likelihood of occurrence during the operating portion of a refueling cycle. The applicant failed to recognize that elevated temperature conditions tend to increase the likelihood of plant trips and transients due to grid instability and assurance needs to be provided that DID and RTNSS SSCs are capable of performing their functions whenever the maximum normal wet bulb temperature is exceeded. The applicant's response did not address the staff's concerns in this regard.

During the June 25, 2009 audit, the applicant clarified that, as stated in its response to RAI-SRP9.2.1-SBPA-05, with respect to the 26.7 °C (80.1 °F) wet bulb for RTNSS, normal cooldown can be accomplished sooner than it can be accomplished with the 30.0 °C (86.1 °F) wet bulb. Further, the applicant clarified that all the temperature limits have margins. The applicant explained that passive safety systems are not needed. Further, the applicant clarified that the normal wet bulb temperature is a realistic value for evaluating DID on investment protection. The maximum safety wet bulb temperature of 30.0 °C (86.1 °F) is applicable for full power operations, as previously presented in the RAI-SRP9.2.2-SBPA-09 response dated May 7, 2009.

The staff agreed that the information was previously adequately presented; therefore, RAI-SRP9.2.2-SBPA-10 is resolved. The higher ambient temperatures 30.0 °C vs. 26.7 °C (86.1 °F vs. 80.1 °F) will not impact safety or investment protection and would only result in an extended time to achieve cooldown.

9.2.2.2.4 Use of Nonmetallic Pipe

The applicant proposed a change to Tier 2 of the DCD, Section 9.2.2.3.5, "Piping Requirements," to allow COL applicants the option of using black polyethylene piping (High Density Polyethylene or HDPE) for CCS applications in accordance with the ASME B31.1 Power Piping Code if deemed appropriate by evaluation. In particular, HDPE could be used in areas of low pressure and low temperature, up to 1,034 kPa (150 psi) and 60 °C (140 °F). The basis for the use of nonmetallic pipe for this application is described in TR-103.

Although the CCS is subject to RTNSS, it is not relied upon for post-72 hour cooling following an accident and the design provisions that pertain to seismic, flooding, and hurricane conditions do not apply. Therefore, from this perspective, the proposed use of HDPE is acceptable. However, since the CCS function is risk important during shutdown conditions when the reactor is open, the impact of using HDPE on CCS reliability and availability assumptions needs to be considered and addressed. Also, the review criteria specified by NUREG-0800 Section 3.6.1 relative to pipe failure evaluations is based on the use of metal pipe. Unless otherwise justified by the applicant, the potential consequences of pipe failure (including flooding) should be evaluated assuming the complete failure of all HDPE piping during seismic events coincident

with metallic pipe failures that are postulated and other considerations that are specified by the SRP. Finally, the specific criteria for allowing the use of HDPE should be specified in the DCD to ensure clarity of the plant licensing basis. The applicant was asked, in RAI-SRP9.2.2-SBPA-11, to revise the DCD (Tier 1 and Tier 2 as appropriate) to address these considerations. The applicant responded to the staff's request in a letter dated June 26, 2008, and referred to its earlier response to RAI-TR103-EMB2-02 dated February 22, 2008. The applicant also indicated that HDPE is not used in the AP1000 CCS design and that there are no current plans to use HDPE in this system. However, because use of HDPE is proposed as an option for COL applicants, its use needed to be fully evaluated and justified by the applicant. The response that was provided by the applicant did not provide the information that was requested in RAI-SRP9.2.2-SBPA-11. As a separate matter, the staff also requested that the applicant describe how the requirements specified by 10 CFR 20.1406 are satisfied with respect to CCS considerations, including provisions that have been established for buried or inaccessible pipe.

During the June 25, 2009 audit, the staff explained that in order to resolve the RAI the applicant should provide more detail explaining the use of nonmetallic piping in the CCS.

In a letter dated August 31, 2009, the applicant stated that the provision for the use of nonmetallic piping is removed from DCD Tier 2 Section 9.2.2. The operating pressure and temperature for the CCS exceeds the limits for HDPE piping imposed by ANSI/ASME B31.1 and applicable code cases and is, therefore, prohibited for use in this application. Additionally, the applicant proposed to revise DCD Tier 2 Section 9.2.2.3.5 "Piping Requirements" to delete references to nonmetallic piping. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Related to 10 CFR 20.1406, in the applicant's submittal of APP-GW-GLN-098 dated April 10, 2007 and the applicant response to RAI-SRP-12.1-CHPB-01 dated September 9, 2008 all radioactive piping is located inside the auxiliary building, which minimizes the potential for leakage to the groundwater from piping and fittings. In addition, no piping containing radioactive fluid is directly buried in the ground. In addition, the use of embedded pipes is minimized to the extent possible, consistent with maintaining radiation doses ALARA as described in DCD Section 12.3.1.1.2, "Common Facility and Layout Designs of ALARA". To the extent possible, pipes are routed in accessible areas such as dedicated pipe routing tunnels or pipe trenches; this provides good conditions for decommissioning. Based on the staff's review, the staff determined 10 CFR 20.1406 has been adequately addressed since the CCS is not buried and radiation monitors, which monitor RCS leakage into CCS, alarm in the MCR. 10 CFR 20.1406 design considerations are further discussed in Chapter 12 of this report.

In DCD Section 9.2.2.3.5 Revision 17, the applicant corrected the code reference applicable to nonmetallic piping. The applicant changed the specification for nonmetallic piping from "used in accordance with ASME B31.1" to "constructed to the requirements of ANSI B31.1 Appendix III" and limited the use of nonmetallic piping to outside containment for the CCS system.

Based on its evaluation, the staff finds that this change limits the use of nonmetallic piping to areas outside containment which are outside the risk-important portions of the CCS function during shutdown conditions. Based on its evaluation, the staff concludes that these changes are acceptable and do not change the NUREG-1793 Section 9.2.2 findings or conclusions related to CCS piping requirements.

9.2.2.2.5 Investment Protection Short-Term Availability Controls

The applicant proposed to change the minimum required CCS flow rate that is specified for the normal shutdown cooling HX in DCD Tier 2, Table 16.3-2, "Investment Protection Short-Term Availability Controls," SR 2.3.1. This SR is revised to specify that each CCS pump needs to provide at least 10,164 Lpm (2685 gpm) through a normal shutdown cooling HX, which is consistent with the flow rate specified in ITAAC Table 2.3.1-2 for Design Commitment 3 (it is noted that a change is not being proposed for the ITAAC value that was originally established). However, SR 2.3.1 previously specified a minimum flow rate of 10,675 Lpm (2820 gpm), and it was not clear why the ITAAC value that was established was not the same as the value that was originally specified by SR 2.3.1 and why the ITAAC value was correct.

In RAI-SRP9.2.2-SBPA-12, the applicant was asked to explain this apparent inconsistency and to adequately justify the proposed change to reduce the minimum flow rate specified in IPSAC SR 2.3.1 in order for the staff to determine if the proposed change was acceptable.

The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant indicated that 10,675 Lpm (2820 gpm) is the normal CCS flow rate to each of the shutdown cooling HXs and a flow rate of this value or higher is expected to be achieved with the CCS configured as required to perform the normal shutdown cooling function related to CCS flow (i.e., each CCS pump supplying one shutdown cooling HX, one spent fuel cooling HX, and CCS auxiliary loads). The value of 10,164 Lpm (2685 gpm) in DCD Tier 1, Table 2.3.1.2 represents the minimum required CCS flow rate to accomplish the shutdown cooling and is therefore the flow that must be demonstrated in the ITAAC.

The staff reviewed the changes in flow rate values and determined they were adequately explained since the value of 10,164 Lpm (2685 gpm) represents the 'minimum' required flow rate to accomplished the shutdown cooling and the value of 10,675 Lpm (2820 gpm) represents the 'normal' CCS flow rate to perform a 'normal' shutdown cooling function. Since the Tier 1 ITAAC and the SRs have consistent values, the staff determined that RAI-SRP9.2.2-SBPA-12 is resolved.

9.2.2.2.6 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Considerations

The applicant proposed to change the CCS flow rate and heat removal capability that are specified in ITAAC Table 2.3.1-2 to reflect proposed changes that are discussed above in Section 9.2.2.1.1. The changes that are proposed for the SWS ITAAC are consistent with the specific changes that are proposed in this section.

9.2.2.2.7 Initial Test Program Considerations

The initial test program for the CCS is discussed in Tier 2 of the DCD, Section 14.2.9.2.5, "Component Cooling Water System Testing." The stated purpose of the CCS test program is to verify that the as-installed CCS performs the DID functions described in DCD Tier 2 Section 9.2.2 of providing cooling water to DID components and transfer heat to the SWS; as well as providing cooling water to other non-safety-related components for heat removal.

9.2.2.3 Conclusion

The staff evaluated proposed changes to Revision 15 of the AP1000 DCD that pertain to the CCS. The proposed changes are documented in Revision 17 of the AP1000 DCD Tier 2,

Section 9.2.2, and are reflected in the Tier 1 ITAAC specified in Table 2.3.1-2 and the Tier 2 IPSAC specified in Table 16.2, Section 2.3. The staff's evaluation, using the guidance provided by NUREG-0800 Section 9.2.2, confirmed that: a) the proposed changes will not adversely affect safety-related SSCs; b) the SWS is capable of performing its DID and RTNSS functions; and c) ITAAC, IPSAC, and initial test program considerations are adequate and appropriate. The proposed changes conform to the existing AP1000 licensing basis as documented in Revision 15 of the approved DCD. The changes contribute to the increased standardization of this aspect of the design. Therefore, these changes meet the finality criterion for changes in 10 CFR 52.63(a)(1)(vii). The staff finds that the proposed changes are acceptable.

9.2.5 Potable Water System

9.2.5.1 Summary of Technical Information

Section 9.2.5, "Potable Water System," (PWS) of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In AP1000 DCD, Revision 17, the applicant has proposed to delete the site-specific PWS design, including the supply source, from the scope of the certified design and to add a proposed COL information item to address this information.

The basis for this change is documented in TR-124, "Removal of PWS Source and WWS Retention Basins from the Westinghouse AP1000 Scope of Certification," APP-GW-GLN-124, Revision 0 of June 2007. The applicant has identified this change in AP1000 DCD Tier 2, Revision 17, Section 9.2.5.

9.2.5.2 Evaluation

The staff reviewed all changes to the PWS in AP1000 DCD Revision 17 in accordance with the guidance in NUREG-0800 Section 9.2.4, "Potable and Sanitary Water Systems." The staff did not re-review descriptions and evaluations of the PWS in AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. The regulatory basis for Section 9.2.5 of the AP1000 DCD is documented in NUREG-1793. The specific criterion that applies to the proposed DCD change is 10 CFR 52.63(a)(1)(vii) which concerns contribution to increased standardization of the certification information.

In TR-124, "Removal of PWS Source and WWS Retention Basins from the applicant AP1000 Scope of Certification", Revision 0, the applicant proposes to remove the PWS source from the certified design including the potable water storage tank, potable water pumps, potable water jockey pump, and their associated piping. The applicant proposes to transfer responsibility for addressing the supply source and components of the PWS outside of the power block to the COL applicant through the addition of COL Information Item 9.2.11.1. Proposed COL Information Item 9.2.11.1 in DCD Revision 17 states:

> The Combined License applicant will address the components of the potable water system outside of the power block, including supply source required to meet design pressure and capacity requirements, specific chemical selected for use as a biocide, and any storage requirements deemed necessary. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room compatibility is addressed in Section 6.4.

The staff identified a wording error in the last sentence of this proposed COL Information Item where "control room *habitability"* should be used in place of the proposed "control room *compatibility."* The applicant agreed during the March 18, 2009 Public Meeting that the correct wording is "control room *habitability."* The staff finds that the proposed COL information item adequately addresses the necessary information that was removed from the DCD by this change. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In DCD Revision 17, the applicant states that no interconnections exist between the PWS and any potentially radioactive system or any system using water for purposes other than domestic water service. The changes to the PWS include the removal of portions of the PWS, and providing site specific information in COL Information Item 9.2.11.1. Because the changes are all outside the power block and do not involve possible contamination by radioactive water, the staff finds that the conclusions of NUREG-1793 Section 9.2.5 remain valid. Specifically, the staff finds that the PWS continues to satisfy GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to design provisions for controlling the release of water containing radioactive material and preventing contamination of the potable water.

9.2.5.3 Conclusion

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

The staff reviewed the applicant's proposed changes to the AP1000 PWS as documented in DCD, and in TR-124. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 PWS to meet the applicable NUREG-0800 Section 9.2.4 acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. The AP1000 PWS design continues to meet all applicable acceptance criteria, and the proposed change meets the criterion of 10 CFR 52.63(a)(1)(vii), on the basis that it contributes to increased standardization of the certification information. The staff finds that all of the changes related to the system design of the AP1000 PWS are acceptable.

9.2.7 Central Chilled Water System

9.2.7.1 Summary of Technical Information

Revision 17 of the AP1000 DCD includes proposed changes to the VWS in order to specify increased wet and dry bulb temperatures for heat load considerations and to provide additional design flexibility for COL applicants. The basis for these proposed changes are discussed in TR-108, TR-103 and TR-107, "AP1000 Technical Support Center," APP-GW-GLR-107, Revision 1, of June 2007.

9.2.7.2 Evaluation

The containment isolation interface of the VWS is safety-related. The balance of the VWS is nonsafety-related and the regulatory basis for evaluating the safety and non-safety systems is documented in Section 9.2.7, "Central Chilled Water System," of NUREG-1793. Although the VWS is nonsafety-related, the low-capacity subsystem is considered to be important to safety

because it provides chilled water for cooling safety-related and DID equipment rooms. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs. In addition, the staff evaluation focused on those items that satisfy the criteria for RTNSS, the capability of the VWS to perform its RTNSS and DID cooling functions, and the adequacy of ITAAC, test program specifications, and RTNSS availability controls that have been established for the VWS. The proposed changes were evaluated using the guidance provided by NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was judged based upon conformance with the existing AP1000 licensing basis, the guidance specified by NUREG-0800 Section 9.2.2 (as applicable), and SECY-94-084.

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

During its evaluation, the staff noted that the description that is provided in Revision 17 of the DCD, Section 9.2.7, does not describe the DID and investment protection functions of the VWS. However, the ITAAC specified in Tier 1 of the DCD, Section 2.7.2, "Central Chilled Water System," the initial test program described in Tier 2 of the DCD, Section 14.2.9.2.9, "Central Chilled Water System Testing," and Table 17.4-1, "Risk-Significant SSCs Within the Scope of D-RAP," indicate that the VWS is important for both DID and investment protection considerations. The staff found that this information was not reflected in the description that is provided for the VWS in Tier 2 of the DCD, Section 9.2.7, and that no investment protection short-term availability controls were established for this system. In RAI-SRP9.2.2-SBPA-02, the applicant was asked to provide clarification as necessary in the AP1000 DCD to better explain the DID and investment protection functions of the VWS, as well as to explain why IPSAC was not necessary recognizing that VWS is needed to support other DID non-safety systems. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant provided additional information concerning the DID and investment protection functions provided by the VWS, and similarly for IPSAC considerations. The staff determined that the additional information was incomplete and inadequate, and Tier 2 of the DCD was not revised to fully explain the VWS design basis relative to DID and investment protection considerations.

Following the June 25, 2009 audit, the applicant responded on September 4, 2009, with an RAI response revision that clarified that the VWS itself is not captured in the RTNSS program. The applicant clarified the functions of the low capacity subsystem that are to maintain the MCR, 1E electrical room and normal RNS pump rooms room temperatures and the associated concrete heat sink temperatures. The applicant identified components of the VWS that are determined to be risk-significant and are included in the scope of D-RAP. These components are identified as:

- Two pumps associated with the low capacity subsystem (as shown on DCD Tier 2 Figure 9.2.7-1 (Sheet 1), "Central Chilled Water System Piping and Instrumentation Diagram".

- Two air-cooled chillers associated with the low capacity subsystem (as shown on DCD Tier 2 Figure 9.2.7-1 (Sheet 1).

In addition the applicant provided a DCD markup adding the above items to Tier 2 DCD Section 9.2.7.2.2, "Component Description," including references to DCD Table 17.4-1, "Risk-Significant SSCs within the Scope of D-RAP." Also included in the DCD markup was a simplified drawing describing the high capacity subsystem (which was omitted in Revision 17 of the DCD) and clarification to DCD Table 9.2.7-1, "Component Data-Central Chilled Water

System," related to clarification of components between the high capacity subsystem and the low capacity subsystem.

The staff reviewed the material from the audit, RAI revision response, and DCD markup and concluded that the clarification to the D-RAP components was adequately addressed and the staff verified that there were no technical changes made to the DCD. In addition, the staff finds that the RAI response corrects the inconsistency between Section 9.2.7 of the DCD and Table 17.4-1. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.2.7.2.1 Impact of Proposed Changes on Safety-Related SSCs and Functional Capability

Increase in Site Interface Temperature Limits

As discussed in TR-108, the applicant proposes to change the site interface temperature limits to encompass a broader range of potential sites for AP1000 plants. The proposed changes for the VWS are reflected in Tier 2 of the DCD, Section 9.2.7.2.4, "System Operation." These changes relate to the design specifications for the VWS and do not result in new VWS failure modes or interactions that can adversely affect the capability of safety-related SSCs to mitigate postulated accident conditions. Also, TR-108 states that the limiting temperature specifications are used for properly sizing the air-cooled chiller, thereby assuring adequate DID cooling capability for the low-capacity subsystem. Therefore, as discussed in Section 9.2.2.2.3 of this report, the proposed changes to the site temperature interface limits are acceptable.

Removal of Smart Valves

The AP1000 design specifies the use of "smart valves" (i.e., valves that contain instrumentation such as temperature, flow and pressure that is used for control or indication) for some system applications. In the case of the VWS, smart valves (V272A/B and V261A/B/C/D) are specified as the modulating control valves. As discussed in TR-103, the applicant proposes to remove the requirement for using smart valves for this function so that standard valves can be used by COL applicants. The proposed change replaces the instrumentation that is included in the smart valve design with standard inline instrumentation as illustrated in Figure 9.2.7-1 of the DCD Tier 2. The staff determined the proposed change does not eliminate or alter the functional capabilities of any VWS valves or instruments, and will not degrade the capability or reliability of the VWS to perform its function. The staff expects this change to improve the capability to service and maintain the affected instrumentation which would tend to improve VWS availability and reliability consistent with the Commission's policy on RTNSS. Therefore, the staff considers the proposed changes to use standard inline instrumentation to be acceptable.

Design Temperature inside Containment

Tier 2 of the AP1000 DCD Revision 17, Section 9.2.7.2.2, modified the piping inside containment for the nonsafety high capacity chilled water system from a design temperature of 160 °C to 93.3 °C (320 °F to 200 °F) to accommodate both cooling and heating service. The staff finds that lowering the design temperature of the CWS piping inside containment is a reduction in conservatism. For this reason, the staff discussed this issue at the June 25, 2009, audit and issued RAI-SRP9.2.2-SBPA-13.

The applicant responded to this RAI on September 4, 2009 and stated the piping has a limit of 160 °C (320 °F) at 1,379 kPa (200 psig) and the applicant will restore the VWS piping design temperature back to 160 °C (320 °F). In addition, a DCD mark up was provided for Section 9.2.7.2.2.

The staff reviewed this response and finds that the proposed DCD change is acceptable since it returned to the design temperature based on its previous value. The higher temperature is based on the high capacity chilled water system being aligned to the hot water system for heating of containment. Since a DCD markup was provided as part of this RAI response, this item is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.2.7.2.2 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Considerations

Tier 1 of the AP1000 DCD, Section 2.7.2, "Central Chilled Water System," specifies ITAAC for the VWS. The proposed changes referred to above include design considerations and editorial changes that do not alter or otherwise affect the design specifications that must be verified to assure the DID capability of the low capacity subsystem. Therefore, the staff finds that the ITAAC for VWS are not affected by the proposed changes and will continue to be acceptable.

9.2.7.2.3 Initial Test Program Considerations

The initial test program for the VWS is discussed in Tier 2 of the DCD, Section 14.2.9.2.9, "Central Chilled Water System Testing." The purpose of VWS testing is primarily to verify that the as-installed low capacity subsystem adequately performs its DID cooling function, as well as confirming the proper function of the high capacity subsystem. The proposed changes referred to above do not alter the fundamental VWS performance considerations that apply and, therefore, the staff finds that the initial test program for VWS will continue to be acceptable.

9.2.7.3 Conclusion

The staff evaluated proposed changes to the VWS as discussed above and reflected in the AP1000 DCD, Tier 2, Section 9.2.9. The proposed changes involve a slight increase in the site temperature interface limits, elimination of smart valves from the VWS design, and design temperature changes inside containment. Based on the results of this evaluation, the staff has determined that the proposed changes will not adversely affect safety-related SSCs, the capability of the VWS to perform its DID cooling functions, or ITAAC and initial test program considerations. Consequently, the staff finds that the proposed changes are consistent with the AP1000 licensing basis and the applicable NRC review guidance specified by NUREG-0800 Section 9.2.2. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii), in that they contribute to increased standardization of the certification information; therefore, the proposed changes are acceptable.

9.2.8 Turbine Building Closed Cooling Water System

9.2.8.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.2.8, "Turbine Building Closed Cooling System," of the AP1000 DCD, Revision 15. In the AP1000, Revisions 16 and 17 the applicant proposed changes to Section 9.2.8.

The applicant proposed the following technical changes to Revision 15 of the AP1000 DCD, which are supported by information in the TRs:

1. Remove descriptive information that pertains to the alternative steam and power conversion design that no longer applies, eliminate nominal heat load values from Table 9.2.8-1, "Turbine Building Closed-Cooling Water System Equipment Load List," and revise the title of the table accordingly, and increase the maximum temperature specifications for the turbine building closed cooling water system (TCS) and its heat sink. Change Table 9.2.8-1 and delete, from the equipment load list, the condensate pump motor air cooler, condensate pump bearing oil cooler, feedwater pump motor air cooler, and condenser vacuum pump.

2. Allow the use of non-metallic pipe in the system design, designate that the heat sink for the TCS is conceptual design information (CDI) instead of the circulating water system, to specify that backwashable strainers are provided upstream of the TCS HXs, and to include information related to TCS HX and upstream strainer operation. Clarify that nonmetallic piping may be used in the TCS and deleted the reference to ASME B31.1, "Power Piping".

The bases for these proposed changes are discussed in TR-86, "Alternate Steam and Power Conversion Design," APP-GW-GLN-018, Revision 1, dated June 2007, and TR-103.

9.2.8.2 Evaluation

The TCS is nonsafety-related and the regulatory basis for evaluating this system is documented in Section 9.2.8 of NUREG-1793. The staff's evaluation of the changes that are proposed focuses on confirming that the changes will not adversely affect SSCs or those that satisfy the criteria for RTNSS. The proposed changes were evaluated using the guidance provided by NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was judged based upon conformance with the existing AP1000 licensing basis and the guidance specified by NUREG-0800 Section 9.2.2 (as applicable).

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

Impact of Proposed Changes on Safety-Related SSCs

As discussed in Section 9.2.8 of NUREG-1793, TCS piping and components are located entirely within the turbine building. No safety-related equipment is located in the turbine building and, therefore, failure of the TCS cannot lead to the failure of any safety-related SSCs. On this basis, the staff found that the proposed change conforms to Regulatory Position C2 of RG 1.29, "Seismic Design Classification," thereby satisfying GDC 2 requirements and the applicable guidance of NUREG-0800 Section 9.2.2 with respect to impact on safety-related SSCs. The proposed changes referred to above and described in TR-86 and TR-103 relative to the TCS do not alter the location of the TCS relative to safety-related SSCs and, consequently, the basis for NRC approval in NUREG-1793 Section 9.2.8 remains valid in this respect.

The staff also noted that TR-103 (Page 23, Item 4) indicates that the cooling medium for the TCS HXs is changed from circulating water to a generic "cooling water" that can be provided by either circulating water and/or raw water makeup to the cooling tower basin. Section 9.2.8.1.2, "Power Generation Design Basis," describes the heat sink for the TCS as circulating water.

However, DCD Tier 2, Revision 17, Section 10.4.5.1.2, "Power Generation Design Basis," for the circulating water system indicates that: "The CWS and/or makeup water from the raw water system supplies cooling water to the turbine building closed cooling water system (TCS) heat exchangers..." Consequently, the information provided in Sections 9.2.8 and 10.4.5 is inconsistent. It is not clear if the intent is to establish a CDI item for COL applicants to address or to provide the option of using the circulating water and/or raw water makeup to the cooling tower basin instead of establishing a CDI item. In RAI-SRP9.2.2-SBPA-01 the applicant was asked to provide additional information to explain the intention of the proposed change, and to revise Tier 2 of the DCD, Sections 9.2.8 and 10.4.5 as necessary to eliminate the inconsistency. The applicant indicated that the intent of the proposed change was not to establish a CDI item for COL applicants but rather to provide an option for COL applicants to utilize circulating water system (CWS) cooling tower makeup water flow or circulating water flow as the cooling water source, at the applicant's discretion, for the TCS HXs. The applicant proposed an additional change to Tier 2 Section 10.4.5.2.2 of the DCD in order to eliminate this inconsistency.

The staff reviewed the proposed DCD markups and determined that the changes eliminated the inconsistency between DCD Tier 2 Sections 9.2.8 and 10.4.5. The staff verified that the proposed DCD markups in the RAI response were added to the DCD. Therefore, RAI-SRP9.2.2-SBPA-01 is resolved.

Impact of Removal of TCS Equipment Heat Loads from DCD Table 9.2.8-1

The applicant deleted equipment heat loads that were originally listed because the cooling water for the deleted equipment is a site-specific design rather than part of the AP1000 standard design.

Based on its evaluation, the staff finds this change acceptable because it does not affect the conclusions in NUREG-1793 Section 9.2.8.

Impact of TCS piping material clarification

The applicant deleted reference to ASME B31.1 and has added nonmetallic as a possible piping material to be utilized in the TCS.

The staff reviewed this clarification and determined that this change is consistent with Section 3.2.2.7, "Other Equipment Classes." TCS is classified as 'Class E', and specific piping codes are not typically described in Section 3.2.2.7 for 'Class E' systems. In addition, the clarification to use nonmetallic piping in the TCS gives flexibility in the use of non-corrosive material as needed. In a corrosive water environment, nonmetallic piping material outperforms metallic materials whereas metallic piping material may require replacing over time. Based on the staff's evaluation, the TCS piping material classification is acceptable.

9.2.8.3 Conclusion

The staff evaluated proposed changes to the TCS that are discussed above and are reflected in the AP1000 DCD, Tier 2, Revision 17, Section 9.2.8. Based on the results of this evaluation, the staff has determined that the proposed changes will not adversely affect safety-related SSCs, that they meet the criterion of 10 CFR 52.63(a)(1)(vii), in that they contribute to increased standardization of the certification information and are, therefore, acceptable.

9.2.9 Waste Water System

9.2.9.1 Summary of Technical Information

Section 9.2.9, "Waste Water System" of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 17, the applicant proposed to make several changes to Subsections 9.2.9.2.1, "General Description," 9.2.9.2.2, "Component Description," 9.2.9.5, "Instrumentation Applications," and 9.2.11.2, "Waste Water Retention Basin" of the certified design. The applicant also made changes to DCD Tier 1 Section 2.3.29, "Radioactive Waste Drain System," as a result of changes to the nonradioactive waste water system (WWS). All of these changes are related to the removal of the PWS from the scope of the DC, as described in TR-124.

9.2.9.2 Evaluation

The staff reviewed the proposed changes to Tier 2 Subsections 9.2.9.2.1, 9.2.9.2.2, 9.2.9.5, and 9.2.11.2, and Tier 1 Section 2.3.29, of the AP1000 DCD Revision 17, in accordance with the guidance of NUREG-0800 Section 9.3.3, "Equipment and Floor Drainage System." The regulatory basis for Section 9.2.9 of the AP1000 DCD is documented in Section 9.2.9 of NUREG-1793.

The specific criterion that applies to the proposed DCD changes is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

The staff finds that the proposed changes in Section 9.2.9.2.1 are limited to the removal of references to the waste water retention basin pump and transfer pumps since these are no longer part of the DC. In addition, the applicant removed the description of the condenser waterbox drains, which were previously routed to the waste water retention basin. The applicant stated that the design and routing of the condenser waterbox drains will be incorporated into the site-specific CWS design (discussed in Section 10.4.5 of NUREG-1793, as supplemented by this report). Because the design includes site-specific criteria, the staff finds acceptable that the applicant defers the design details of the condenser waterbox drains to the COL applicant. However, the staff determined that this information should be included in the COL information item described in DCD Tier 2 Section 10.4.12. The staff requested that the applicant include this information in the DCD in RAI-SRP9.3.3–SBPA-01.

In DCD Section 9.2.9.5, the applicant removed references to the level instrumentation and pump controls located in the waste water retention basin. The staff finds this change acceptable because it is not related to the guidance and acceptance criteria in NUREG-0800 Section 9.3.3.

In a letter dated June 20, 2008, the applicant responded to RAI-SRP9.3.3-SBPA-01 by proposing a markup of Section 10.4.12.1 of the DCD to state that the COL applicant will identify the action of routing the condenser waterbox drains with the site-specific CWS. The staff finds this markup acceptable because the applicant made the DCD clear with regard to what is within the scope of the DC. This DCD change was incorporated into Revision 17; RAI-SRP9.3.3-SBPA-01 is, therefore, resolved.

In DCD Section 9.2.9.2.2, the applicant removed the component description of the waste water retention basin and its associated basin transfer pumps. The applicant replaced this information with a reference to the COL information item described in Section 9.2.11, which requires the

COL applicant to provide the site-specific information for these components. Because the waste water retention basin and its associated basin transfer pumps and piping are site-specific components, the staff finds it acceptable to defer this to the COL applicant.

In DCD Section 9.2.9.5, the applicant also relocated the radiation monitor from the waste water retention basin to the turbine building sump. The staff reviewed this change to ensure that all effluents in the WWS that discharge to the turbine building sump will be monitored prior to disposition, as required by GDC 60. However, based on the information provided, the staff was unable to verify that all nonradioactive effluents will be monitored prior to disposition. For example, in DCD Revision 15, the condenser waterbox drains were routed directly to the WWS retention basin. The staff requested that the applicant address this in RAI-SRP9.3.3–SBPA-02.

By letter dated June 20, 2008, the applicant responded to RAI-SRP9.3.3-SBPA-02. In its response, the applicant identified all the sources of waste water that will drain downstream of the turbine building sump, which include diesel fuel area sump (upstream of the oil separator), SWS/CWS backwash, and other site specific effluent (e.g., CWS waterbox drain). The service water flow is provided with a radiation monitor. All systems interfacing with the CWS that have plausible potential for radioactive contamination are provided with radiation monitoring. The diesel fuel area sump effluent does not interact with any potentially radioactive sources during operation, nor are there any recognized radioactive sources located in the vicinity of the portion of WWS. Effluents that are site specific are under the responsibility of the COL applicant to ensure proper radiation monitoring is designed into the system, as noted in COL Information Item 11.5-1 (Section 11.5.7 of the AP1000 DCD). Based on the above, the staff concludes that all potentially radioactive effluents from the standard plant are properly monitored for radiation prior to disposition off site, as required by GDC 60.

Based on the evaluation of the DCD information and the above response to RAI-SRP9.3.3-SBPA-02, the staff finds this change acceptable because it does not affect the NUREG-1793 Section 9.2.9 findings and conclusions related to controlling release of radioactive materials. Therefore, RAI-SRP9.3.3-SBPA-02 is resolved.

In DCD Revision 17, Tier 1, Section 2.3.29, the applicant removed the references to the waste water retention basin and replaced them with reference to the turbine building sump. These changes are consistent with the changes to Tier 2 Section 9.2.9.5 (discussed above). In short, since the waste water retention basin is no longer in the scope of the certification, the applicant relocated the system's detection and isolation functions to the turbine building sump, rather than the waste water retention basin, thus providing conformance with GDC 60. Therefore, the staff finds this Tier 1 change acceptable.

9.2.9.3 Conclusion

The staff reviewed the applicant's proposed changes to the AP1000 regarding the nonradioactive WWS as documented in AP1000 DCD Tier 2 Subsections 9.2.9.2.1, 9.2.9.2.2, 9.2.9.5, and 9.2.11.2, and Tier 1 Section 2.3.29. The staff finds that the proposed changes meet the acceptance criteria in NUREG-0800 Section 9.3.3. The staff concludes that the AP1000 WWS continues to meet all applicable acceptance criteria and that proposed changes are properly documented in the updated AP1000 DCD. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii); in that they contribute to increased standardization of the certification information.

9.2.10 Hot Water Heating System

9.2.10.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.2.10, "Hot Water Heating System" of the AP1000 DCD, Revision 15.

In AP1000 DCD Revision 17, the applicant identified the following Tier 2 changes associated with the Hot Water Heating System, DCD Sections 9.2.10.2.1, 9.2.10.2.3, and 9.2.10.3.

1. The applicant proposed to modify the AP1000 DCD Section 9.2.10.2.1, "General Description," to delete the method of matching the system heat load and regulating system temperature using a heater bypass valve.

2. The applicant proposed to modify the AP1000 DCD Section 9.2.10.2.3 "System Operation," to delete reference to a three-way diverting valve that regulates the temperature of the hot water system. No technical basis for this change is provided. The staff notes that the system operation already includes a description of the intended method temperature control to individual heating coils. Temperature regulation through the use of a heater bypass is not required in order for the system to perform its functions.

3. The applicant proposed to modify the AP1000 DCD Section 9.2.10.3, "Safety Evaluation," to delete reference to a three-way diverting valve that regulates the temperature of the hot water system.

The applicant identified no Tier 1 changes associated with the hot water heating system.

9.2.10.2 Evaluation

The specific criterion that applies to the proposed DCD changes is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

The staff reviewed all changes to the hot water heating system (VYS) in the AP1000 DCD Revision 17 and finds them acceptable.

The VYS has no safety-related function and, therefore, no nuclear safety design basis. The following evaluation discusses the results of the staff's review of the Revision 17 changes.

Deletion of 3-Way Diverting Valve for Temperature Regulation

In DCD Revision 17, Sections 9.2.10.2.1, 9.2.10.2.3, and 9.2.10.3, the applicant proposed to delete discussion of the method of regulating VYS system temperature through the use of a 3-way diverting valve that would bypass the hot water heaters. The applicant deleted this information to allow flexibility in designing and constructing the VYS. The staff finds this change acceptable, since it does not affect the NUREG-1793 Section 9.2.10 findings and conclusions.

9.2.10.3 Conclusion

In NUREG-1793, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the

requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 VYS as documented in DCD, Revision 17. The staff concludes that the VYS will continue to comply with the conclusions documented in NUREG-1793 Section 9.2.10. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to increased standardization of the certification information.

9.3 Process Auxiliaries

9.3.1 Compressed and Instrument Air System

9.3.1.1 Summary of Technical Information

In the certified design, AP1000 DCD Revision 15, the staff approved Section 9.3.1, "Compressed and Instrument Air System."

In AP1000 DCD Revision 17 the applicant identified the following Tier 2 changes associated with the compressed and instrument air system (CAS), DCD Table 9.3.1-1. Table 9.3.1-2, Table 9.3.1-3, and Table 9.1.3-4.

1. The applicant proposed to change the AP1000 DCD Section 9.3.6.3.7, Table 9.3.1-1, and Figure 9.3.6-1 Sheet 1 of 2, such that the normal position of air operated containment isolation valve CVS-PL-V092 is open. CVS-PL-V092, "Hydrogen Addition Containment Isolation Valve," is now normally open and fails closed on loss of air. Conforming changes have been made to Table 9.3.1-1, Safety-Related Air-Operated Valves and Figure 9.3.6-1.

2. The applicant proposed to modify the AP1000 DCD Table 9.3.1-2, "Nominal Component Design Data – Instrument Air Subsystem," to clarify that the capacity of the air receivers is a minimum of 19 cubic meters (m^3) (672 cubic feet (f^3)) instead of exactly 19 m^3 (672 f^3). The technical basis for this change is to account for growth in demand on the system by ensuring that the receiver tank capacity is at least 19 m^3 (672 f^3).

3. The applicant proposed to modify the AP1000 DCD Table 9.3.1-3, "Nominal Component Design Data – Service Air Subsystem," to clarify that the capacity for the air receiver is a minimum of 19 m^3 (672 f^3) instead of exactly 19 m^3 (672 f^3). The technical basis for this change is to account for growth in demand on the system by ensuring that the receiver tank capacity is at least 19 m^3 (672 f^3).

4. The applicant proposed to modify the AP1000 DCD Table 9.3.1-4, "Nominal Component Design Data – High Pressure Air Subsystem," to clarify that the system design pressure is reduced from 34.5 MPa (5,000) psig to 27.6 MPa (4,000 psig). The technical basis for this change is to revise the system design pressure.

9.3.1.2 Evaluation

The staff reviewed all changes to the CAS in the AP1000 DCD Revision 17 in accordance with NUREG-0800 Section 9.3.1, "Compressed Air System." The regulatory basis for Section 9.3.1 of the AP1000 DCD is documented in NUREG-1793, which states that staff acceptance of the

design is contingent on compliance with the following requirements in NUREG-0800 Section 9.3.1:

- GDC 1, as it relates to systems and components being designed, fabricated, and tested to quality standards in accordance with the importance of the safety functions to be performed.

- GDC 2, as it relates to the capability of safety-related CAS components to withstand the effects of earthquakes.

- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions

The CAS has no safety-related function other than containment isolation. The following evaluation discusses the results of the staff's review of the Revision 17 changes.

The specific criterion that applies to the proposed DCD changes is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

Containment Isolation Valve CVS-PL-V092 Change in Normal Position

In AP1000 DCD Table 9.3.1-1, the applicant proposed to change the normal position of air-operated containment isolation valve, CVS-PL-V092, from closed to open. The applicant made a conforming change on Figure 9.3.6-1 Sheet 1 of 2, "Chemical and Volume Control System Piping and Instrumentation Diagram," showing valve CVS-PL-V092 to be normally open, in order to facilitate zinc acetate injection into the RCS. The evaluation of the valve position is in Section 9.3.6 of this report. The basis for this proposed change is APP-GW-GLN-002 (TR-32).

Based on its evaluation, the staff finds that this change does not affect the NUREG-1793 Section 9.3.1 assumptions, findings, or conclusions with respect to compliance with GDC 1, 2, or 5, as referenced in NUREG-0800 Section 9.3.1, and is, therefore, acceptable.

The impact of this change in the containment isolation system is discussed in Section 6.2.4, "Containment Isolation System," of this report.

DCD Tables 9.3.1-2 and 9.3.1-3, Nominal Component Design Data – Instrument Air and Service Air Subsystems

In DCD Tables 9.3.1-2 and 9.3.1-3, the applicant clarified that the capacity of the air receivers is a minimum of 19 m^3 (672 f^3) instead of being exactly 19 m^3 (672 f^3). This change is conservative in that the receiver tank capacity can be greater than 19 m^3 (672 f^3) and still meet design criteria.

Based on its evaluation, the staff finds that this change does not affect the NUREG-1793 Section 9.3.1 assumptions, findings, or conclusions with respect to compliance with GDC 1, 2, or 5, as referenced in NUREG-0800 Section 9.3.1, and is, therefore, acceptable.

DCD Table 9.3.1-4, Nominal Component Design Data – High Pressure Air Subsystem

In DCD Table 9.3.1-4, the applicant modified the high-pressure air subsystem design pressure from 34.5 MPa (5,000) psig to 27.6 MPa (4,000 psig).

The staff finds that this change does not affect the NUREG-1793 Section 9.3.1 assumptions, findings, or conclusions with respect to compliance with GDC 1, 2, or 5, as referenced in NUREG-0800 Section 9.3.1, and is, therefore, acceptable.

9.3.1.3 Conclusion

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

The staff reviewed the applicant's proposed changes to the AP1000 CAS as documented in DCD, Revision 17. The staff concludes that the CAS will continue to comply with GDC 1, 2, and 5 as stated in NUREG-1793 Section 9.3.1. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to increased standardization of the certification information.

9.3.2 Plant Gas System

In the certified design, AP1000 DCD Revision 15, the staff approved Section 9.3.2, "Plant Gas System."

In DCD Revision 17, the applicant proposed to make the following changes to DCD Revision 15:

1. In Section 9.3.2.2.1, "General Description" the applicant proposed to change the location of both the packaged nitrogen system and the carbon dioxide portion of the plant gas system from inside the turbine building to "in the gas storage area in the yard."

2. In Section 9.3.2.2.2, "Component Description" the applicant proposed to change the cryogenic liquid carbon dioxide insulated storage tank from double wall to single wall.

Section 3.5.1.4 of this report includes an analysis of storage tanks as a potential missile source and Section 6.4 of this report includes analyses of onsite chemicals.

9.3.3 Primary Sampling System

9.3.3.1 Summary of Technical Information

In the certified design, AP1000 DCD Revision 15, the staff approved Section 9.3.3, "Primary Sampling System."

In DCD Revision 17, the applicant proposed to make the following change to DCD Revision 15:

- In Section 9.3.3.2.2, "Nuclear Sampling System - Gaseous" the applicant changed the discharge location of the purge gas return from the effluent holdup tank of the liquid radwaste system to the containment sump.

9.3.3.2 Evaluation

The staff reviewed the proposed change to Tier 2 Section 9.3.3.2.2 of the AP1000 DCD Revision 17, in accordance with the guidance of NUREG-0800 Sections 9.3.2, 9.3.3, and 11.5. The regulatory basis for Section 9.3.3 of the AP1000 DCD is documented in Section 9.3.3 of NUREG-1793.

The specific criterion that applies to the proposed DCD change is 10 CFR 52.63(a)(1)(vii) which concerns contribution to increased standardization of the certification information.

NUREG-0800 Section 9.3.2 states that provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures at ALARA. As the sample line for the containment atmosphere returns the purge gas back to the containment, the proposed change meets the acceptance criteria of the NUREG-0800 guidance and the ALARA requirements. The staff finds the change in location of the purge line discharge from the effluent holdup tank to the containment sump acceptable.

9.3.3.3 Conclusion

The staff reviewed the applicant's proposed changes to the AP1000 regarding the Primary Sampling System as documented in AP1000 DCD Tier 2 Section 9.3.3, Revision 17. The staff concludes that the AP1000 Primary Sampling System continues to meet all applicable acceptance criteria and the proposed changes are properly documented in the updated AP1000 DCD. The proposed change meets the criterion in 10 CFR 52.63(a)(1)(vii) on the basis that it contributes to increased standardization of the certification information and is, therefore, acceptable.

9.3.5 Equipment and Floor Drainage System

9.3.5.1 Summary of Technical Information

Section 9.3.5, "Equipment and Floor Drainage System" (EFDS) of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 17, the applicant proposed to make changes to Sections 9.3.5.1.2, "Power Generation Design Basis," and 9.3.5.2.2, "Component Description" of the certified design as follows:

1. The applicant clarified that an exception exists to the minimum slope of 10.42 millimeter per meter (mm/m) (1/8 inches per foot (in/ft)) for drain lines, which is the embedded drain piping in area 2 of the auxiliary building at elevation 20.27 m (66 ft, 6 in) with a minimum slope of 5.21 mm/m (1/16 in/ft) for embedded drain piping.

2. The applicant clarified that each sump is fitted with a vent connection to the radiologically controlled area ventilation system (VAS) exhaust system to exhaust potential sump gases, instead of exhausting into the room.

9.3.5.2 Evaluation

The staff reviewed the proposed changes to Tier 2 Sections 9.3.5.1.2 and 9.3.5.2.2 of the AP1000 DCD Revision 17, in accordance with the guidance of NUREG-0800 Section 9.3.3.

The regulatory basis for Section 9.3.5 of the AP1000 DCD is documented in Section 9.3.5 of NUREG-1793.

The specific criterion that applies to the proposed DCD changes is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

DCD Revision 15, Section 9.3.5.1.2 provides for a minimum slope of 10.42 mm/m (1/8 in/ft) for embedded drain lines in Level 1 (Area 2) of the auxiliary building at elevation 20.27 m (66 ft, 6 in). In DCD Revision 17, Section 9.3.5.1.2, the applicant changed the minimum slope from 10.42 mm/m (1/8 in/ft) to 5.21 mm/m (1/16 in/ft) for embedded drain lines.

In order for the staff to complete its evaluation, the staff asked the applicant, in RAI-SRP9.3.5-SBPA-01, to justify the change from the minimum slope of 10.42 mm/m (1/8 in/ft) of drain pipe length, and to address its impact on flooding in that part of the auxiliary building by evaluation.

The applicant provided a response to RAI-SRP9.3.5-SBPA-01 in a letter dated September 17, 2009. The RAI response stated that the drain lines located in Level 1 of the auxiliary building (Elevation 20.3 m (66 ft, 6 in)), nonradioactive controlled area, are not credited in the flooding analysis. Because no credit is taken for the drains in this location, the staff finds the applicant's justification for the embedded drain piping slope in this area acceptable and, therefore, GDC 4 is met. Therefore, RAI-SRP9.3.5-SBPA-01 is resolved.

In DCD Section 9.3.5.2.2, the applicant clarified that each sump is fitted with a vent connection to the VAS exhaust system instead of exhausting directly into the room. This allows potential sump gases from each sump to be directed to an exhaust system for the control of airborne radioactivity. The staff finds that this change would minimize potential release of airborne radioactivity or other harmful gases in sump rooms and is, therefore, acceptable.

The staff finds that the applicant adequately identified changes that would not adversely impact the compliance of the EFDS with the guidance in NUREG-0800 Section 9.3.3. Thus, the staff finds that the applicant continues to meet GDC 2, 4, and 60.

9.3.5.3 Conclusion

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

The staff reviewed the applicant's proposed changes to the AP1000 EFDS as documented in DCD, Revision 17. The staff concludes that the EFDS will continue to comply with the conclusions documented in NUREG-1793 Section 9.3.5. The proposed change meets the criterion of 10 CFR 52.63(a)(1)(vii); on the basis that it contributes to increased standardization of the certification information.

9.3.6 Chemical and Volume Control System

9.3.6.1 Summary of Technical Information

In a letter dated April 5, 2006, the applicant submitted TR-32, "Zinc Addition," APP-GW-GLN-002, which proposed changes to the AP1000 CVS design to incorporate the ability to inject a small quantity of zinc acetate into the RCS through the CVS. In TR-80, "Markup of AP1000 Design Control Document Chapter 7," APP-GW-GLR-080, Revision 0, of October 2007, the applicant proposed CVS design changes related to boron dilution events. In DCD Revision 17, the applicant proposed to change DCD Section 9.3.6, "Chemical and Volume Control System," to incorporate the CVS design changes described in TR-32 and TR-80.

9.3.6.2 Evaluation

The AP1000 CVS as described in DCD Section 9.3.6 is designed to perform the functions of the RCS purification, inventory control and makeup, chemical shim and control, oxygen control, pressurizer auxiliary spray, and borated makeup to the auxiliary equipment. The safety evaluation accepting the CVS design of DCD Section 9.3.6, Revision 15, was described in NUREG-1793, Section 9.3.6. The review was performed using the guidance of NUREG-0800 Section 9.3.4, "Chemical and Volume Control System (PWR) (Including Boron Recovery System)," to assess compliance with the requirements for system performance of necessary functions during normal, abnormal, and accident conditions described in GDC 1; GDC 2; GDC 5; GDC 14, "Reactor Coolant Pressure Boundary"; GDC 29, "Protection Against Anticipated Operational Occurrences"; GDC 33, "Reactor Coolant Makeup"; GDC 35, "Emergency Core Cooling"; GDC 60; and GDC 61.

In Revision 17 of DCD, the applicant proposed changes to DCD Section 9.3.6 on the CVS design with: (1) the capability for zinc addition to the RCS; and (2) modifications related to boron dilution events. The staff's review of these changes is to assure continued compliance with the relevant requirements specified in the above GDCs.

The specific criterion that applies to the changes referred to above include 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

Zinc Addition

In TR-32, the applicant proposed a modification to the CVS design to provide the capability to inject a small quantity of zinc acetate into the RCS. In DCD Revision 17, the following subsection is added to DCD Section 9.3.6.2 as one of the CVS chemical control functions:

> 9.3.6.2.3.3 Zinc Addition
>
> A soluble zinc compound may be added to the coolant as a means to reduce radiation fields within the primary system and to reduce the potential for crud-induced power shift (CIPS). The zinc used may be either natural zinc or zinc depleted of ^{64}Zn.

Also, DCD Section 9.3.6.2.1.1, "Ionic Purification," is revised to include the removal of zinc during periods of zinc addition as an added ionic purification function of the mixed bed

demineralizers in the purification loop, in addition to removing ionic corrosion products and certain ionic fission products.

TR-32 provides a description and evaluation of the proposed CVS design change to incorporate the ability to inject a small quantity of zinc acetate into the RCS through the CVS. For AP1000, zinc addition will be an optional mode of operation, and the equipment specifically used for storing and pressurizing the zinc acetate is not described. However, as discussed below, minor changes to the base AP1000 CVS design are required to allow for zinc addition to be used as an optional mode of operation.

TR-32 also states that zinc acetate will be added using the same piping and valving as the hydrogen (H_2) addition. The proposed hardware change is to replace a portion of the one-inch pipe downstream of the containment isolation valve with a heavier wall half-inch pipe. To accomplish this design change, in DCD Revision 17, Figure 9.3.6-1 was changed by (1) changing the hydrogen addition line from "H2 ADD" to a hydrogen/zinc addition line "H2/ZINC ADD"; (2) adding a 2.5 cm by 1.27 cm (1 in by .5 in) reducer downstream of valve V065 in the "H2/ZINC ADD" line; and (3) renumbering the portion of the "H2/ZINC ADD" line downstream of the reducer L064 with the specification changed from 2.5 cm (1 in) BBC to 1.27 cm (0.5 in) BBC. According to TR-32, this is made to reduce the piping volume and reduce the transit time for the H_2 and the zinc acetate supply, and will not alter the load of the supply piping. The staff finds this change acceptable because there is no effect on the integrity of the reactor coolant pressure boundary. Also, in DCD Revision 17, in Section 9.3.6.3.7, the hydrogen addition containment isolation valve V092, which is located outside the containment, was changed from the "normally closed, fail closed" position to "normally open, fail closed" position; and V092 in Figure 9.3.6-1 was changed to the "normally open" position. According to TR-32, this change is made to reflect the time that the valve must be open to permit zinc additions. The staff finds this change acceptable because this containment isolation valve automatically closes on a containment isolation signal from the protection and safety monitoring system and, therefore, the containment isolation function is not affected.

Chemistry and Materials Impacts During Normal Operation

According to TR-32, zinc addition has been demonstrated to change the oxide film on primary piping and components, significantly reducing occupational exposure (due to less nickel and cobalt in deposit) and the potential for CIPS. The applicant also stated that laboratory tests indicate a beneficial effect of zinc addition on the major materials in a PWR system because of the reduced corrosion rates and that operating industry experience has shown that up to 40 parts per billion can be added with no adverse effects. The applicant stated that the effect of zinc on the reactor coolant was calculated to be less than 0.2 pH units, which it considers negligible. The staff reviewed industry experience with zinc injection in operating plants and agrees that there is sufficient experience to support the conclusion that there is no deleterious effects on reactor coolant chemistry or reactor coolant pressure boundary materials at zinc injection concentrations up to the maximum limit proposed by the applicant. Although the beneficial effect of zinc addition on major materials at higher concentrations has not yet been fully established through laboratory tests, the reduction in occupational exposure and CIPS has been demonstrated through operating experience.

Post Accident Water Chemistry Impacts

Hydrogen generation caused by corrosion of reactive metals such as zinc and aluminum following a design-basis accident (DBA) is a concern that should be addressed according to

NUREG-0800 Section 6.1.1. However, according to TR-32, because zinc exists as a divalent cation (Zn^{+2}) in solution in the primary coolant and embedded in the corrosion film, hydrogen generation is not expected. The staff agrees that because zinc is added as zinc acetate salt rather than metallic zinc, the zinc would only exist in ionic form that cannot produce hydrogen as a corrosion reaction byproduct. Further, even if the zinc cations in solution could react to form hydrogen, the staff concurs that the amount of zinc is small and would not produce a significant amount of hydrogen.

In addition, given that the volume of reactor coolant is small compared to the borated water volume in the sump, the applicant stated that the effect of zinc on the sump pH following a DBA would also be negligible. The staff finds this statement reasonable due to the low concentration of zinc compared to the concentrations of buffer and boric acid in the sump water following a DBA.

Based on the above, the staff finds the applicant's evaluation of zinc addition on post accident chemistry to be acceptable because the evaluation is based on basic principles of chemistry and the quantity of zinc involved is small.

Fuel Corrosion and Crud Effects

According to TR-32, the addition of zinc to the RCS could result in additional crud deposit on the fuel cladding surface. The applicant performed oxide thickness measurements that showed the crud deposit was thin and the effect of the corrosion rate with zinc injection was statistically insignificant as compared to the corrosion rate without zinc injection. In response to RAI-TR32-07, the applicant provided data from 12 fuel surveillance campaigns, based on which it concluded that there were no observed adverse effects on the fuel cladding performance due to zinc addition. The staff reviewed the plant surveillance data for oxide thickness that were provided in the response to RAI-TR32-07 and agrees with the applicant's conclusion because there was no statistical difference in the measured oxide thickness for cladding with and without exposure to zinc in the coolant. Further, the staff's review of industry experience related to zinc impacts on fuel indicates there are no adverse effects for low- to medium-duty cores. The applicant indicated that the absence of deleterious effects on cladding will be confirmed through cycle-specific reload analyses with zinc addition. The staff identified Open Item OI-SRP9.3.6-SRSB-01 for the applicant to explain how cycle-specific reload analyses can confirm no adverse effect of zinc addition for the AP1000 with a high-duty core and why a fuel surveillance program is not needed to confirm the absence of adverse crud effects.

To provide further definition to assist the applicant in understanding the open item, the staff communicated the following questions to the applicant via an email dated October 23, 2009:

1. Is the AP1000 core design as described in DCD Revision 17 considered a high-duty core?

2. If the AP1000 is considered a high-duty core design, how will it be assured that there will not be problems with excessive crud buildup or uneven crud buildup in operating AP1000 plants? This answer may necessitate either operating experience that demonstrates that a fuel surveillance program is unnecessary, or a recommendation for a fuel surveillance program to be implemented (COL Item).

3. Explain how cycle-specific reload analyses would confirm that zinc would not have a significant effect on cladding corrosion.

4. Since one of the objectives of zinc addition is to reduce the potential for CIPS, will an evaluation of CIPS potential be performed as part of the cycle-specific reload analysis? If so, provide details of the evaluation and a relative comparison to operating plants with and without CIPS. Also, since zinc injection will initially increase the reactor coolant Ni concentration, explain in detail the zinc injection strategy to be employed to minimize the CIPS potential.

In a letter dated February 18, 2010, the applicant responded to Open Item OI-SRP9.3.6-SRSB-01 and specifically the 4 questions as follows:

1. The applicant stated that "According to EPRI HDCI [High-duty Core Index], AP1000 would be classified as a low to medium duty plant." The applicant provided a table containing the parameters used to calculate the HDCI. The staff performed a confirmatory calculation using the same parameters and obtained the same result. The applicant further stated that "since in terms of boiling duty the AP1000 is approaching that of other Westinghouse high duty plants, we are conservatively treating AP1000 as a high duty plant. There are other currently operating PWRs that are higher duty and also use zinc addition, so AP1000 is bounded by current operating experience."

The staff finds the response to Item 1 acceptable because a confirmatory calculation verified the HDCI for the AP1000, and because the applicant is conservatively treating the AP1000 core design as high-duty.

2. The applicant indicated that zinc addition will be employed to reduce RCS surface corrosion rates beginning from hot functional testing. The applicant also stated that experience with zinc addition in current PWRs following steam generator replacement indicates substantial benefits in reducing corrosion rates when zinc is applied to fresh metal surfaces such as those found following steam generator replacement. Papers on first cores show similar benefits will occur with AP1000, but right from the beginning of plant operation.

   The applicant also indicated that there have been several additional high-duty plants that began zinc addition since 2003, which were not all reflected in the EPRI Zinc Addition Guidelines cited by the staff. The applicant listed Callaway, Vogtle 1 and 2, Byron 2, Braidwood 2, South Texas 1 and 2, and Watts Bar 1 as high-duty plants that have successfully operated with zinc addition and no problems with crud deposition or fuel performance related to zinc. The applicant also stated that fuel examinations following zinc addition have been completed at numerous high-duty PWRs and continue to show no increase in cladding corrosion and no deleterious impact on fuel crud deposits.

   Finally the applicant stated that AP1000 will have a robust fuel inspection program looking not only at crud but other things using EPRI fuel reliability guidelines.

The staff finds the applicant's response to Item 2 acceptable because:

a) Based on our review of industry experience related to zinc addition, the staff agrees that excessive nickel release should not occur if zinc addition starts during hot functional testing. This conclusion is supported by data from plants that started injecting zinc concurrently with steam generator replacement reported in EPRI

Report #1013420, "Pressurized Water Reactor Primary Water Zinc Application Guidelines," of December 2006.

b) The applicant cited additional industry experience with zinc addition in high-duty cores that supports the applicant's assertion that zinc addition will not cause increased CIPS risk in high-duty cores.

3. The applicant stated that zinc has been shown not to interact with zircaloy clad fuel and does not increase clad corrosion. The applicant further stated that reload-specific corrosion analyses do not need to be penalized due to zinc addition because as zinc began to be applied to higher duty cores additional fuel surveillance was undertaken to determine if any increased cladding corrosion was occurring. Finally the applicant stated that the surveillances have not shown any indication of enhanced clad corrosion for zinc application in these higher duty cores where crud deposits were present.

   Additionally, the applicant indicated that the cycle specific reload analysis described in the response to Item 4 will demonstrate that the reload designs should not result in excessively thick crud deposits.

The staff finds the applicant's response to Item 3 acceptable because its review of operating experience related to zinc addition confirms the applicant's claim that there are no adverse effects on fuel-cladding corrosion caused by zinc addition.

4. In response to the first part of Item 4, the applicant stated that a CIPS risk analysis is currently performed using EPRI guidelines and methods for every reload design performed in Westinghouse, and that this process will be performed as part of the initial core and each reload core analysis for AP1000. The applicant further stated that the VIPRE/BOA methods will be used as recommended in the EPRI AOA Guidelines (PWR Axial Offset Anomaly (AOA) Guidelines, Revision, 1008102, Final Report, June 2004, EPRI).

   In response to the second part of Item 4, the applicant indicated that no increase in reactor coolant nickel concentration is expected since zinc addition will start during hot functional testing; thus, the zinc will be incorporated into the corrosion films as they form on the fresh metal surfaces. According to the applicant, this is a more favorable situation compared to the addition of zinc to existing plants with mature corrosion films, which causes nickel to be displaced from the corrosion films. Therefore, the applicant's response indicates they expect no increase in crud due to displacement of nickel.

The staff finds the response to the first part of Item 4 acceptable because the applicant will be using accepted industry computer codes for analysis of the crud and CIPS risk (VIPRE/BOA). This is consistent with the approach used for operating Westinghouse plants.

The staff finds the response to the second part of Item 4 acceptable because, based on its review of industry experience related to zinc addition; the staff agrees that excessive nickel release should not occur if zinc addition starts during hot functional testing. This conclusion is supported by data from plants that started injecting zinc concurrently with steam generator replacement reported in EPRI Report #1013420.

The staff finds that the applicant has adequately addressed potential fuel corrosion and crud effects of zinc addition because:

- The applicant has presented sufficient industry experience with zinc addition in reactors with high-duty cores to demonstrate that problems with cladding corrosion, excessive crud buildup, or CIPS are not expected.

- The applicant has proposed to use industry-accepted computer codes to model and predict crud formation, for each reload, to confirm that a problem will not occur with crud or CIPS.

Although the applicant did not propose to modify the DCD to include any of the information supplied in the open item response, or provide a COL information item to ensure the COL performs the activities described, the staff notes that there is no specific regulatory requirement for a plant to monitor, model or test for crud buildup or CIPS. Therefore, the proposed reload analyses and fuel surveillance program are not mandatory. The staff considers Open Item OI-SRP9.3.6-SRSB-01 resolved.

Based on the above evaluation, the staff concludes that the zinc addition into the RCS as an operational option, and the associated DCD changes associated with zinc addition discussed above, are acceptable.

Modification to Boron Dilution Event

In DCD Revision 17, the applicant proposed the following changes to DCD Section 9.3.6 associated with boron dilution events:

In DCD Revision 17, Section 9.3.6.3.7, "Chemical and Volume Control Systems Valves," a sentence is added regarding the makeup line containment isolation valves, which are normally open motor-operated globe valves, to state that the valves close on a source range flux doubling signal to terminate possible unplanned boron dilution events. In DCD Section 9.3.6.7, "Instrumentation Requirements," a change is also made to the "makeup isolation valves" to state that the two makeup isolation valves automatically close on a signal from the protection and safety monitoring system derived from source range doubling high-2 pressurizer level, high steam generator level, or a safeguards signal coincident with high-1 pressurizer level. In DCD Revision 17, Section 9.3.6.4.5.1, "Boron Dilution Events," the CVS response to a boron dilution event is revised from "closing either one of two redundant safety-related, air-operated valves from the demineralized water system to the makeup pump suction" to "closing redundant safety-related valves, tripping the makeup pumps and/or aligning the suction of the makeup pumps to the boric acid tank." The description of dilution events during shutdown is also revised to state that "the source range flux doubling signal is used to isolate the makeup line to the RCS by closing the two safety-related, MOVs, isolate the line from the demineralized water system by closing the two safety-related, air-operated valves and trip the makeup pumps." This is a change from the statement in Revision 15 that "the source range flux doubling signal is used to isolate the line from the demineralized water system by closing the two safety-related, remotely operated valves. The three-way pump suction control valve aligns the makeup pumps to take suction from the boric acid tank and therefore stops the dilution."

In TR-80, Item II.9 "Flux Doubling/Boron Dilution Modifications," provides a rationale for the changes related to boron dilution events in DCD Section 9.3.6. The existing CVS design 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 begin to reach the RCS. The function is therefore changed to close the makeup line isolation valves (as well as the demineralized water isolation valves) to terminate the event as soon as possible. Long term recovery from the event would then be accomplished using either a different flow path with a smaller unpurged volume or by using the makeup line after purging most of the unborated water in it. The staff finds that the revised boron dilution events description is consistent with the modifications on the CVS makeup isolation valves in DCD Sections 9.3.6.3.7 and 9.3.6.7. This is also consistent with the boron dilution events described in Section 15.4.6.1 and Sections 15.4.6.2.2 through 15.4.6.2.5 for boron dilution events during Modes 5 through mode 2, respectively. Therefore, the staff finds the changes in the AP1000 DCD Sections 9.3.6.3.7, 9.3.6.4.5.1, and 9.3.6.7 are acceptable.

In DCD Section 9.3.6.6.1.2, "Flow Testing," the maximum makeup flow for the pump testing verification is changed from 757 Lpm to 662 Lpm (200 gpm to 175 gpm) with both pump operating. The change in the maximum makeup flow rate from 757 Lpm to 662 Lpm (200 gpm to 175 gpm) reflects the maximum flow rate through the cavitating venturi nozzle on the makeup pump discharge header. The CVS makeup pump testing is performed to verify that the maximum makeup flow with both makeup pumps operating is less than 662 Lpm (175 gpm). The staff finds that this maximum makeup flow rate is consistent with the assumptions of the boron dilution events analyzed in DCD Revision 17, Section 15.4.6, where the assumed unborated water flow rate is also changed from 757 Lpm to 662 Lpm (200 gpm to 175 gpm). The reduction of the maximum makeup flow to 662 Lpm (175 gpm) limits the unborated water flow rate to 662 Lpm (175 gpm) in the inadvertent boron dilution events and, thus, allows more time for isolation of the unborated water and termination of the events. Therefore, the staff finds that this change is acceptable.

In DCD Revision 17, Table 9.3.6-2, the following CVS nominal design parameters are changed: the letdown HX shell side and tube side design temperatures are changed from 93.3 °C (200 °F) and 343.3 °C (650 °F), respectively, to 65.5 °C (150 °F) and 315.5 °C (600 °F); the design flows of the mixed bed and cation bed demineralizers, and the reactor coolant filter are changed from 379 Lpm to 946 Lpm (100 gpm to 250 gpm); and the boric acid storage tank volume is changed from 264,979 L to 278,28 5 L (70,000 to 73,515 gallons). In RAI-SRP9.3.6-SRSB-02 the staff requested that the applicant provide the basis and justification for these changes to determine their acceptability. In its response dated January 26, 2009, the applicant provided the bases for these changes. The applicant states that the shell side temperature of the letdown HX cannot exceed 65.5 °C (150 °F) because shell side coolant is the CCS with a normal operating temperature of no higher than 37.8 °C (100 °F). The tube side temperature cannot exceed 315.5 °C (600 °F) as the purification flow is first cooled through the regenerative HX. The staff finds the tube side temperature acceptable because the purification flow is drawn from the RCS cold leg, which has a temperature lower than 315.5 °C (600 °F). The staff concludes that the changes of letdown HX shell side and tube side temperatures to 65.5 °C (150 °F) and 315.5 °C (600 °F), respectively, still maintain sufficient margin to the operating temperatures and are, therefore, acceptable.

The applicant states that the changes of the design flows from 379 Lpm to 946 Lpm (100 gpm to 250 gpm) for the demineralizers and the reactor coolant filters accommodate shutdown purification flows that could be as high as 810 Lpm (214 gpm) when the normal RNS provides the motive force for reactor coolant purification. The staff considers these design flow increases conservative changes that provide margin for the shutdown purification flow and are, therefore, acceptable. The applicant states that the boric acid storage tank increase in volume from 264,979 L to 278,285 L (70,000 to 73,515 gallons) represents the usable volume of the tank,

which includes the volume to accommodate a shutdown to cold shutdown followed by refueling at the end of the fuel cycle, plus the volume needed for normal operation and operating margin, and this increased volume is calculated with updated inputs that more accurately represents the AP1000 design. Because this more accurately reflects the design information, the staff finds this change acceptable.

In DCD Revision 15, the CVS demineralizer resin flush line containment isolation thermal relief valve (CVS-PL-V042) was located outside of the containment and discharged to the WLS waste holdup tank. In DCD Revision 17, Figure 9.3.6-1 is revised to relocate CVS-PL-V042 to inside containment between the two containment isolation valves. This relief valve is provided to prevent overpressurization of the resin sluice line that is used to sluice resin from the mixed bed and cation bed demineralizers to the waste processing system. The staff reviewed this change and finds that the location of the relief valve inside containment does not affect the functional capability of the CVS, but provides a protection against release of potential radioactive products outside containment. Therefore, the staff concludes that the proposed change is acceptable.

9.3.6.3 Conclusion

The staff reviewed the changes to DCD Section 9.3.6 regarding the AP1000 CVS design as described in DCD Revision 17. Based on the evaluation described above, the staff concludes that these changes would not adversely impact the required AP1000 CVS functions, and that the requirements of GDC 1, 2, 5, 14, 29, 33, 35, 60, and 61 continue to be met. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii); in that they contribute to increased standardization of the certification information.

9.4 Air Conditioning, Heating, Cooling, and Ventilation System

In NUREG-1793, the staff approved Section 9.4, "Air Conditioning, Heating, Cooling, and Ventilation System," of AP1000 DCD, Revision 15. In AP1000 DCD, Revision 17, the applicant proposed changes to this section, supported by TR-103.

The staff reviewed the changes to AP1000 DCD, Revision 15, Section 9.4, which are described in Revision 17. NUREG-1793 includes the regulatory basis for Section 9.4 of AP1000 DCD. The staff reviewed the proposed changes to DCD Section 9.4 against the applicable acceptance criteria in NUREG-0800 related to Section 9.4. Those changes that involve NRC review considerations as reflected in NUREG-0800 are described and evaluated in this section.

The specific criterion that applies to the changes evaluated in this section is 10 CFR 52.63(a)(1)(vii), which concerns the contribution to increased standardization of the certification information.

9.4.1 Nuclear Island Nonradioactive Ventilation System (VBS)

9.4.1.1 Summary of Technical Information

In Section 9.4.1.2.2, "Component Description," the applicant proposed to change the humidifier description such that the design and rating of the VBS Humidifiers will be in accordance with ARI 640, "Commercial and Industrial Humidifiers."

The DCD previously referenced standard ARI 620, "Self Contained Humidifiers for Residential Applications" in the design of the humidifiers. This was incorrect as ARI 620 states that the

intended application of the humidifiers is typically for non-ducted applications, and is independent of a central air system. The AP1000 uses humidifiers in ducted central air applications; therefore, ARI 640, "Commercial and Industrial Humidifiers" is the correct specification, and has replaced ARI 620. This correction applies to VAS, VBS, health physics and hot machine shop heating, ventilation, and air conditioning (HVAC) system (VHS), turbine building ventilation system (VTS), and annex/auxiliary buildings nonradioactive HVAC system (VXS) systems.

In Section 9.4.1.2.3.1, "Main Control Room/Control Support Area HVAC Subsystem," the applicant proposed various modifications to both the normal plant operation and abnormal plant operation sections.

In NUREG-1793, the staff approved a VBS HVAC system that has one heater in each air handler and one pair of temperature sensors to control the temperature in the control support area (CSA) and MCR areas. The airflow to each space is selected to properly cool each space at the summer design weather conditions. During winter conditions, cooling is required to maintain design conditions in some spaces, including the MCR, some electric/electronic equipment spaces, and the CSA computer rooms. Since the VBS system, as previously configured, could not heat some spaces and cool others simultaneously, additional heaters and temperature sensors have been added in the return ducting from the computer room for temperature control.

In Table 9.4-1 and Table 9.4.1-1, changes to the VBS leakage rates to MCR/CSA were proposed. The control logic depicted on the Figure 9.4.1-1 (sheet 4 of 7) of the DCD for the VBS fans serving the Class 1E Division B and D electrical rooms has been changed so that starting an air handling unit will start the chilled water system associated with that air handling unit. Previously, starting air handling unit MS03B sent a signal to the VWS to start chilled water pump MP02, which provides chilled water to air handling unit MS03D. In the same manner, starting air handling unit MS03D sent a signal to the VWS to start chilled water pump MP03, which provides chilled water to air handling unit MS03B. In both cases starting the air handling unit fails to start the correct water chiller and, therefore, the cooling system would not have operated correctly. The logic has been corrected so that starting air handling unit MS03B will start chilled water pump MP03 and starting air handling unit MS03D will start chilled water pump MP02; thus, starting each air handling unit will start the respective supporting pump.

9.4.1.2 Evaluation

The above mentioned DCD changes are technical improvements, corrections to design errors, or changes of facility descriptions. For the reasons discussed above, all of the changes are acceptable.

9.4.1.3 Conclusion

The NRC staff has reviewed the proposed changes to the VBS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VBS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.2 Annex/Auxiliary Buildings Nonradioactive HVAC System (VXS)

9.4.2.1 Summary of Technical Information

In Section 9.4.2.1.2, "Power Generation Design Basis," for the annex building nonradioactive HVAC system, there were some room rearrangements: Office areas, conference rooms, and security rooms and areas were added. A central alarm station and a security access area were deleted from the system and a security room in the mechanical equipment room was added.

In Section 9.4.2.2.1.1, "General Area HVAC Subsystem," the "VXS General Area HVAC Subsystem" was expanded. This expansion would add two more supply air handling units and other equipment to provide ventilated air to personnel areas in the annex building outside the security area.

In Section 9.4.2.2.2, "Component Description," the applicant changed the humidifier description such that the performance rating of the VAS Humidifiers will be in accordance with ARI 640.

9.4.2.2 Evaluation

The above mentioned DCD changes are technical improvements or changes of facility descriptions. For the reasons discussed above, all of the changes are acceptable.

9.4.2.3 Conclusion

The NRC staff has reviewed the proposed changes to the VXS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VXS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.3 Radiological Controlled Area Ventilation System (VAS)

9.4.3.1 Summary of Technical Information

In Section 9.4.3.2.2, "Component Description," the applicant proposed to change the humidifier description such that the performance rating of the VAS humidifiers would be in accordance with ARI 640 and Figure 9.4.3-1 would be revised accordingly.

9.4.3.2 Evaluation

The change to ARI 640 is evaluated in Section 9.4.1.

9.4.3.3 Conclusion

The NRC staff has reviewed the proposed changes to the VAS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VAS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.7 Containment Air Filtration System (VFS)

9.4.7.1 Summary of Technical Information

In Section 9.4.7.1.2, the applicant proposed to specify a VFS design pressure of "< -0.125 in. water gauge."

9.4.7.2 Evaluation

This change is evaluated in Section 16.4.12 of this report.

9.4.8 Radwaste Building HVAC System (VRS)

9.4.8.1 Summary of Technical Information

In Section 9.4.8.1.2, "Power Generation Design Basis," the applicant proposed to add "Truck Staging Area" to the rooms/areas covered by the VRS with a specified temperature range.

9.4.8.2 Evaluation

The above mentioned DCD change is an acceptable design improvement.

9.4.8.3 Conclusion

The NRC staff has reviewed the proposed changes to the VRS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VRS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.9 Turbine Building Ventilation System (VTS)

9.4.9.1 Summary of Technical Information

In Section 9.4.9.2.1.2, "Electrical Equipment and Personnel Work Area HVAC," the applicant proposed to provide details regarding the electrical equipment, south bay equipment, and personnel work area air conditioning subsystems including area temperature re-designation from 10.0 °C to 40.5 °C (50 °F to 105 °F) to 10.0 °C to 37.7 °C (50 °F to 100 °F). The following descriptive paragraph was also provided:

> The south bay equipment area HVAC system consists of two 50-percent capacity air handling units of about 7000 cfm capacity each. The air handling units are located on elevation 117'-6" of the turbine building between column 11 and 11.2. The temperature of the room is maintained by the thermostats that control the chilled water control valves for cooling and the integral face bypass dampers for heating. Outside air is mixed with the recirculation air to maintain a positive pressure.

In Section 9.4.9.2.2, "Component Description," the applicant proposed to change the humidifier description such that the design and rating of the VTS humidifiers will be in accordance with ARI 640.

9.4.9.2 Evaluation

The temperature change is acceptable because a lower maximum temperature is preferred for electrical equipment. The change to ARI 640 is evaluated in Section 9.4.1.

9.4.9.3 Conclusion

The NRC staff has reviewed the proposed changes to the VTS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VTS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.10 Diesel Generator Building Heating and Ventilation System (VZS)

9.4.10.1 Summary of Technical Information

In Section 9.4.10.2.1.1, "Normal Heating and Ventilation System," the applicant proposed to add the following paragraph:

> Electric unit heaters are provided in the diesel generator stairwell and security room to maintain the space at a minimum temperature.

9.4.10.2 Evaluation

The above mentioned DCD change is technically sound. The design change is acceptable.

9.4.10.3 Conclusion

The NRC staff has reviewed the proposed change to the VZS system and finds it acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VZS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.11 Health Physics and Hot Machine Shop HVAC System (VHS)

9.4.11.1 Summary of Technical Information

In Section 9.4.11.1.2, "Power Generation Design Basis," of the DCD the applicant proposed to add the "Security Room," the "Elevator Machine Room" and the "Stairwell" to the VHS ventilation system coverage rooms/areas with specified temperature ranges.

In Section 9.4.11.2.2, "Component Description," the applicant proposed to change the humidifier description such that the design and rating of the VHS humidifier will be in accordance with ARI 640.

9.4.11.2 Evaluation

The staff finds this change acceptable because the VHS ventilation system coverage has no safety significance. The change to ARI 640 is evaluated in Section 9.4.1.

9.4.11.3 Conclusion

The NRC staff has reviewed the proposed change to the VHS system and finds it acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VHS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

9.5.1.1 Summary of Technical Information

Section 9.5.1, "Fire Protection System" of the AP1000 DCD, Revision 15, was approved by the staff in NUREG-1793. In AP1000 DCD, Revision 17, the applicant has proposed to make the following change to Section 9.5.1:

1. In the AP1000 DCD, Revision 17, the applicant proposes to revise Section 9.5.1.8, alleviating the need for the COL applicant to submit additional information and to close out COL Action Items 9.5-5 and 9.5-8.

2. In Table 9.5.1-1, the applicant proposes to add carpeting into the MCR (e.g., for sound abatement or other human factors).

Multiple Spurious Actuations

Section 9A.2.7.1, "Criteria and Assumptions," of Appendix 9A, "Fire Protection Analysis," of the DCD, states the following with respect to the approach to evaluating multiple spurious actuations that result from fire-induced electrical shorts: "spurious actuations or signals resulting from the fire are postulated one-at-a-time (except for high/low pressure interfaces)." However, as noted in RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 1, the "one-at-a-time" assumption for spurious actuations may not adequately address the potential risk attributed to fire as demonstrated by NRC and industry fire tests.

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

9.5.1.2 Evaluation

In the AP1000 DCD, Revision 17, the applicant proposes to add carpeting into the MCR as allowed by RG 1.189, Regulatory Position 6.1.2. Although the DCD stated that the MCR carpeting issue is addressed in the fire protection analysis, the NRC staff noted that only the additional fire loading has been documented in the fire hazard analysis. Since the introduction of carpeting as proposed in DCD Revision 17 increases the fire duration in Fire Area 1242 AF 01 (MCR Complex) beyond 60 minutes, the current fire protection adequacy evaluation based primarily on the "light hazard" assumption for this fire area may no longer be valid. In addition, since the fire duration in Fire Zone 1242 AF 12401B increases to 75 minutes, which exceeds the 1-hour fire barrier rating between the two fire zones within the MRC Complex, the assumption that a fire is limited to one fire zone within the MCR Complex is no longer valid. Furthermore, while RG 1.189 references American Society for Testing and Materials (ASTM) D2859, "Standard Test Method for Flammability of Finished Textile Floor

Covering Materials," for establishing the acceptable flammability characteristics of the material, the applicant establishes compliance by referencing ASTM E-648, "Standard Test Method for Critical Radiant Flux of Floor-Covering Systems Using a Radiant Heat Energy Source," and National Fire Protection Association (NFPA) 253, "Standard Method of Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source," flame tests. In reviewing the above standards, the NRC staff cannot establish a direct correlation in the testing methods and acceptance criteria between ASTM D2859 and ASTM E-648 or NFPA 253.

In RAI-SRP9.5.1-SFPB-02, the staff requested that the applicant provide additional information to demonstrate that the alternative fire testing standards meet ASTM D2859 at the minimum. In addition, it was requested that the applicant revise the DCD and provide an evaluation that shows the additional fire loading does not impact the existing fire protection adequacy evaluation for the MCR Complex and it was requested that the applicant justify that the exclusion of a fixed fire suppression system in this fire area is still acceptable.

In a letter dated September 23, 2009, the applicant proposed several DCD revisions to address the information requested in RAI-SRP9.5.1-SFPB-02. The applicant included the testing standard used and revised Table 9.5.1-1 to indicate that the flammability characteristics of the carpeting are acceptable when tested to ASTM D2859. In addition, the applicant has stated that a revised combustible loading/fire severity calculation has been performed using a lower carpet quantity (weight per square foot), based on carpet vendor data, for carpeted areas and further reduced quantities to reflect areas not carpeted. Based on the new calculation, the applicant affirmed that the fire severity in the affected zone/area remains under the 1-hour fire duration, thus the "light hazard" assumption in the affected fire zone/area is maintained. Based on the above, the NRC staff determined that the applicant has adequately addressed the technical concerns in RAI-SRP9.5.1-SFPB-02. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Multiple Spurious Actuations

On February 19, 2009 and again on March 31, 2009, the NRC staff conducted audits of the applicant's fire hazards analysis supplemental report APP-FPS-G1R-002 and discussed the "one-at-a-time" assumption used in the report. The applicant stated that this assumption is not applied in supplemental report APP-FPS-G1R-002 or any other safe-shutdown analyses. The applicant further stated that by design, any unsafe plant conditions created by spurious equipment actuations (regardless of single or multiple spurious actuations) will be detected by the redundant safety instrumentation logics and will be mitigated by operators using the preferred safe-shutdown method or ultimately by the AP1000 passive safe-shutdown system. Since the applicant maintained that the "one-at-a-time" assumption is not used in the fire hazards analysis or any other safe-shutdown analysis, the NRC staff requested in RAI-SRP9.5.l-SFPB-01 that the "one-at-a-time" assumption be replaced with "multiple spurious actuations" assumption in applicable sections of the AP1000 DCD and that supplemental report APP-FPS-G1R-002 be referenced in the DCD. In a letter dated June 9, 2009, the applicant accepted the NRC staff's recommendation and proposed the above changes in the next revision of the DCD. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.5.1.2.1 Evaluation of COL Action Items

COL Action Item 9.5-5

In the AP1000 DCD, Revision 17, the applicant proposes to revise Section 9.5.1.8, alleviating the need for the COL applicant to submit additional information and to close out COL Action Item 9.5-5.

COL Action Item 9.5-5:

> The COL applicant will provide an analysis to demonstrate that operator actions that minimize the potential for spurious actuation of the automatic depressurization system (ADS) as a result of a fire can be accomplished within 30 minutes following detection of the fire, as well as the procedure for manual actuation of the fire water containment supply isolation valve to allow fire water to reach the automatic fire system in the containment maintenance floor.

This proposed change is supported by TR-45, "Operator Actions Minimizing Spurious ADS Actuation," APP-GW-GLR-027, Revision 1, dated June 2006, in which the applicant provided an analysis demonstrating that operator manual action can be accomplished within 30 minutes to minimize the potential for spurious automatic depressurization system (ADS) actuation in the event of a fire. The TR also stated that the procedure for the manual actuation of the valve to allow fire water to reach the automatic fire system in the containment maintenance floor has been written.

The ADS consists of four different stages of valves that are designed to open sequentially when actuated and to remain open for the duration of an automatic depressurization event. The sequential valves actuation logic relies on a combination of time delays and level indicators to prevent simultaneous opening of more than one stage at a time, and thus provides for a controlled depressurization of the RCS. Manual actuation switches are also provided in the MCR and on the remote shutdown console (RSC).

The applicant had previously asserted in the AP1000 DCD, and in subsequent responses to the staff's RAIs, that the inherent design of the ADS actuation logic, and the spatial separation of the manual activation switches in the MCR, made fire-induced spurious ADS actuation a highly unlikely event and therefore not a concern. However, the NRC staff has maintained that for certain plant areas, including the MCR, Division A Instrumentation and Control (I&C) Room, Division B I&C Room, Division C I&C Room, Division D I&C Room, Division A Penetration Room, and Division C Penetration Room, fire-induced spurious ADS actuation cannot be precluded due to the potential of multiple hot shorts and smoke-induced integrated circuit failures. Consequently, the applicant introduced post-fire operator actions as prescribed in TR45 to further minimize the probability of spurious ADS actuation in the above plant areas.

The staff initially had a concern regarding the feasibility and reliability of the proposed post-fire operator manual actions prescribed in TR-45 and issued RAI-TR45-006 in a letter dated March 30, 2007. In a letter dated July 13, 2007, the applicant responded to RAI-TR45-006 asserting that although spurious ADS actuation is undesirable, if ADS were to actuate as the result of a fire, the AP1000 plant would still be able to achieve and maintain post-fire safe shutdown. Also, the disabling of the ADS as proposed has no impact on the ability to achieve and maintain post-fire safe shutdown. Since the proposed post-fire operator manual actions are neither required for achieving and maintaining post-fire safe shutdown nor will create an

adverse impact on the post-fire safe-shutdown capability, the staff concludes that these actions do not have to meet the feasibility and reliability criteria outlined in RIS 2006-10, "Regulatory Expectations with Appendix R Paragraph III.G.2 Operator Manual Actions." Furthermore, these operator manual actions, aimed to minimize ADS actuations, have no adverse impact on the fire protection program.

In addition to providing the analysis supporting the minimization of spurious ADS actuation, TR-45 also provided that the applicant's document, APP-GW-GJP-305, "AP1000 Fire Emergency Response," includes steps for manual alignment of valves to allow fire water to reach the automatic fire system in the containment maintenance floor. This procedure, in addition to the above analysis, provided adequate information to satisfy COL Action Item 9.5-5.

COL Action Item 9.5-8

Section 9.5.1.8, "Special Protection Guidelines (Regulatory Position C.8 of BTP cmEB 9.5-1)" of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 17, the applicant proposes to revise Section 9.5.1.8, alleviating the need for the COL applicant to submit additional information and to close out COL Action Item 9.5-8. This proposed change is supported by TR-46, "Fire Resistance Test Data," APP-GW-GLR-019, Revision 0, dated May 2006, in which the applicant provided the fire resistance test data for the concrete/steel composite building material selected for use in certain firefighting and safe shutdown access/egress routes, in particular the stairwell towers within the Auxiliary Building. COL Action Item 9.5-8:

> The COL applicant will provide 2-hour fire resistance test data in accordance with American Society for Testing and Materials (ASTM) Standard E-119 and NFPA 51 for the composite material selected for stairwell fire barriers.

Regulatory Position C.5.a.6 of Branch Technical Position (BTP) cmEB 9.5.1 specifies that "Stairwells outside primary containment serving as escape routes, access routes, for firefighting, or access routes to areas containing equipment necessary for safe shutdown should be enclosed in masonry or concrete towers with a minimum fire rating of 2 hours and self-closing Class B fire doors." The AP1000 design, however, deviates from the above guideline by specifying concrete/steel composite material with an equivalent fire resistance rating of 2 hours in lieu of masonry or concrete for the auxiliary building stairway towers. The applicant, however, did not provide test reports to demonstrate that the designed concrete/steel composite configuration would meet the applicable regulation (GDC 3, "Fire Protection") and the applicable guidance (BTP cmEB 9.5-1) requiring an equivalent level of safety of concrete or masonry. The responsibility to provide test data was deferred to the COL applicant. Consequently, the NRC staff assigned COL Action Item 9.5-8 requiring the COL applicant to provide fire resistance test data in accordance with ASTM E-119, "Standard Test Method of Fire Tests of Building Construction and Materials," and NFPA 251, "Tests of Fire Endurance of Building Construction and Materials," for the concrete/steel composite material.

TR-46 supported the close-out of COL Action Item 9.5-8 by providing the fire resistance test data for the concrete/steel composite material. TR-46 confirmed that test results from Factory Mutual Research Corporation's Report J. I. 1R7Q3 successfully demonstrated at least a 2-hour fire resistance rating for the concrete/steel composite material as required by regulatory Position C.5.a.6 of BTP CMEB 9.5-1. This TR also confirmed that the fire resistance test was performed in accordance with ASTM E-119 and NFPA 251 as required per Section 3.1.6 of Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements."

Based on the above, the staff concluded that the applicant has provided adequate information to demonstrate the adequacy of the concrete/steel composite material to provide a fire resistance equivalent to that of a 2-hour rated concrete or masonry barrier as specified in Regulatory Position C.5.a.6 of BTP CMEB 9.5-1. Accordingly, COL Action Item 9.5-8 can be closed.

9.5.1.3 Conclusion

The NRC staff finds that the applicant has adequately addressed the staff's concern regarding the additional combustible loading/fire severity in the MCR Complex. The NRC staff also finds that the applicant's proposed DCD revision to replace the fire-induced "one-at-a-time" assumption with the "multiple spurious actuations" assumption is acceptable. The staff also concludes that the applicant has provided adequate information to close COL Action Item 9.5-5 and COL Action Item 9.5-8.

Based on the evaluation above, the staff concludes that the proposed changes are acceptable because the AP1000 design continues to meet the applicable requirements and guidance in GDC 3, BTP CMEB 9.5-1 and 10 CFR 50.48. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii), because they contribute to increased standardization of the certification information.

9.5.2 Communications Systems

9.5.2.1 Summary of Technical Information

The staff has reviewed the amendments to Section 9.5.2 of the AP1000 DCD, in accordance with the acceptance criteria in NUREG-0800 Section 9.5.2, "Communications System."

In Section 9.5.2.2.3, "Private Automatic Branch Exchange (PABX) System," the applicant has proposed to change the PABX interface to communications system requirements with the following modifications:

The applicant states that the hotlines to specified locations (for example, dedicated communication lines with load dispatcher to support and coordinate the system gird) are as described in Section 9.5.2.5, "Combined License Information." The hotline circuits are dedicated channels that provide direct communication between the MCR and the headquarters, or other facilities, are as specified in Section 9.5.2.5.

Direct extensions from the PABX locations exterior to the plant is as dictated by Section 9.5.2.5.

In addition, the applicant has proposed to modify the requirements for commercial telephone lines that are provided by the local area telephone company. Specifically, the applicant specifies that the number of lines will be defined as required in Section 9.5.2.5.

9.5.2.2 Evaluation

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

In RAI-SRP9.5.2-ICE-01, the staff requested additional information regarding the transfer of requirements from Section 9.5.2.2.3 to Section 9.5.2.5 for the COL applicant to address PABX

interfaces to hotlines and direct extensions from the PABX locations exterior to the plant. In the July 31, 2008, response to the staff's request for additional information, the applicant provided justification for these modifications. The applicant states that the changes made in Section 9.5.2.2.3 were administrative as described in TR-130, "Editorial Format Changes Related to Combined License Applicant and COL Information Items," APP-GW-GLR-130, Revision 0 of June 2007. TR-130 states:

"Up through Revision 15 of the DCD, there were many instances where the text states, "...the Combined License applicant will...." or words exist stating something similar. The applicant has taken the position that in the "Combined License Information" sections in Tier 2 of the DCD, these words are appropriate. Having the words appear in other sections of the DCD, however, leads to confusion, especially when the COL applicant is attempting to incorporate by reference the DCD section (or subsection) into their individual COL application. As a result, the applicant has reviewed the DCD and has removed these types of phrases from Tier 2... "In no case is it the intent of Westinghouse to change the commitment that exists in DCD, as the changes are intended to be editorial."

9.5.2.3 Conclusion

The applicant states that the requirements in Section 9.5.2.2.3 have not been modified or removed nor have they been transferred to Section 9.5.2.5. As delineated in Section 9.5.2.5, the COL applicant will describe how it meets the requirements specified in Section 9.5.2. The staff finds this response acceptable. The staff verified that the modifications made in Section 9.5.2.2.3 are administrative and do not affect the requirements within the COL action items described in Section 9.5.2.5. The staff concludes that the changes made in Revision 19 of the AP1000 DCD do not impact the staff's conclusions within Section 9.5.2, "Communication Systems" of NUREG-1793 regarding the AP1000 communication system's compliance. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii); on the basis that they contribute to increased standardization of the certification information and are, therefore, acceptable.

9.5.4 Diesel Generator Auxiliary Support Systems

9.5.4.1 Summary of Technical Information

Section 9.5.4, "Standby Diesel Fuel Oil System" of the AP1000 DCD, Revision 15, was approved by the staff in NUREG-1793. In AP1000 DCD, Revision 17, the applicant has proposed to make the following changes to Section 9.5.4 of the certified design:

1. The applicant has proposed to delete the function of supplying diesel fuel oil to the auxiliary boiler. In the previously approved AP1000 DCD, Revision 15, the standby diesel fuel oil system supplied fuel oil to the auxiliary boiler.

2. The applicant has proposed to complete COL Information Item 9.5-12 of Table 1.8-2, which is addressed in Section 9.5.4.7 of the AP1000 DCD Revision 15 and states:

   > Combined License applicants referencing the AP1000 certified design will address the site-specific need for cathodic protection in accordance with NACE Standard RP-01-69 for external metal surfaces of metal tanks in contact with the ground.

3. The applicant has proposed to partially complete COL Item 9.5-13 of Table 1.8-2, which is addressed in Section 9.5.4.7 of the AP1000 DCD Revision 15 and states:

> Combined License applicants referencing the AP1000 certified design will address site-specific factors in the fuel oil storage tank installation specification to reduce the effects of sun heat input into the stored fuel, the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations, and will address measures to protect against fuel degradation by a program of fuel sampling and testing.

9.5.4.2 Evaluation

The staff reviewed all changes to the Standby Diesel Fuel Oil System (SDFOS) in the AP1000 DCD Revision 17 in accordance with the guidance in NUREG-0800 Section 9.5.4, "Emergency Diesel Engine Fuel Oil Storage and Transfer System." The regulatory basis for Section 9.5.4 of the AP1000 DCD is documented in NUREG-1793. The following evaluation discusses the results of the staff's review.

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

9.5.4.2.1 Delete the Function of Supplying Diesel Fuel Oil to the Auxiliary Boiler

In DCD Revision 17, the applicant changed Tier 1 Section 2.3.3, ITAAC Table 2.3.3-2; and Tier 2 pages 9.5-24 through 9.5-28, Table 9.5.4-1, Table 9.5.4-2, and Figure 9.5.4-1. The applicant deleted the function of supplying diesel fuel oil to the auxiliary boiler. In TR-114, "AP1000 Auxiliary Boiler Sizing and Design," APP-GW-GLN-114, Revision 0 of June 2007, the applicant states that utilities have reported operational problems due to fouling of the fuel in diesel boilers that sit idle for extended periods of time. Some have changed their auxiliary boilers to electric boilers. For AP1000, the applicant has proposed a design change from a diesel fired auxiliary boiler to an electric auxiliary steam boiler. The fuel oil pumps and fuel oil piping associated with the auxiliary boiler have been removed from the DCD. The staff finds this change does not affect the function of the SDFOS to supply diesel fuel oil to the standby DGs. The staff also finds that the statements regarding the auxiliary boiler fuel oil supply in Section 9.5.9 of NUREG-1793 are no longer applicable. Since the changes to the SDFOS, which are the removal of the portion that would supply the auxiliary boiler, do not affect the function to supply fuel oil to the standby DGs, the staff finds that the conclusions of NUREG-1793 regarding the acceptability of the SDFOS remain valid.

9.5.4.2.2 Resolution of COL Item 9.5-12

In Revision 17 to the AP1000 DCD, the applicant proposed to resolve COL Information Item 9.5-12 which addresses cathodic protection of diesel fuel oil storage tanks. COL Information Item 9.5-12 in the DCD is also discussed in NUREG-1793 as COL Action Item 9.5.9-1. The applicant submitted TR-120, "Cathodic Protection for Metal Tanks in Contact With the Ground," APP-GW-GLR-120, of 25 May, 2007, for staff review to close out COL Information Item 9.5-12. The proposed change will eliminate the need for COL applicants to address cathodic protection of diesel fuel oil storage tanks as stated in COL Information Item 9.5-12.

In Revision 15, Section 9.5.4.7 to the AP1000 DCD, COL Information Item 9.5-12 states:

> Combined License applicants referencing the AP1000 certified design will address the site-specific need for cathodic protection in accordance with NACE Standard RP-01-69 for external metal surfaces of metal tanks in contact with the ground.

This COL item refers to National Association of Corrosion Engineers (NACE International) Standard Recommended Practice RP0169, "Control of External Corrosion on Underground or Submerged Metallic Piping Systems," which is referenced in RG 1.137, "Fuel-Oil Systems for Standby Diesel Generators." Since the diesel fuel tanks proposed for the AP1000 are on grade rather than buried, TR-120 proposes to address the COL item based on an alternative to NACE International RP0169.

In Revision 17 of the AP1000 DCD, the applicant proposed to resolve COL Information Item 9.5-12 by addressing the cathodic protection of diesel fuel oil storage tanks in TR-120, Section 9.5.4.7 of the DCD was revised to add Section 9.5.4.7.1, which states:

> The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-120 (Reference 24), and the applicable changes are incorporated into the DCD. No additional work is required.
>
> The following words represent the original COL Information item commitment, which has been addressed as discussed above:
>
> > Combined License applicants referencing the AP1000 certified design will address the site-specific need for cathodic protection in accordance with NACE Standard RP-01-69 for external metal surfaces of metal tanks in contact with the ground.

According to Section 9.5.4.2.2.1 of the DCD, the fuel oil storage tanks are located on grade and will be erected on a continuous concrete slab contained within a concrete dike. The tanks will have no direct contact with the soil, and the design is intended to minimize the intrusion of groundwater and rainwater into the interface between the tank bottom and concrete foundation.

In its report, the applicant explained that because NACE International RP0169 applies to underground tanks and the AP1000 diesel tanks are above ground, this COL item is being addressed according to guidance in American Petroleum Institute (API) Recommended Design Practice 651 (API 651), "Cathodic Protection of Aboveground Storage Tanks." The applicant quoted a portion of Section 5.3.3 in API 651 that indicates: (1) a properly designed concrete tank pad may be effective in eliminating external corrosion from the soil and the need for cathodic protection (CP); and (2) moisture may still collect between the tank bottom and pad, and CP is generally not an effective corrosion control method under these conditions. The corrosion protection proposed for these tanks is, therefore, based on a combination of keeping the tank bottom dry and coating it with an appropriate epoxy-urethane paint system.

Because cathodic protection of a structure relies on the flow of electrical charge (i.e., current) from external electrodes (anodes) to the structure (cathode), successful application of CP requires an electrolyte path between the anodes and the structure. (See, for example, "Cathodic Protection," in Metals Handbook, Ninth Edition, Volume 13, ASM International, 1987.) For external protection, CP is normally used in a soil or aqueous environment in conjunction

with an external coating on the structure; hence, current is needed only at coating defects that expose the underlying metal. For the AP1000 diesel tanks, the staff concludes CP would not be useful under the design conditions because the tank bottoms are expected to be dry. Any anticipated moisture accumulation causing partial or intermittent wetness would not be expected to cause significant corrosion of tanks utilizing a well designed and maintained coating system.

The staff notes that another NACE International Standard Recommended Practice, RP0193, "External Cathodic Protection of On-Grade Carbon Steel Storage Tank Bottoms," provides guidance similar to API 651:

> 5.7 On-grade tanks that are set on solid concrete or asphalt pad foundations generally require specialized measures for corrosion protection, because cathodic protection may be ineffective. In this circumstance, the external surface of the tank bottom should be coated. In all cases, steps should be taken to ensure that water does not migrate between the tank bottom and the pad.

The industry recommended practices cited by the applicant and by the staff indicate cathodic protection is not required for the on-grade diesel fuel storage tanks proposed for the AP1000, and the proposed resolution of COL Information Item 9.5-12 is, therefore, technically sound. Therefore, the staff found that the DCD changes, as proposed by the applicant in TR-120, are acceptable, and COL Information Item 9.5-12 is resolved. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes incorporated into Revision 17 contribute to the increased standardization of the certification information in the AP1000 DCD.

The staff notes that the standby DGs and their support systems (e.g., fuel storage tanks) have no safety-related functions in the AP1000 passive plant design and, therefore, no nuclear-safety design basis. Hence, storage tank design features discussed above are not safety significant. NUREG-1793, which was written based on Revision 14 of the AP1000 DCD, stated this conclusion as follows:

> Based on its review, the staff determined that the DG and auxiliary boiler fuel oil system is a non-safety-related system and serves no safety-related function. Its failure does not lead to the failure of any safety systems. The staff, therefore, concludes that the requirements of GDC 2, 4, 5, and 17, and the guidance of SRP Section 9.5.4, do not apply.

9.5.4.2.3 COL Item 9.5-13

In Revision 17 to the AP1000 DCD, the applicant proposed to resolve the portion of COL Information Item 9.5-13, that addresses the need to reduce heat input to the fuel oil storage tanks from the sun. COL Information Item 9.5-13 in the DCD is also discussed in NUREG-1793 as COL Action Item 9.5.9-2, which is associated with ensuring the quality of diesel fuel oil. The applicant submitted TR-120 for staff review to close out COL Information Item 9.5-13. The proposed change will eliminate the need for COL applicants to address the site-specific requirement for reducing heat input to the fuel oil storage tanks from the sun. With respect to the other two parts of COL Information Item 9.5-13, the proposal is clear that addressing the fuel specifications grade, properties, and sampling/testing program remains the responsibility of the COL applicant.

In Revision 15, Section 9.5.4.7 to the AP1000 DCD, COL Information Item 9.5-13 states:

> Combined License applicants referencing the AP1000 certified design will address site-specific factors in the fuel oil storage tank installation specification to reduce the effects of sun heat input into the stored fuel, the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations, and will address measures to protect against fuel degradation by a program of fuel sampling and testing.

COL Information Item 9.5-13 addresses three separate issues: 1) reducing heat input to the tank from the sun; 2) specifying the proper fuel grade and properties based on the engine manufacturer recommendations; and 3) protecting against fuel degradation with a fuel sampling and testing program. The applicant proposes to partially resolve this COL item by closing out that portion of the COL item related to reducing heat input to the tank from the sun. COL Information Item 9.5-13 will remain open and COL applicants are required to specify the proper fuel grade and properties based on the engine manufacturer's recommendations, and provide measures to protect against fuel degradation with a fuel sampling and testing program.

In Revision 17 of the AP1000 DCD, the applicant proposed to partially resolve COL Information Item 9.5-13 by addressing heat input to the tank from the sun in TR-120. Section 9.5.4.7 and Table 1.8-2 of the DCD was revised to add Section 9.5.4.7.2, which states:

> 9.5.4.7.2 The Combined License information requested in this subsection has been partially addressed in APP-GW-GLR-120, (Reference 24), and the applicable changes are incorporated into the DCD. No additional work is required to address the information requested in this subsection as delineated in the following paragraph:
>
> > The epoxy-urethane paint color selected for the exterior of the standby diesel fuel oil storage tanks shall be white to minimize radiant sunlight heat transmission to the tank oil stored fuel volume.
>
> The following activities are to be addressed by the Combined License applicant:
>
> > Address the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations and the measures to protect against fuel degradation by a program of fuel sampling and testing.
>
> The following words represent the original Combined License Information item commitment, which has been addressed as discussed above:
>
> > Combined License applicants referencing the AP1000 certified design will address site-specific factors in the fuel oil storage tank installation specification to reduce the effects of sun heat input into the stored fuel, the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations, and will address measures to protect against fuel degradation by a program of fuel sampling and testing.

Heat input to the tank is addressed by stating that the color of the epoxy-urethane exterior coating will be white to minimize heat input to the tank contents. This approach is technically sound. For example, it is consistent with ASTM D-975, "Standard Specification for Diesel Fuel Oils," – 07b, Section X3.7, which states reflective paint color should be used to prevent thermal degradation of the fuel oil.

With respect to the other two parts of COL Information Item 9.5-13, the proposal is clear that addressing the fuel specifications grade, properties, and sampling/testing program remains the responsibility of the COL applicant.

Based on the above, the staff finds that the portion of COL Information Item 9.5-13 that addresses reducing heat input to the tank from the sun is resolved. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design.

9.5.4.3 Conclusion

Based on the above evaluation, the staff finds the applicant's proposed design change from a diesel fired auxiliary boiler to an electric auxiliary steam boiler does not affect the function of the SDFOS to supply diesel fuel oil to the standby DGs and is, therefore, acceptable. The staff also finds that the applicant's proposed resolution to COL Information Item 9.5-12 is consistent with applicable industry guidance and is, therefore, acceptable. The staff also finds that the applicant's proposed partial resolution to COL Information Item 9.5-13 for reducing heat input to the tank from the sun is consistent with applicable industry guidance and is acceptable. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to increased standardization of the certification information.

10. STEAM AND POWER CONVERSION SYSTEMS

10.1 Introduction

Westinghouse Electric Company LLC (Westinghouse or the applicant) proposed to the U.S. Nuclear Regulatory Commission (NRC) an alternative design for its steam and power conversion (SPC) system in its technical report (TR)-86, "APP-GW-GLN-018, Revision 0, "Alternate Steam & Power Conversion Design," dated February 8, 2007. On June 25, 2007, the applicant submitted Revision 1 to TR-86 and proposed a new standard design with a single turbine-generator and steam-cycle unit for all AP1000 plants. In Revision 17 to the AP1000 design control document (DCD), Tier 2, Section 10.2, "Turbine Generator," the applicant incorporated all the changes associated with the new standard design.

The most significant differences between the designs that the NRC staff evaluated in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004, and the proposed new design of the main turbine-generator set are as follows:

(1) replacement of the mechanical overspeed protection with a diverse electrical overspeed trip;

(2) triplicate channels for turbine speed indication and turbine trip signal versus two redundant channels;

(3) three modes of turbine overspeed protection versus two modes;

(4) hydraulic trip manifold, which enables online testing;

(5) addition of 7th stage feedwater heaters;

(6) decrease in diameter from 137 centimeters (cm) (54 inch (in)) to 132 cm (52 in) for the low-pressure turbine last stage blade (LSB);

(7) addition of moisture extraction blades (MEB); and

(8) static excitation provided by solid-state thyristors versus brushless excitation.

In the AP1000 DCD, Tier 1, Section 2.4.2, Table 2.4.2-2, the applicant proposed adding an "emergency electrical overspeed trip device" as a component located in the turbine building.

In addition, AP1000 DCD, Section 10.2, includes the following proposed changes:

- In DCD Tier 2, Section 10.2.6, "Combined License Information on Turbine Maintenance and Inspection," the applicant proposed revising the implementation timing for combined license (COL) Information Item 10.2.6. This was identified as COL Action Item 10.5-2 in Section 10.5 of Appendix F, "Combined License Action Items," to NUREG-1793.

- In DCD Tier 2, Section 10.2.3.6, "Maintenance and Inspection Program Plan," the applicant proposed revising the turbine valve testing intervals from quarterly to semiannually.

- In DCD Tier 2, Section 10.2, the applicant proposed a number of changes related to the layout and general arrangement in the turbine building as a result of replacing the original design with the alternative design, which includes an additional stage of feedwater heaters.

- In DCD Tier 2, Section 10.2.2.4, "Digital Hydraulic System Description"; Section 10.2.2.5, "Overspeed Protection"; and Table 10.2-2, "Turbine Overspeed Protection"; the applicant proposed revisions to reflect the replacement of the Toshiba Turbine Control System with an Ovation® Turbine Control System.

- In DCD Tier 2, Figure 10.2-1, "Turbine Generator Outline Drawing," the applicant revised the drawing to provide consistency with the Toshiba Turbine-Generator Steam Cycle Design.

All other changes to Section 10.2 were determined by the staff to be editorial in nature and do not affect the conclusion in NUREG-1793 and are, therefore, acceptable.

10.2 Turbine Generator

The staff reviewed all changes to the turbine generator, in accordance with Section 10.2, "Turbine Generator," of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." The new design of the turbine-generator system was evaluated against the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities." Specifically, the design must meet the requirements of General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, as they relate to protecting the structures, systems, and components (SSCs) that are important to safety from the effects of turbine missiles. GDC 4 provides guidance for the turbine overspeed protection system (with suitable redundancy and diversity) to minimize the generation of turbine missiles. NUREG-0800 Section 10.2.II, "Acceptance Criteria," describes the specific criteria needed to meet the requirements of GDC 4. Also, the staff used NUREG-0800 Section 10.2.III, "Review Procedures," to determine whether the preliminary design in the DCD meets the acceptance criteria in NUREG-0800 Section 10.2.II.

The staff focused its review on changes 1 through 4 identified above, COL Action Item 10.5-2, turbine valve testing, and the general layout arrangements, since these changes affect the protection of the turbine from overspeed conditions that could lead to the generation of turbine missiles. These items relate directly to the specific criteria described in Items 1 A and 1 C of NUREG-0800 Section 10.2.II that are necessary to meet the requirements of GDC 4, as they relate to the turbine overspeed protection and periodic testing of components while the unit is operating at rated load. Also, the staff reviewed changes 5 through 8 identified above and determined that these changes do not affect the safety conclusions in NUREG-1793.

For the purpose of this evaluation, the NRC refers to the design of the turbine-generator system provided in the AP1000 DCD, Revision 15, and evaluated in Section 10.2 of NUREG-1793, as

the "original" design. The staff refers to the alternative Toshiba design in the proposed changes to the AP1000 DCD as the "new" design.

The safety design bases and power generation design bases for the turbine generator are the same for both the original and new designs. The turbine generator of the new design is designated as Model TC6F, with a diameter of 132 cm (52 in) for the low-pressure turbine LSB unit. The low-pressure turbine LSB in the original design had a diameter of 137 cm (54 in). DCD Tier 2, Table 10.2-1, identifies the design parameters of the new design. DCD Tier 2, Figure 10.3.2-2, provides the piping and instrumentation diagram containing the stop, control, intercept, and reheat valves. Also, the new design consists of a double-flow, high-pressure turbine; three double-flow, low-pressure turbines; and two external moisture separator/reheaters. The difference between the two designs is that the new design has two reheating stages, while the original design had only one.

The single direct-driven generator is cooled by hydrogen gas and de-ionized water (water-cooled in the original design). Other related system components include a turbine-generator-bearing lubrication system, a digital electrohydraulic control (D-EHC) system with supervisory instrumentation, a turbine steam-sealing system, overspeed protective devices, turning gear, a stator cooling-water system, a generator hydrogen and seal oil system, a generator carbon-dioxide system, a rectifier section, an excitation transformer, and a voltage regulator. The D-EHC system, the overspeed protective devices, and the excitation system in the new design differ from those in the original.

The turbine-generation foundation in both designs is a spring-mounted support system. The springs dynamically isolate the turbine-generator deck from the remainder of the structure in the range of operating frequencies.

The turbine-generator cycle is basically the same for both designs. Steam from each of the two steam generators enters the high-pressure turbine through stop valves and control valves (CVs). After expanding through the high-pressure turbine, exhaust steam flows through two external moisture separator/reheater vessels. The reheated steam flows through separate reheat stop and intercept valves in each of six reheat steamlines leading to the inlets of the three low-pressure turbines. Turbine steam extraction connections are provided for seven stages of feedwater heating (six stages in the original design). Moisture separation in the new design differs from the original; however, the difference does not affect the evaluation and the staff's conclusion in NUREG-1793.

10.2.1 Overspeed Protection

A D-EHC system and an emergency trip system provided overspeed protection in the original design. The D-EHC system opened a drainpath for the hydraulic fluid in the overspeed protection control header, if the turbine exceeded 103 percent of rated speed. As a result of the loss of fluid pressure in the header, the control and intercept valves closed.

The emergency trip system tripped the turbine if speeds exceeded 110 percent of the rated speed. The emergency trip system has an emergency trip control block, trip solenoid valves, a mechanical overspeed device, three test trip blocks with pressure sensors and test solenoid valves, rotor position pickups, speed sensors, and a test panel. The mechanical overspeed trip device consisted of a spring-loaded trip weight mounted in the rotor extension shaft.

In the new design, the two systems that provide overspeed protection, the D-EHC control system and the electrical overspeed trip system, differ from those in the original.

The new D-EHC has three modes of operation to protect the turbine generator against overspeed. The first is the intercept-valves control function, which initiates closure of the intercept valves when the error between the demand position signal and the actual position signal of the intercept valve exceeds the setpoint. The second mode is load unbalance, which operates at greater than or equal to 30-percent load rejection. It causes all control and intercept valves to close quickly. The third mode is the emergency overspeed trip. All CVs and intercept valves are fully closed quickly by the actuation of each fast-acting solenoid valve of the electrical overspeed trip system.

The electrical overspeed trip system consists of redundant processors, three speed channel circuits, and trip relays. An independent electrical overspeed system replaces the mechanical overspeed device of the original design. A trip of the system opens a drainpath for the hydraulic fluid in the emergency trip supply. A loss of fluid pressure in the trip header causes the main stop and reheat valves to close. Also, a relay trip valve in the connection to the emergency trip supply opens to drop the pressure in the relay emergency trip supply and cause the control and intercept valves to close.

The specific criteria that apply to the overspeed protection related changes include 10 CFR 52.63(a)(1)(vi), "Finality of standard design certifications," in that the changes substantially increase overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

The following Sections 10.2.2 and 10.2.2.1 provide further details on the D-EHC and the electrical overspeed trip systems, respectively.

10.2.2 Digital Electrohydraulic Control System

AP1000 DCD, Section 10.2.2.4, "Digital Electrohydraulic System Description," states that the turbine generator is equipped with a D-EHC system that combines the capabilities of redundant processors and high-pressure hydraulics to regulate steamflow through the turbine. The system provides the functions of speed control, load control, and automatic turbine control. A hydraulic system, independent of the bearing lubrication system, provides valve opening actuation in the D-EHC system. Upon reduction or relief of fluid pressure, springs provide valve-closing actuation. The system is designed so that loss of fluid pressure, for any reason, leads to valve closing and, consequently, a turbine trip. Steam valves are provided in an in-line configuration. The overspeed trip system trips the stop valves; the CVs are modulated by the control system and are actuated by the trip system.

The speed control function of the D-EHC master controller provides speed control, acceleration, and overspeed protection. The speed error signal is derived by comparing the desired setpoint with the actual speed of the turbine. A failure of one speed input generates an alarm. The failure of two or more speed inputs generates an alarm and trips the turbine.

The speed control function exists in triplicate channels. The master controller speed function also contains 110-percent and 111-percent overspeed trips. The 110-percent trip signal is sent to a fast-acting solenoid valve in the hydraulic trip manifold that actuates closure of the stop,

control, intercept, and reheat valves. An independent emergency electrical trip is available at 111-percent turbine speed, as a backup to the 110-percent electrical overspeed trip.

The load control function of the D-EHC develops signals that are used to regulate unit load. Signal inputs are based on a proper combination of the speed error and actual load setpoints (turbine megawatt) to generate flow demand on the CVs.

The automatic turbine control provides start up and loading of the turbine generator. The automatic turbine control programs monitor the applicable limits and precautions. When the operator selects the automatic turbine control, the programs both monitor and control the turbine. The automatic turbine control is capable of performing the following activities:

- changing speed
- changing acceleration
- generating speed holds
- changing load rates
- generating load holds

The staff's evaluation finds that the D-EHC system combines the capabilities of redundant processors and high-pressure hydraulics to regulate steamflow through the turbine. The control system provides the functions of speed control, load control, and the automatic turbine control, which may be used either for control or for supervisory purposes, at the option of the plant operator.

The D-EHC system employs three electric speed inputs with signals that are processed in redundant processors. A hydraulic system that is independent of the bearing lubrication system provides valve-opening actuation. Upon reduction or relief of fluid pressure, springs and steam forces provide valve-closing actuation. The system is designed so that a loss of fluid pressure, for any reason, leads to valve closing and consequent turbine trip.

The staff discusses the acceptability of the redundancy and diversity functions of the new D-EHC system design below.

The overspeed trips for the turbine consist of a 110-percent electrical trip and a 111-percent independent emergency electrical trip. In DCD Tier 2, Section 10.2.2.4.1, the applicant stated that the independent emergency electrical trip was available at 111-percent turbine speed, as a backup to the 110-percent electrical overspeed trip.

The electrical overspeed trip system includes an online testable hydraulic manifold, speed sensors, a trip relay, independent power supplies, and a test panel. The emergency trip supply pressure is established when the master trip solenoid valves are closed. The valves are arranged in two channels for testing purposes. Both valves in a channel will open to trip the channel. Both channels must trip before the emergency trip supply pressure collapses to close the turbine steam inlet valves. Each tripping function of the electrical emergency trip system can be individually tested either by the operator or from the test panel, without tripping the turbine, by separately testing each channel of the appropriate trip function. The solenoid valves may be individually tested.

The 110-percent electrical overspeed trip system has triplicated (i.e., redundant) speed sensors, separate from the 111-percent emergency electrical overspeed trip speed sensors,

that provide backup overspeed protection, and use the master trip solenoid valves in the master trip device to drain the emergency trip hydraulic supply. The hydraulic fluid in the trip and overspeed protection control headers is independent of the bearing lubrication system to minimize the potential for contamination of the fluid.

Separate instruments sense low bearing oil pressure, low electrohydraulic fluid pressure, and high condenser back pressure. Each assembly consists of triplicate pressure transmitters with instrument valves. Each assembly is arranged into three channels. If two of the three signals (pressure or vacuum) reach a trip setpoint, then the pressure sensors cause the master trip device to operate. A test device can check the trip function by simulating pressure to activate the trip outputs from the modules.

The mechanical overspeed device, evaluated in NUREG-1793, has been replaced with an independent, diverse electrical overspeed trip system. The electrical overspeed trip system consists of redundant processors, three speed channel circuits, and trip relays, as well as other protective functions, such as trip anticipations and power load unbalance detection. The overspeed controllers execute offline and online testing, both of which are conducted from the control room and require no technicians at the turbine. Offline testing is performed during startup and trips the turbine, based on an internal setpoint rather than actual turbine speed. Online testing is automatically executed through the internal injection of a ramping signal into all three independent speed channels at once. This test verifies the proper operation of the software, the hardware, and the components of the hydraulic trip manifold. Loss of one signal will neither cause nor prevent a trip; however, failure of two of the three channels will initiate a turbine trip.

Regarding redundancy, Westinghouse stated that its design achieves additional redundancy by using three speed channel circuits in the master controller of the D-EHC turbine control system. The master controller is the primary controller for starting, synchronizing, and megawatt-loading the turbine/generator. This emergency overspeed trip system residing in the master controller is redundant, independent, and diverse to the electrical overspeed trip system, as discussed above. Each speed channel also has its own independent tripping relays. Loss of one signal will neither cause nor prevent a trip. Further, Section 3.0, "Technical Background," of TR-86, Revision 1, states: "Both the electrical overspeed trip system and the emergency overspeed trip systems in the master controller meet the single-failure criterion and are testable when the turbine is in operation." However, the staff's review of the DCD markup could not find any commitment for either the electrical overspeed trip system or the emergency overspeed trip system in the master controller to meet the single-failure criterion. Item 1.A of NUREG-0800 Section 10.2 states that the overspeed protection system should meet the single-failure criterion and should be testable when the turbine is in operation. Therefore, in request for additional information (RAI)-SRP10.2-SBPA-01, the staff asked the applicant to provide additional information and justification for its claim that its design meets the NUREG-0800 guidance for the single-failure criterion.

In its response to RAI-SRP10.2-SBPA-01, in a letter dated June 20, 2008, the applicant stated that the detailed design of the emergency overspeed trip system for AP1000 is still being completed and that the overspeed protection system design will meet the NUREG-0800 guidance for the single-failure criterion. The applicant is to provide a date for the completed design. The staff had identified this as Open Item OI-SRP10.2-SBPA-01.

The applicant addressed the single failure criteria in its response to RAI-SRP10.2-SBPA-02, dated June 12, 2009. The response included a mark-up of DCD Tier 2, Section 10.2.2.5.3

stating that the overspeed protection system will function for abnormal conditions, including a single failure of any component or subsystem. The staff finds the DCD mark-up meets the NUREG-0800 guidance and is acceptable. Therefore, Open Item OI-SRP 10.2-SBPA-01 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.

With respect to the diversity of overspeed protection, in an earlier request, in Item 3 of RAI-TR86-SBPB-01, the staff asked the applicant to compare the reliability of the proposed turbine overspeed protection capability to the reliability that is afforded by the diverse capability for existing plants. In its letter response dated July 27, 2007, the applicant stated: "Another degree of diversity is provided by the software based trip that takes the speed reading from the input/output (I/O) modules and applies control builder logic to determine the trip function which is then output via separate relay modules." The applicant's response did not specify whether this applies to the primary overspeed trip of 110 percent, or the emergency backup overspeed trip of 111 percent, or both. The staff could not find any further details on the software configuration for the overspeed trip system, either in the changes to the DCD identified in TR-86 or in the responses to other RAIs. The staff's concern is that, if both the 110-percent and 111-percent overspeed trips use the same software, a common-cause failure (CCF) could render both systems inoperable. Also, Item 2.A of NUREG-0800 Section III states that the defense-in-depth provided by the turbine-generator protection system to preclude excessive overspeeds should include diverse protection means. Therefore, with respect to diverse protection against CCF in its design, the staff asked the applicant, in RAI-SRP10.2-SBPA-02, to provide further details to show that it meets the NUREG-0800 guidance, as described above.

In response to RAI-SRP10.2-SBPA-02, the applicant stated that the original design approach was to use Ovation® speed detector module firmware for both trips in parallel with an Ovation® controller software-based logic that provides a level of redundancy and diversity. Now, the applicant has committed to implementing the two overspeed trips using diverse (hardware and software/firmware) electronic means (i.e., one of the trips will not be implemented using the Ovation® speed detector module), so that the 110-percent and 111-percent trips are not susceptible to a common-cause software failure that would render them both inoperable. The staff finds that this commitment meets the guidance described in Item 2.A of NUREG-0800 Section 10.2.III, which states: "The design of the in-depth defense provided by the turbine-generator protection system to preclude excessive overspeeds should include diverse protection means." The staff finds this acceptable; however, the staff's position is that the applicant should update the Tier 1 and Tier 2 sections of the DCD, with inspections, tests, analyses, and acceptance criteria (ITAAC), to confirm that the design acceptance criteria requiring diverse hardware, firmware, and software between the two overspeed trips are met. The staff had identified this as Open Item OI-SRP10.2-SBPA-02a.

Revision 3 of the applicant's response to RAI-SRP10.2-SBPA-02a, dated December 28, 2009, addresses Open Item OI-SRP10.2-SBPA-02a. The applicant provided a mark-up of DCD Tier 1 Table 2.4.2-1 (ITAAC) where Design Commitment 3 has been added, which addresses the applicant's commitment to provide adequate diversity between the two electrical overspeed trips. Design Commitment 3 states: "The trip signals from the two turbine electrical overspeed protection trip systems within the PLS are isolated from, and independent of, each other." For the inspections, tests, and analyses of this ITAAC, the applicant identified three subparts, which include: i) the system design review; ii) testing of the as-built system; and iii) an inspection to be performed for the existence of a report verifying that the two turbine electrical overspeed protection systems have diverse hardware and software/firmware. Further, the applicant described an associated acceptance criteria for each of these inspections, tests, and analyses,

where the Acceptance Criteria 3.iii stating: "A report exists and concludes that the two electrical overspeed protection systems within the PLS have diverse hardware and software/firmware." The staff finds this ITAAC acceptable since it ensures that the two electrical overspeed protection systems consist of diverse hardware and software/firmware.

Additionally, in the same December 28, 2009 response, the applicant provided a mark-up of DCD Tier 2, Section 10.2.2.5.3, where it states that the 110 percent and 111 percent trip systems have diverse hardware and software/firmware to eliminate CCFs from rendering the trip functions inoperable. Also in the response, the applicant added Figure 10.2-2, "Emergency Trip System Functional Diagram," to the end of DCD Tier 2, Section 10.2, which shows schematically the diverse hardware and software/firmware of the overspeed trip systems. The staff finds the DCD Tier 1 and Tier 2 mark-ups and added Tier 2 Figure 10.2-2 acceptable as they meet the guidance in Item 2.A of NUREG-0800 Section 10.2.III. Therefore, Open Item OI-SRP 10.2-SBPA-02a 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.

Also, in NUREG-0800 Section 10.2.III, Item 2.D, the guidance states: "An independent and redundant backup electrical overspeed trip circuit senses the turbine speed by magnetic pickup and closes all valves associated with speed control at approximately 112 percent of rated speed." However, the applicant's response to the above RAI does not state whether the electrical backup senses the turbine speed by magnetic pickup. The staff had identified this as Open Item OI-SRP10.2-SBPA-02b.

In its response to RAI-SRP10.2-SBPA-02, dated June 12, 2009, the applicant provided a mark-up of DCD Tier 2, Section 10.2.2.5.3 where it is stated that an independent and redundant backup electrical overspeed trip circuit senses the turbine speed by magnetic pickup and closes all valves associated with speed control at approximately 111 percent of rated speed. The staff finds the DCD mark-up acceptable as it meets the guidance in NUREG-0800 Section 10.2.III-2c. Therefore, Open Item OI-SRP 10.2-SBPA-02b 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.

In RAI-SRP10.2-SBPA-04, the staff asked the applicant to provide a timeframe for completion of the system design and submittal of an analysis of failure modes and effects; specifically, common-mode failures of the software for the 110-percent and 111-percent overspeed protection features. In response to RAI-SRP10.2-SBPA-04, the applicant stated that, based on its response to RAI-SRP10.2-SBPA-02 on the diversity of the two turbine overspeed trips provided for the 110-percent and the 111-percent speed points, an analysis would not be necessary; the staff finds this acceptable.

Further, NUREG-0800 Section 10.2.III, Item 2.B, states: "For normal speed-load control, the speed governor of the electrohydraulic control system fully cuts off steam at approximately 103 percent of rated turbine speed by closing the control and intercept valves." The original design included this provision, as shown in Table 10.2-2 in Revision 15 of the AP1000 DCD. However, for the new design, the applicant eliminated the 103-percent trip without providing any justification in the revised DCD or in TR-86, Revision 1. Therefore, in RAI-SRP10.2-SBPA-03, the staff asked the applicant to justify the elimination of the 103-percent trip.

In response to RAI-SRP10.2-SBPA-03, in a letter dated July 31, 2008, the applicant stated that the 103-percent trip value previously provided in Table 10.2-2 in Revision 15 of the DCD was

not for a turbine-trip condition. The described condition was for the speed-control mode of the turbine-control system for a load reject event and a generator breaker open condition. The applicant eliminated the 103-percent value from DCD Table 10.2-2 because the Toshiba turbine valves and hydraulic system are not designed to fully close at the 103-percent rated speed. The CVs are closed at approximately 105 percent of rated turbine speed, and the intercept valves are closed at approximately 107 percent of rated speed. However, before these overspeed points are reached, the CVs and intercept valves begin to close at 101 percent of the rated turbine speed, as stated in DCD Table 10.2-2. The system is designed to prevent the peak transient of 108 percent of rated speed from being exceeded, as stated in the table. The applicant stated that this had not changed from DCD Revision 15—as speed is reduced, the valves will reopen and modulate as needed to achieve and maintain 100-percent rated speed. The staff accepts the applicant's response to RAI-SRP10.2-SBPA-03, for the reasons explained above, as well as the applicant's explanation that it had eliminated the 103-percent value from DCD Table 10.2-2 because of the new Toshiba design.

In the changes to Tier 1 Section 2.4.2, Table 2.4.2-2 of the AP1000 DCD, the applicant added the emergency electrical overspeed trip device as a component located in the turbine building. This component, in conjunction with the electrical overspeed trip device (which is already listed in Tier 1 Table 2.4.2-2) provides the diversity and redundancy for the turbine overspeed trip function that are specified by NUREG-0800 Section 10.2. The staff agrees that the emergency electrical overspeed trip device is an important design feature that should be included in the Tier 1 design description for the main turbine system. Therefore, the addition of the emergency electrical overspeed trip device as a component located in the turbine building in Tier 1 Table 2.4.2-2 is considered to be acceptable.

10.2.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for design certification (DC) met the requirements of Subpart B, "Standard Design Certifications," to 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the changes the applicant proposed to the AP1000 turbine generator in the AP1000 DCD. The staff finds that incorporating the proposed changes does not adversely affect the ability of the turbine generator to meet the applicable acceptance criteria in NUREG-0800 Section 10.2. The staff also finds that the applicant has properly incorporated the design changes into the appropriate sections of the AP1000 DCD. Because the AP1000 turbine-generator design continues to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 turbine generator are acceptable.

In addition, the changes establish the proposed Toshiba design as the single, standard design for all AP1000 plants. These DCD changes are generic and are expected to be included in all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes to the AP1000 DCD contribute to the increased standardization of the certification information in the AP1000 DCD.

10.2.5 Valve Control

10.2.5.1 Summary of Technical Information

NUREG-0800 Section 10.2, Criterion II.1.B, states that the applicant should provide turbine main steam stop and CVs and reheat steam stop and intercept valves to protect the turbine from exceeding set speeds, as well as to protect the reactor system from abnormal surges. To ensure that turbine overspeed is controlled within acceptable limits, the reheat stop and intercept valves should be capable of closure concurrent with the main steam stop valves (MSVs) or of sequential closure within an appropriate time limit. The valve arrangements and valve closure times should ensure that a failure of any single valve to close will not result in an excessive turbine overspeed in the event of a turbine-generator system trip signal.

10.2.5.2 Evaluation

The valve arrangement in the new design is basically the same as in the original design, with the following exception: The No. 2 and No. 4 stop valves in the new design have a bypass valve, which is controlled by an electrohydraulic servo actuator for CV warming. The closure time for all stop, control, reheat, and intercept valves is 0.3 seconds for both the new and the original design.

The specific criterion that applies to the change evaluated above is 10 CFR 52.63(a)(1)(vi) in that the change substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security.

10.2.5.3 Conclusion

On the basis of its review, the staff finds that the design change is acceptable, because the availability and adequacy of the CVs in the AP1000 design conform to Criterion II.1.B and Criterion II.3 in NUREG-0800 Section 10.2.

10.2.8 Turbine Rotor Integrity

10.2.8.1 Summary of Technical Information

In the proposed changes to the AP1000 DCD, the applicant proposed an alternative SPC design that includes changing the design of the turbine rotors from a Westinghouse/Mitsubishi model to a Toshiba model. This evaluation addresses only the turbine-rotor design change. Revision 0 of TR-86, dated February 8, 2007, provided the technical justification for the proposed changes. Revision 1 of TR-86, submitted on June 25, 2007, established the Toshiba design as the single, standard turbine-rotor design for all AP1000 plants. In a letter dated July 12, 2007, the applicant submitted additional information on the TR. The staff evaluates the rotor design and probability of missile generation for the Toshiba turbine design below.

10.2.8.2 Evaluation

GDC 4 requires SSCs important to safety to be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. The staff reviewed the AP1000 DCD changes related to this section that ensure turbine-rotor integrity and a low probability that turbine-rotor failure will result in the generation of missiles.

In the proposed changes to AP1000 DCD, the applicant proposed a change to the design of the turbine rotors from a Westinghouse/Mitsubishi model to a Toshiba model. The Toshiba design includes two types of bucket (blade) fixations instead of side-entry blades, as in the Westinghouse/Mitsubishi design. The two types of fixations are an outside dovetail and a fork-type dovetail. The outside dovetail will be used in every stage of the high-pressure turbine and the fork-type dovetail will be used in the last two stages of the low-pressure turbine.

As a result of the design change, the applicant performed a new analysis of the missile generation probability for fully integral rotors of the Toshiba turbine-generator design. Westinghouse Commercial Atomic Power (WCAP)-16650-P, Revision 0, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low-Pressure Turbines," issued February 2007, includes the methodology and results of the analysis. This report evaluates the probability of missiles resulting from four failure mechanisms: ductile burst from destructive overspeed, high-cycle fatigue cracking, low-cycle fatigue cracking, and stress-corrosion cracking. The staff reviewed the methodology used in WCAP-16650-P and its applicability to the Toshiba design. The applicant's analysis concluded that an inspection interval of up to 24 years was sufficient to satisfy the requirement that the probability of missile generation be less than 1.0×10^{-5} per year. The applicant included conservative assumptions in its analysis for parameters, such as the stress required for ductile burst, the probability for stress-corrosion crack initiation, the probability of rotor overspeed, and the number of startup and shutdown cycles. In RAI-SRP10.2.3-CIB1-01, the staff requested additional information about the way in which the new missile probability analysis addressed high-trajectory missiles.

In a letter dated June 20, 2008, the applicant stated that the AP1000 design has a favorable turbine orientation that prevents turbine missiles from causing unacceptable damage to safety-related SSCs. The analysis in WCAP-16650-P demonstrates that for high-trajectory missiles, the probability of generating a missile from a burst turbine rotor (P1) is 10^{-5}. This meets the guidance in NUREG-0800 Section 3.5.1.3 of 10^{-4} for a favorable turbine orientation and 10^{-5} for an unfavorable turbine orientation. The staff finds this acceptable, because the applicant's analysis for high-trajectory missiles meets the guidance in NUREG-0800 Section 3.5.1.3.

The applicant also stated that the analysis in WCAP-16650-P applies only to high-trajectory missiles and that high-velocity, low-trajectory missiles cannot directly strike safety-related systems and components. However, the turbine generators at dual-unit sites must be considered unfavorably oriented, because safety-related systems and components are in the low-trajectory missile strike zone. Therefore, to address COL applications for dual-unit sites, the staff requested that the applicant provide a bounding turbine missile analysis for low-trajectory missiles or provide a COL action item that requires COL applicants to provide a turbine missile analysis for low-trajectory missiles at dual-unit sites. In a letter dated April 13, 2009, the applicant stated that WCAP-16650 also applies to low-trajectory turbine missiles for dual unit sites that have unfavorable orientation. However, the applicant did not provide justification for applying this analysis for low-trajectory turbine missiles, and did not address whether the analysis is dependent on missile trajectory. The staff identified this as Open Item OI-SRP10.2.3-CIB1-01.

In a letter dated September 22, 2009, the applicant provided further clarification of the applicability of WCAP-16650. The applicant stated that WCAP-16650-P determines the probability of generating a missile due to a burst turbine rotor, regardless of whether the missile is high-trajectory or low-trajectory. In addition, the applicant stated that the AP1000 DCD does

not consider low-trajectory missiles for a single unit site. However, if a dual unit site is considered, the probability, P1, from WCAP-16650-P can be used to evaluate low-trajectory missiles. Therefore, the staff finds that since the angle of trajectory is not used in WCAP-16650-P for determining the total probability of generating a missile due to a burst turbine rotor, P1, the results of the analysis can be used for both low and high-trajectory missiles. The P1 result from WCAP-16650-P is then used to ensure that the probability of striking SSCs, P4, is less than 10^{-7} per the guidance of regulatory guide (RG) 1.115, "Protection Against Low-Trajectory Turbine Missiles," Revision 1, and NUREG-0800 Section 3.5.1.3. On this basis, the staff finds the response acceptable; therefore, Open Item OI-SRP10.2.3-CIB1-01 is closed.

Based on the conservative assumptions incorporated into this analysis and the fact that the rotors will be inspected during plant operation at approximately 10-year intervals, the staff finds the methodology and results acceptable. The staff, therefore, finds the proposed rotor design changes acceptable, because they do not change the probability of missile generation for fully integral low-pressure rotors previously accepted by the NRC staff in NUREG-1793.

The staff reviewed the proposed changes to the AP1000 DCD. The changes establish the proposed design as the single, standard design for all AP1000 plants. These DCD changes are generic and are expected to be included in all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD.

10.2.8.3 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design change is acceptable, because it meets the requirements of GDC 4 and does not change the probability of missile generation for fully integral rotors that the staff previously accepted. Furthermore, the staff finds that the conclusions regarding the proposed turbine-rotor design and the methodology for analyzing the probability of missile generation are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable, under 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the DC information.

10.2.10 Valve Testing Intervals

10.2.10.1 Summary of Technical Information

Section 10.2.3.6 of the AP1000 DCD, proposed a decrease in the frequency of turbine-valve testing from every 3 months to every 6 months. To support this change, the applicant submitted TR WCAP-16651-P, Revision 0, "Probabilistic Evaluation of Turbine Valve Test Frequency," issued February 2007. This report is similar to WCAP-15785, issued April 2002, with the same title, which used the same methodology to evaluate turbine-valve test frequency for the Westinghouse/Mitsubishi turbine design approved in Revision 15 of the AP1000 DCD. The methodology in WCAP-16651-P is based on an analysis of the operating experience with Toshiba Corporation nuclear steam turbines. The applicant introduced a proposed change to the DCD by replacing the word "quarterly" with "six-month" or "semi-annual" as follows:

> Turbine valve testing is performed at six-month intervals. The semi-annual testing frequency is based on nuclear industry experience that turbine-related

tests are the most common cause of plant trips at power. Plant trips at power may lead to challenges of the safety-related systems. Evaluations show that the probability of turbine missile generation with a semi-annual valve test is less than the evaluation criteria.

10.2.10.2 Evaluation

For the plant-specific turbine-rotor test data, the current COL Information Item 10.2-1 states that the COL holder "will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis." The applicant proposed to revise this information item by adding, to the end, the following time commitment: "prior to fuel load after the fabrication of the turbine." TR-6, "AP1000 As-Built COL Information Items" (APP-GW-GLR-021), Revision 0, issued June 2006, stated that an applicant cannot provide the as-built data at the time of the COL application. This justification is appropriate because the as-built material data for turbine rotors will not be available until the rotor material is procured for fabrication. In addition, the proposal to provide the as-built data before fuel load after the fabrication of the turbine is acceptable for reasons similar to those allowing the deferral of the maintenance and inspection program.

Regarding the proposal to extend the valve-test interval from 3 months to 6 months, the analysis in WCAP-16651-P showed that, for a favorably oriented turbine, the missile-probability criterion of 10^{-4} per year could be met, based on the operating experience with Toshiba turbines. In response to RAI-SRP10.2.3-CIB1-02, the applicant confirmed that the AP1000 turbine has a favorable orientation with respect to high-trajectory missiles. The analysis also concluded that the turbine-missile probability criterion of 10^{-5} per year could be met with a 6-month valve-test interval. This is significant because, for dual-unit sites, the turbine orientation is considered unfavorable with respect to low-trajectory missiles potentially striking safety-related systems and components. In response to RAI-SRP10.2.3-CIB1-02, the applicant provided clarifying information about the calculations used to determine the probability of missile generation, based on the frequency of valve tests.

The analysis presented in WCAP-16651-P, along with clarifying information provided in response to RAI-SRP10.2.3-CIB1-02, indicates that the existing operating experience for Toshiba turbines supports a missile generation probability of less than 10^{-5} per year, using 95-percent confidence-level values for the system separation frequency and valve-failure frequency. Therefore, based on the information provided, the staff concludes that the 6-month valve testing frequency meets the missile generation acceptance criteria for both favorable orientation (10^{-4} per year) and unfavorable orientation (10^{-5} per year).

However, the staff notes that an apparent error exists in Table 6-5 of WCAP-16651-P, Revision 0. Based on the equation given on page 18 of WCAP-16651-P, Revision 0, the staff's calculations indicated that, for a 6-month valve-test interval, the value of "probability of a turbine missile" for "1/Time Interval" was an order of magnitude lower than the value shown in Table 6-5. The value in Table 6.5 of Revision 0 does not support a six-month valve test frequency, and the staff identified this as Open Item OI-SRP10.2.3-CIB1-02. In a letter dated July 10, 2009, the applicant addressed this open item by submitting WCAP-16651-P, Revision 1. The applicant confirmed that the exponent in the probability value in Table 6-5 of Revision 0 was a typographical error. The corrected value in Table 6-5 of Revision 1 is consistent with the applicant's calculated value of the annual probability of a turbine missile that was used to support the six-month valve test interval. Therefore, Open Item OI-SRP10.2.3-CIB1-02 was closed.

10.2.10.3 Conclusion

Based on this evaluation, the staff concludes that the DCD changes meet the requirements of GDC 4 and thus are acceptable. The proposed DCD changes are acceptable, under 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information.

10.2.11 Turbine Rotor Maintenance and Inspection Program

10.2.11.1 Summary of Technical Information

The proposed changes to the AP1000 DCD revised COL Information Item 10.2-1 to change the timing for the COL holder to provide information on the turbine-rotor maintenance and inspection program from within 3 years of obtaining a COL until prior to fuel load. NUREG-1793 also discusses this information item under COL Action Item 10.5-2. TR-6 justified the revision to COL Information Item 10.2-1. The DCD changes also proposed a decrease in the frequency of turbine-valve testing from every 3 months to every 6 months.

In AP1000 DCD Tier 2, Revision 15, Section 10.2.6, COL Information Item 10.2-1 states the following:

> The Combined License holder will submit to the staff for review and approval within 3 years of obtaining a Combined License, and then implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in Subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis.

In the proposed changes to the AP1000 DCD, the applicant proposed a modification to COL Information Item 10.2-1 by changing the COL holder commitments to "prior to fuel load." The proposed revision to DCD Tier 2, Section 10.2.6, states the following:

> The Combined License holder will submit to the NRC staff for review prior to fuel load, and then implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in Subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis prior to fuel load after the fabrication of the turbine.

10.2.11.2 Evaluation

GDC 4 requires that SSCs important to safety be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. The staff reviewed the AP1000 DCD changes related to this section that ensure turbine rotor integrity and a low probability of missile generation caused by turbine rotor failure.

COL Information Item 10.2-1 addressed two issues: the turbine maintenance and inspection program and the plant-specific turbine rotor test data. TR-6 proposed modifications to both commitments.

For the turbine maintenance and inspection program, TR-6 states the following:

> A turbine maintenance and inspection program that is consistent with the frequency in DCD Subsection 10.2.3 does not need further approval from the NRC. The testing and inspection frequency in DCD Subsection 10.2.3 are supported by evaluations that are based on operating and inspection program experience in operating power plants.

For the maintenance and inspection program, the applicant proposed that the current COL Information Item 10.2-1 replace the commitment "submit to the staff for review and approval within 3 years of obtaining a Combined License" with "submit to the NRC staff for review prior to fuel load." With respect to the proposed change in valve-test frequency, the staff requested, in RAI-SRP10.2.3-CIB1-02, that the applicant clarify the analysis in WCAP-16651-P with respect to the evaluation criterion.

Since the deferral of the activities in the COL information item does not alter the design of the turbine, turbine valves, or connected piping systems, it will not affect turbine rotor integrity. Further, this change does not involve a test or experiment, does not affect design features associated with the mitigation of severe accidents, and does not alter barriers or alarms that affect security assessments of the AP1000. Therefore, the staff finds that the applicant revised COL Information Item 10.2-1 to be acceptable. This determination is also consistent with the acceptance criteria in NUREG-0800 Section 10.2.3, "Turbine Rotor Integrity."

10.2.11.3 Conclusion

Based on this evaluation, the staff concludes that the DCD changes meet the requirements of GDC 4 and thus are acceptable. Furthermore, the staff finds that the TR-6 conclusions regarding the submittal and availability of COL holder information for a turbine-rotor maintenance and inspection program are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable, under 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information.

10.3 Main Steam Supply System

10.3.1 Main Steam Supply System Design

10.3.1.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.3, "Main Steam Supply System (MSSS)," of the AP1000 DCD, Revision 15. In the proposed changes to the AP1000 DCD, the applicant proposed changes to this section, supported by the following Westinghouse reports: TR-86, "Alternate Steam and Power Conversion Design," (APP-GW-GLN-018), Revision 1, issued June 25, 2007; TR-103, "Fluid System Changes" (APP-GW-GLN-019), Revision 2, dated January 29, 2008; and TR-125, "Corrections to Tier 1 ITAAC 2.2.4 and Tier 2 Section 3.6.1.3.3 and 10.3" (APP-GW-GLR-125), Revision 0, issued May 2007.

10.3.1.2 Evaluation

The staff reviewed the changes to AP1000 DCD, Section 10.3. NUREG-1793 includes the regulatory basis for Section 10.3 of the AP1000 DCD. The staff reviewed the proposed changes to DCD Section 10.3 against the applicable acceptance criteria in NUREG-0800 Section 10.3, "Main Steam Supply System." Those changes that involve NRC review considerations as reflected in NUREG-0800 Section 10.3 are described and evaluated in this section. The specific criteria that apply to the MSSS design related changes include 10 CFR 52.63(a)(1)(vi) in that the changes substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information. The results of the staff's evaluation are provided as follows:

(1) Section 10.3.2.2.4, "Main Steam Isolation Valves," Page 10.3-8:

The staff reviewed the applicant's additional language stating that, on loss of electric power, the main steam isolation valves (MSIVs) remain in their current position. Since the closure of the MSIVs is credited for the mitigation of several design-basis transients and accidents, such as an inadvertent opening of a steam generator relief or safety valve, steam system piping failure, or steam generator tube rupture, the staff asked the applicant, in RAI-SRP10.3-SBPA-01, to evaluate the effect that maintaining the MSIVs in an open position after a loss of electrical power would have on the mitigation of these transients and accidents. Also, the staff asked the applicant to explain whether it had analyzed the events in DCD Chapter 15, "Accident Analyses," with this MSIV fail-as-is logic.

In its response, dated July 18, 2008, the applicant clarified that the DCD statement, "On loss of electrical power the valves remain in their current position," meant that the MSIVs would not close on loss of electric power to the actuator, since the solenoids are normally deenergized and closed. This is not to imply that the MSIVs will not close, once a steamline isolation signal is generated. As discussed in DCD Section 10.3.2.2.4, each MSIV is provided with a hydraulic/pneumatic actuator. The valve actuator consists of a hydraulic cylinder with a stored energy system to provide emergency closure of the MSIV. The energy to operate the valve actuator is stored in the form of compressed nitrogen contained in one end of the actuator cylinder. The MSIVs are normally maintained in an open position by high-pressure hydraulic fluid. For emergency valve closure, normally deenergized and closed redundant solenoids for each MSIV are energized, resulting in the high-pressure hydraulic fluid being dumped to a fluid reservoir and the compressed nitrogen closing the MSIV. Additionally, the applicant stated that the redundant solenoids for each MSIV are powered by separate divisions from the Class 1E direct current and uninterrupted power supply systems. Energizing either solenoid for each MSIV will close the MSIV. The loss of both redundant power sources is beyond the design basis of the plant.

Based on the above explanation, the staff finds the applicant's response acceptable for the following reasons:

- As described above, high-pressure hydraulic fluid in the actuators maintains the MSIVs open and, therefore, a loss of power will have no effect on the MSIV

position. In addition, since the solenoids are normally deenergized, a loss of power will have no effect on their ability to keep the MSIVs open.

- To close the MSIVs during an emergency, the redundant solenoids powered by separate power sources will be used to dump the high-pressure fluid from the actuators. Since the solenoids are provided by two separate sources of power, the loss of an electric power source should have no effect on the MSIV position. Therefore, the staff finds the applicant's response acceptable, and the concern described in RAI-SRP10.3-SBPA-01 is resolved.

(2) Table 10.3.2-1, "Main Steam Supply System Design Data," Page 10.3-17:

The applicant modified the steam generator flow rates to conform to the new SPC design described in TR-86, Revision 1. In DCD Revision 15, the steamflow rate per steam generator was 3,396,535 kilograms per hour (kg/h) (7,488,000 pounds per hour (lb/h)) and the total steamflow rate was 6,793,069 kg/h (14,976,000 lb/h). In the revised DCD, the applicant changed the steamflow rate per steam generator to 3.40×10^6 kg/h (7.49×10^6 lb/h) and the total steamflow rate to 6.79×10^6 kg/h (14.97×10^6 lb/h). The staff reviewed these changes and determined that they are not significant and that they continue to meet all the acceptance criteria in NUREG-1793. Therefore, the staff considers these changes to be acceptable.

(3) Table 10.3.2-2, "Design Data for Main Steam Safety Valves," Page 10.3-18:

The staff reviewed the applicant's revisions to the set pressures and relieving capacities of the main steam safety valves (MSSVs). Since the MSSVs are credited for the mitigation of many design-basis transients and accidents, including overpressure protection of the heatup events, such as a loss of external electrical load and a turbine trip, the staff asked the applicant, in RAI-SRP10.3-SBPA-02, to provide additional information with respect to the effects of the revised MSSV set pressures and relieving capacities on the event analysis in Chapter 15. The staff also asked whether the applicant had performed the Chapter 15 event analysis with the revised MSSV setpoints and relieving capacities.

In its response, dated July 18, 2008, the applicant stated that it included the revised MSSV setpoints and capacities, as shown in Table 10.3.2-2 of the DCD, in its evaluation of the limiting Chapter 15 event analyses that were provided in its response to RAI-TR29-SRSB-01. The results of the evaluation provided in the response to RAI-TR29-SRSB-01 show that the effect of the changes to the MSSV parameters meets the acceptance criteria of the limiting Chapter 15 events and that the existing analysis is bounding.

Based on its review of the Chapter 15 event analyses, the staff finds that the applicant did include the revised MSSV setpoints and capacities in its evaluation of the limiting Chapter 15 event analyses. In addition, based on the results of the Chapter 15 event evaluation, the staff finds that the changes to the MSSV parameters meet the acceptance criteria of the limiting Chapter 15 events. Because the applicant performed the Chapter 15 event analyses with the revised MSSV setpoint pressures and relieving capacities and because the results remained within the acceptance criteria of the limiting Chapter 15 events, the staff's concern described in RAI-SRP10.3-SBPA-02 is resolved.

(4) Main Steam Piping and Instrumentation Diagrams (Safety-Related System), Pages 10.3-37 and 10.3-39 (Figure 10.3.2-1, two sheets):

The applicant revised the AP1000 DCD Tier 2, Figure 10.3.2-1, to be consistent with the applicant's proposed changes to DCD Tier 1, Section 2.2.4, "Steam Generator System," described in TR-125. This change pertains to the safety-related portion of the MSSS.

During an April 1, 2009, conference call, the staff asked the applicant to provide legible DCD Tier 2, Figures 10.3.2-1 and 10.3.2-2, and to incorporate the July 3, 2008, response to RAI-SRP 3.6.1-SBPA-01 into DCD Section 10.3.1.1. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD, which resolves this issue.

(5) Main Steam Piping and Instrumentation Diagrams (Safety-Related System), Pages 10.3-37 and 10.3-39 (Figure 10.3.2-1, two sheets):

The applicant changed the system arrangement and configuration design in DCD Tier 2, Figure 10.3.2-1, to conform to TR-103.

During an April 1, 2009, conference call, the staff asked the applicant to provide legible Tier 2, Figures 10.3.2-1 and 10.3.2-2 and to incorporate the July 3, 2008, response to RAI-SRP3.6.1-SBPA-01 into DCD Section 10.3.1.1 In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD, which resolves this issue.

(6) Main Steam System Diagram, Page 10.3-41 (Figure 10.3.2-2):

The applicant modified the MSSS arrangement to conform to the new SPC design, as proposed in TR-86, Revision 1. The applicant's changes to sections, tables, and figures of the DCD support the specifics of the new design. The staff reviewed these changes and finds that they are modifications to the system arrangement and configuration design. The staff finds that these changes will have no impact on the ability of the MSSS to supply steam from the steam generator to the main turbine-generator system. The staff finds that the MSSS will continue to meet all of the acceptance criteria in NUREG-1793 and, therefore, the changes are acceptable.

(7) a) Section 10.3.1.1, "Safety Design Basis," Page 10.3-3:

The applicant added the turbine bypass valve and also changed the moisture separator reheater stop valves to moisture separator reheater 2nd stage steam isolation valves in the list of valves credited in a single failure analysis. These valves are credited in the analysis to mitigate the event for those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event. These two changes conform to the applicant's response to RAI-SRP3.6.1-SBPA-01, which was found acceptable in Section 3.6.1 of this report.

b) Tier 1, Table 2.2.4-3, "Components which Provide Backup Isolation of the Steam Generator System," Page 2.2.4-15:

The applicant added two moisture separator reheater 2nd stage steam isolation valves to the list of components, which provide backup isolation of the steam generator system and deleted other components. The staff reviewed this design change and determined that it was made to be consistent with the change made in a) above. Therefore, the staff finds the change to be acceptable.

c) Section 10.3.4.2, "In-service Testing," Page 10.3-12:

The applicant revised the scope of the inservice testing (IST) program for the AP1000 reactor to replace the moisture separator reheater steam supply control valves, MSS-PL-V016A, MSS-PL-V016B, MSS-PL-V017A, and MSS-PL-V017B, with the moisture separator reheater 2nd stage steam isolation valves, MSS-PL-V015A and MSS-PL-V015B, in response to the design change described in a) above. The staff reviewed the revised IST program description for consistency with the design change. MSS-PL-V015A and MSS-PL-V015B have been included in AP1000 DCD Tier 2, Table 3.9-16, because the design change will have those valves perform a backup isolation function for the safety-related MSIVs in the AP1000 reactor design rather than MSS-PL-V016A, MSS-PL-V016B, MSS-PL-V017A, and MSS-PL-V017B. MSS-PL-V015A and MSS-PL-V015B are nonsafety related valves (without a specific safety-related leakage limitation) that are outside the scope of the IST requirements specified in 10 CFR 50.55a, "Codes and standards," and will be included in the IST program for the AP1000 reactor to augment their performance capability. Based on its review, the staff finds the revision to the IST program scope to be consistent with the design change described in a) above.

(8) Piping Material Changes to Section 10.3.2.2.1, "Main Steam Piping," Page 10.3-5; Table 10.3.2-3, "Description of Main Steam and Main Feedwater Piping," Page 10.3-19; Main Steam Piping and Instrumentation Diagrams (Safety-Related System) and Page 10.3-37 (Figure 10.3.2-1, "Main Steam Piping and Instrumentation Diagram (Safety-Related System) – Sheet 1 of 2):

The applicant changed the material used for the main steam piping and segments of the main feedwater line to low alloy steel to facilitate a design life of 60 years with respect to minimizing the effects of erosion/corrosion. In the certified design, carbon steel is the main steam piping material for some portions. The changes proposed in Revision 17 make P11 material (nominally 1.25 percent chromium and 0.5 percent molybdenum) the only material for main steam piping. The staff finds this material is acceptable because testing and operating experience have shown P11 is resistant to flow-accelerated corrosion (sometimes called erosion-corrosion) in main steam piping, including wet steam. Current industry guidance (Electric Power Research Institute (EPRI) NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program") allows P11 main steam components to be excluded from flow-accelerated corrosion inspection programs due to the flow accelerated corrosion resistance imparted by the chromium and, to a lesser extent, molybdenum.

(9) Section 10.3.3, "Safety Evaluation," Page 10.3-10:

The applicant changed from inline main steam line radiation monitors to adjacent-to-line radiation monitors.

The applicant made the change to Section 10.3.3 to be consistent with the detailed monitor description provided in DCD Section 11.5.2.3.1. DCD Section 11.5.2.3.1 has not changed from the DCD Revision 15 description of these monitors as "...adjacent to the steam lines." The staff concludes that both DCD sections are now consistent and that the change to Section 10.3 is acceptable. The staff provides additional evaluation of the main steam line monitors in Section 11.5.2.4 of this report.

(10) Table 10.3.2-2, "Design Data for Main Steam Safety Valves," Page 10.3-18:

In Table 10.3.2-2, the applicant changed the size of MSSV and added clarifying text. The MSSVs are designed to American Society of Mechanical Engineers (ASME) Code Section III, Class 2, seismic Category I. Based on the system's accumulation pressure of 3 percent, in accordance with Subsection NC-7512 of ASME Code, Section III, Division 1, 1989 Edition, Subsection NC, Class 2 components, the applicant increased the relieving capacity of MSSVs. The changes were made to ensure that the MSSV design meets the requirements and conforms to the ASME Code. The staff finds the changes acceptable.

(11) Table 10.3.2-4, "Main Steam Branch Piping (2.5-Inch and Larger) Downstream of MSIV," Page 10.3-20:

The applicant changed, in DCD Table 10.3.2-4: the size of the shutoff valves for the turbine bypass lines to the condenser from 30.5 cm (12 in) to 40.6 cm (16 in), the size of the shutoff valves for the reheating steam to moisture separator reheater lines from 30.5 cm (12 in) to 25.4 cm (10 in), and the numerical value for the maximum steam flow rate for these lines. The description of the shutoff valve and number of lines under the description column was also changed. Further, the size of the shutoff valve for the main steam supply to auxiliary steam system line was changed from 15.2 cm (6 in) to 25.4 cm (10 in).

As reflected in Table 10.3.2-4, the applicant made changes to the main steam branch piping, in order to meet the analyzed closure time for the MSIV backup isolation valves.

The applicant made changes to the main steam shutoff valves in Table 10.3.2-4 to support the change in the turbine generator design and to optimize the auxiliary steam supply system. The staff reviewed these design changes and determined that they are bounded by existing safety analysis because the shutoff valve closure times remain unchanged. The design will continue to meet all acceptance criteria in NUREG-1793. The staff finds the changes are acceptable.

10.3.1.3 Conclusion

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

The staff reviewed the applicant's proposed changes to the AP1000 MSSS. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 MSSS to meet the

applicable acceptance criteria in NUREG-0800 Section 10.3. The staff also finds that the applicant properly incorporated the design changes into the appropriate sections of AP1000 DCD. Because the AP1000 MSSS design continues to meet all of the applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 MSSS are acceptable.

10.4 Other Features

10.4.1 Main Condensers

10.4.1.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.1, "Main Condenser," of the AP1000 DCD, Revision 15. In the proposed changes to the AP1000 DCD, the applicant proposed to make the following changes to Section 10.4.1 of the DCD.

In DCD Section 10.4.1.2, "System Description," Paragraph 2, the applicant revised the condenser tube material providing an option to the COL applicant to substitute "stainless steel" tubes for "titanium" tubes for fresh water cooled plants. The revised description reads as, ".....The condenser is equipped with titanium or stainless steel tubes. The titanium material provides good corrosion and erosion resisting properties. Fresh water cooled plants do not require the high level corrosion and erosion resistance provided by titanium; therefore, 304L, 316L, 904L, or AL-6X may be substituted if desired." The revisions are indicated by the underlined text. The applicant added a corresponding note to Table 10.4.1-1, "Main Condenser Design Data." In addition, a second note added to Table 10.4.1-1 states that if one of the alternate tube materials (i.e., stainless steel) is used, the tubesheet will be carbon steel material with a cladding of the same material as the tubes. The certified design specified titanium or titanium-clad carbon steel as the tubesheet material.

10.4.1.2 Evaluation

The staff reviewed the proposed changes to AP1000 DCD, Section 10.4.1. The regulatory basis for AP1000 DCD, Section 10.4.1, is documented in NUREG-1793. The staff has reviewed the proposed changes to DCD Section 10.4.1 against the applicable acceptance criteria of NUREG-0800 Section 10.4.1, "Main Condensers."

The changes identified above are related to the material of the condenser tubes. Several stainless steel alloys (304L, 316L, 904L, and AL-6X) were added as alternatives to titanium condenser tube material for plants cooled by fresh water. The staff finds the use of stainless steel materials is acceptable because the corrosion of stainless steel in cooling waters is strongly related to the chloride content. Operating experience has shown that the stainless steels proposed by the applicant are suitable for condenser tubes in fresh water applications. For example, EPRI characterizes the corrosion resistance of these stainless steels as very good to excellent in fresh water (EPRI TR-102922, "High-Reliability Condenser Application Study," Final Report, November 1993). The corresponding change to allow the tubesheet to be carbon steel with a cladding of the same stainless steel as the tube material is also acceptable based on the operating experience referenced above.

10.4.1.3 Conclusion

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

The staff reviewed Section 10.4.1, "Main Condensers," as documented in the AP1000 DCD, Revision 17. On the basis of its review, the staff finds that the design changes have been adequately incorporated into the appropriate sections of the AP1000 DCD, Revision 17. The staff also finds, based on operating experience with stainless steel condenser tubes, that the main condenser continues to meet all acceptance criteria as documented in NUREG-1793 and are, therefore, considered to be acceptable.

10.4.2 Main Condenser Evacuation System

10.4.2.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.2, "Main Condenser Evacuation System," of the AP1000 DCD, Revision 15. In the proposed changes to the AP1000 DCD, the applicant revised Sections 10.4.2.2.1, "General Description"; and 10.4.2.2.2, "Component Description," to identify the circulating water system (CWS) as conceptual design information (CDI).

10.4.2.2 Evaluation

The staff reviewed all of the changes to AP1000 DCD, Section 10.4.2. The staff did not re-review descriptions and evaluations of Section 10.4.2 that were previously approved and that are not affected by the new changes.

The specific criterion that applies to the main condenser evacuation system related changes is 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.2 of the AP1000 DCD. The staff has reviewed the proposed changes to DCD Section 10.4.2 against the applicable acceptance criteria in NUREG-0800 Section 10.4.2.

The only change the applicant proposed to AP1000 DCD, Section 10.4.2, was to identify the CWS as CDI. The staff discusses the issue of identifying the CWS as CDI in Section 10.4.5 of this report. Since the changes made to DCD Sections 10.4.2.2.1 and 10.4.2.2.2 conform to the changes made to DCD Section 10.4.5 and do not adversely affect the main condenser evacuation system, the staff finds these changes to be acceptable.

10.4.2.3 Conclusion

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

The staff reviewed the proposed changes to AP1000 DCD, Section 10.4.2. The staff finds no changes except the two conforming changes identified above. On the basis of its review, the staff finds that the main condenser evacuation system continues to meet all of the acceptance criteria in NUREG-1793, and these changes are, therefore, acceptable.

10.4.3 Turbine Gland Seal System

10.4.3.1 Summary of Technical Information

In NUREG-1793, the staff approved AP1000 DCD, Revision 15, Section 10.4.3, "Gland Seal System." In the proposed changes to the AP1000 DCD, the applicant proposed changes to Section 10.4.3 of the DCD.

In AP1000 DCD, Section 10.4.3, the applicant made the following changes, which reflect the new single turbine-generator design (i.e., alternative SPC design) that is proposed in TR-86, Revision 1:

(1) Section 10.4.3.2.2, "System Operation," of the DCD, Revision 16:

AP1000 DCD, Revision 15, Section 10.4.3.2.2, stated that the sealing steam to the turbine shaft seals is supplied from either the auxiliary steam system or from the main steam system extracted ahead of the high-pressure turbine throttle valve. In the DCD Revision 17, the applicant revised Section 10.4.3.2.2 to refer to the "high-pressure turbine control valve."

In addition, DCD Section 10.4.3.2.2, Revision 15, stated that, at times other than initial startup, turbine-generator sealing steam is supplied from the auxiliary steam system or from main steam. In the DCD, Revision 17, the applicant modified paragraph 3 on page 10.4-7 to add "the MSV (main steam stop valve) and CV (control valve) gland steam leak-off" as an additional source of steam to the turbine gland seals.

(2) Section 10.4.3.5, "Instrumentation Applications," of the DCD, Revision 17:

The applicant removed the following sentence: "Pressure control valves are used to provide appropriate pressures to operate both the low and high pressure turbine steam seals."

10.4.3.2 Evaluation

The staff reviewed all of the proposed changes to AP1000 DCD, Section 10.4.3. The staff did not review descriptions and evaluations of Section 10.4.3 that were previously approved and that are not affected by the new changes.

NUREG-1793 includes the regulatory basis for Section 10.4.3 of the AP1000 DCD. The staff has reviewed the proposed changes to AP1000 DCD, Section 10.4.3, against the applicable acceptance criteria of NUREG-0800 Section 10.4.3, "Turbine Gland Sealing System."

The specific criterion that applies to the turbine gland seal system related changes is 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

The staff finds that the changes to DCD Section 10.4.3.2.2 associated with specifying a high-pressure turbine CV and the addition of sealing steam supply from MSV and CV gland steam leakoff are conforming changes to the new design, in accordance with TR-86, Revision 1, and do not affect the staff's conclusion in NUREG-1793 that the gland seal system design meets the requirements of GDC 60, "Control of Releases of Radioactive Materials to the Environment," with respect to the design features in place to control releases of radioactive materials to the environment. Therefore, the staff finds these changes acceptable.

However, with respect to the removal of the pressure CVs from DCD Section 10.4.3.5, "Instrumentation Application," the applicant did not provide a basis for this change. Therefore, in RAI-SRP10.4.3-SBPA-01, the staff asked the applicant to explain and justify this deletion of the pressure-regulating valves.

In response to RAI-SRP10.4.3-SBPA-01, in its letter dated June 20, 2008, the applicant stated that the AP1000 gland seal system uses pressure-regulating valves to control steam pressure to the turbine glands. In the approved design, as described in the DCD, Revision 15, the high-pressure and low-pressure turbine glands used steam at different pressures and, therefore, had their own pressure-regulating valves to deliver their respective pressures. In the proposed changes to the DCD; however, the applicant changed its design to a single-pressure system, in which the high- and low-pressure turbine glands use the same steam pressure from a common gland seal steam supply header. Figure 10.4.3-1 of the DCD, Revision 17, includes the diagram depicting the gland seal steam piping and instrumentation. The gland seal steam header is supplied with steam from both the auxiliary steam and main steam systems, and the above-cited pressure-regulating valves are located in each of these systems upstream of the header. Based on the above discussion and a review of Figure 10.4.3-1, the staff finds the elimination of the gland seal steam dual-pressure system acceptable, since it is the new Westinghouse design and does not affect any safety-related systems or equipment. Therefore, the staff's concern described in RAI-SRP10.4.3-SBPA-01 is resolved.

10.4.3.3 Conclusion

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

The staff reviewed the proposed changes to Section 10.4.3of the AP1000 DCD. The staff finds that the changes with respect to specifying a high-pressure turbine CV, the addition of seal steam supply from MSV and CV gland steam leakoff, and other changes to the gland seal system design proposed in TR-86, Revision 1, do not affect the staff's conclusion in NUREG-1793 that the gland seal system design meets the requirements of GDC 60, with respect to the design features in place to control releases of radioactive materials to the environment. The proposed changes do not adversely affect the AP1000 turbine gland seal steam system.

10.4.4 Turbine Bypass System

10.4.4.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.4, "Turbine Bypass System," of the AP1000 DCD, Revision 15.

In the proposed changes to the AP1000 DCD, the applicant proposed to clarify Section 10.4.4.2.2, "Component Description," regarding the manual use of the bypass valves during cooldown. The revised sentence reads: "...the low T_{avg} block can be manually bypassed for ~~one of~~ the bypass valves that are designated as cooldown valves to allow operation during plant cooldown." The addition is indicated by underlined text.

10.4.4.2 Evaluation

The staff reviewed the proposed changes to Section 10.4.4. The specific criterion that applies to the change is 10 CFR 52.63(a)(1)(vii), in that the proposed change contribute to increased standardization of the certification information.

The regulatory basis for AP1000 DCD, Section 10.4.4, is documented in NUREG-1793. The staff reviewed the proposed changes to DCD Section 10.4.4 against the applicable acceptance criteria of NUREG-0800 Section 10.4.4. The acceptability of the system design is based on meeting the following GDC as described in NUREG-0800:

- GDC 4, as it relates to the system being designated such that a failure of the system (due to a pipe break or system malfunction) does not adversely affect safety-related systems or components

- GDC 34, "Residual Heat Removal," as it relates to the ability to use the turbine bypass system for shutting down the plant during normal operations by removing residual heat without using the turbine generator

The change proposed by the applicant is considered a clarification and does not alter the ability of the system to manually bypass a T_{avg} block and allow operation of the bypass valves during plant cooldown. Therefore, the change is considered acceptable.

10.4.4.3 Conclusion

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

The staff reviewed the proposed changes to Section 10.4.4, "Turbine Bypass System." On the basis of its review, the staff finds the turbine bypass system continues to meet all acceptance criteria as documented in NUREG-1793 and is, therefore, considered to be acceptable.

10.4.5 Circulating Water System

10.4.5.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.5, "Circulating Water System," of the AP1000 DCD, Revision 15. In the proposed changes to the AP1000 DCD, the applicant proposed the following changes to Section 10.4.3.

In Section 10.4.5, the applicant identified CDI items related to the alternative SPC design proposed in TR-86, Revision 1. Additionally, as part of identifying the CDI information, the

applicant revised Section 10.4.5 to include makeup water as an additional source to the CWS by stating the following:

> The CWS and/or makeup water from the raw water system supplies cooling water to the turbine building closed cooling water system (TCS) heat exchangers and the condenser vacuum pump seal water heat exchangers under varying conditions of power plant loading and design weather conditions.

The underlined text indicates the revision.

10.4.5.2 Evaluation

The staff reviewed the proposed changes to the CWS in the AP1000 DCD. The staff did not review descriptions and evaluations of Section 10.4.5 that were previously approved and that are not affected by the new changes. The specific criterion that applies to the changes is 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.5 of the AP1000 DCD. The staff has reviewed the proposed changes to DCD Section 10.4.5 against the applicable acceptance criteria of NUREG-0800 Section 10.4.5, "Circulating Water System," and other applicable criteria.

Since the CWS system configuration and its components, such as circulating water pumps, cooling tower or heat sink, and piping and valves, are plant-specific items, the applicant identified these as CDI items. Further, the addition of makeup water to the CWS for the supply of cooling water to the turbine control system heat exchangers and the condenser vacuum pump seal water heat exchangers are plant-specific CDI items.

Therefore, the staff finds that the addition of makeup water and the identification of CDI items are changes to conform to the new design, in accordance with TR-86, Revision 1, and do not affect the staff's conclusion in NUREG-1793 that the CWS design meets the requirements of GDC 4 with respect to the effects of discharging water that may result from a failure of a component or piping in the CWS. Therefore, the staff finds these changes to be acceptable.

10.4.5.3 Conclusion

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

The staff reviewed the proposed changes to Section 10.4.5. The staff finds that the addition of the makeup water system and the identification of the CDI items are acceptable, as they relate to the new design proposed in TR-86, Revision 1. Further, these revisions to Section 10.4.5 of the DCD, do not adversely affect the ability of the CWS to meet the requirements of GDC 4. On the basis of its review, the staff finds that the CWS continues to meet all of the acceptance criteria in NUREG-1793 and is, therefore, acceptable.

10.4.7 Condensate and Feedwater System

10.4.7.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.7, "Condensate and Feedwater System," of the AP1000 DCD, Revision 15. The applicant proposed changes to Section 10.4.7 of the AP1000 DCD.

The applicant proposed the following four technical changes to the AP1000 DCD, three of which are supported by information in Westinghouse TRs:

(1) In Revision 17 to the AP1000 DCD, pages 10.4-21, 10.4-22, 10.4-26, and drawing page 10.4-71 (Figure 10.4.7-1, Sheet 3 of 4), the applicant proposed changing the number of high-pressure feedwater heater stages from one to two. The AP1000 DCD, Revision 15, designates the one high-pressure feedwater heater stage as feedwater heater number 6, with the stage consisting of two parallel feedwater heaters, numbers 6A and 6B. The AP1000 DCD, Revision 17, designates the additional stage feedwater heater number 7, with the stage consisting of two parallel feedwater heaters, numbers 7A and 7B. TR-86, Revision 1, documents the basis for this change.

(2) In the AP1000 DCD, Revision 17, page 10.4-25, the applicant changed text in the first and second paragraphs under the title "Low-Pressure Feedwater Heaters." The current text reads as follows:

> Except for the No. 1 feedwater heaters, the closed low-pressure feedwater heaters have integral drain coolers, and their shell side drains cascade to the next lower stage feedwater heater. The drains from the No. 1 heaters are dumped to their respective condenser shell.
>
> A drain line from each heater allows direct discharge of the heater drains to the condenser in the event the normal drain path is not available or flooding occurs in the heater.

The revised text reads as follows:

> The closed low-pressure feedwater heaters use drain coolers. The cascaded drains from the heaters are dumped to their respective condenser shell.
>
> A drain line from the heater allows direct discharge of the heater drains to the condenser in the event the normal drain path is not available or flooding occurs in the heater.

TR-86, Revision 1, documents the basis for this change. The applicant also incorporated this change in the AP1000 DCD, on drawing page 10.4-69 (Figure 10.4.7-1, Sheet 2 of 4), to show that the number 1 and number 2 feedwater heater drains dump into their respective condenser shells.

(3) In the AP1000 DCD, Revision 17, on page 10.4-73 (Figure 10.4.7-1, Sheet 4 of 4), the applicant proposed eliminating pressure transmitters PT043, PT044, and PT045, and

pressure instrument controllers PIC043, PIC044, and PIC045, and replacing them with PT042 and PC042. TR-103, Revision 2, documents this change.

(4) In the AP1000 DCD, Revision 17, on pages 10.4-69 and 10.4-71 (Figure 10.4.7-1, Sheets 2 and 3), the applicant apparently made a change to the arrangement of the deaerator feedwater heater and its associated storage tank.

10.4.7.2 Evaluation

The staff reviewed all changes to the condensate and feedwater system (CFS) in the AP1000 DCD. The staff did not re-review descriptions and evaluations of Section 10.4.7 that were previously approved and are not affected by the new changes. The specific criteria that apply to the CFS related changes include 10 CFR 52.63(a)(1)(vi) in that the changes substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.7 of the AP1000 DCD. The staff has reviewed the proposed changes to DCD Section 10.4.7 against the applicable acceptance criteria of NUREG-0800 Section 10.4.7, "Condensate and Feedwater System." The following evaluation discusses the results of the staff's review.

10.4.7.2.1 Condensate and Feedwater System—Addition of High-Pressure Feedwater Heater Stage 7

In TR-86, Revision 1, Section 1, the applicant stated that an alternative SPC design was presented in Revision 0 of TR-86 as DCD Chapter 10A, and that all standard AP1000 units built will now employ a single turbine-generator and steam-cycle design. The applicant proposed incorporating changes from DCD Chapter 10A as a single design for AP1000 and stated that one of the significant differences between the reference design and the alternative new design of the turbine-generator set is the addition of a seventh-stage feedwater heater. Because the alternative design of the turbine-generator set uses a seventh-stage high-pressure feedwater heater in its feedwater system design, and because the addition of the seventh-stage heater has no effect on CFS compliance with GDC 2, "Design Bases for Protection against Natural Phenomena"; GDC 4; GDC 44, "Cooling Water"; GDC 45, "Inspection of Cooling Water System"; and GDC 46, "Testing of Cooling Water System," the staff finds the addition of the second high-pressure feedwater heater to the AP1000 CFS design acceptable. The staff also finds that the addition of the seventh-stage heater does not result in a change to the CFS compliance with NUREG-0800 Section 10.4.7 in NUREG-1793.

10.4.7.2.2 Condensate and Feedwater System, Low-Pressure Feedwater Heater Stages 1 and 2 Drain to Their Respective Condensers

In TR-86, Revision 1, on page 52, the applicant proposed a design change for low-pressure feedwater heater stages 1 and 2 drains to dump directly into their respective condenser shells. This design change is part of the alternative SPC design described in Revision 0 of TR-86 as DCD Chapter 10A. Also, on page 60 of TR-86, Revision 1, the applicant proposed a similar change to DCD Figure 10.4.7-1 (Sheet 2 of 4), which has been changed schematically to show that the drains from low-pressure feedwater heaters 1A, 1B, 1C, 2A, 2B, and 2C dump into the respective condenser shells A, B, and C. The low-pressure feedwater heater stage 2 drain no

longer cascades into feedwater heater stage 1. Because the alternative design of the turbine-generator set uses a different drainpath for the No. 2 low-pressure feedwater heater stage, which does not affect the safety-related functions of the CFS or its compliance with GDC 2, 4, 44, 45, and 46, the staff finds this design change to the AP1000 CFS acceptable. The staff also finds that the change to the heater drain system does not result in a change to the CFS compliance with NUREG-0800 Section 10.4.7 in NUREG-1793.

10.4.7.2.3 Condensate and Feedwater System Pressure Transmitters PT043, PT044, and PT045, and Pressure Instrument Controllers PIC043, PIC044, and PIC045 Replaced with PT042 and PC042

In TR-103, Revision 2, on page 7, the applicant proposed eliminating the control function shown in Figure 10.4.7-1 (Sheet 4 of 4) for PT-043–45, because the startup feedwater header pressure is no longer an input to the startup feedwater control system. In TR-103, the applicant further stated that all three pressure transmitters are no longer necessary, since they do not have a control function and pressure transmitters PT-044 and PT-045 have been deleted. Additionally, the applicant has revised Note 4 to reflect the updated control logic diagram APP-PLS-J1-143 and has deleted the statement in the note that mentions startup feedwater header pressure as an input to the control system. The applicant also made a change in Note 3 to indicate that control logic diagram APP-PLS-J1-114 has been updated and is now APP-PLS-J1-144.

Because the startup feedwater header pressure signal no longer provides input to the startup feedwater control system, and the new control logic does not affect the CFS compliance with GDC 2, 4, 44, 45, and 46, the staff finds this design change to the AP1000 CFS acceptable. Section 7.7.1.5 of this report reviews the acceptability of instrumentation and controls changes related to AP1000 DCD Sections 7.7.1.8.1, “Feedwater Control”; and 7.7.1.8.2, “Startup Feedwater Control.”

10.4.7.2.4 Figure 10.4.7-1 (Sheets 2 and 3) Revised To Show the ME 05 “Deaerator Feedwater Heater and Deaerator Storage Tank” as a Single Component

The approved DCD shows the deaerator and the deaerator storage tank as separate components. However, AP1000 DCD, Revision 16, Figure 10.4.7-1 (Sheets 2 and 3), shows the deaerator feedwater heater and the deaerator storage tank as a single component. Furthermore, Section 10.4.7.2.2, “Component Description,” of the DCD describes the deaerator as “a tray type, horizontal shell, direct contact heater located on top of a horizontal storage tank.” Since the design of the deaerator and the deaerator storage tank in the AP1000 DCD, Revision 16, differs from the approved design and from the component description provided in the DCD, the staff, in RAI-SRP10.4.7-SBPA-01, dated April 28, 2008, asked the applicant to explain why the design change is acceptable.

In its response, dated June 20, 2008, the applicant stated that there was no change to the function of the deaerator or the feedwater and condensate systems, and the confusion was a result of the various descriptions and pictorial representations of the deaerator used throughout the DCD. A more consistent representation of the deaerator in the DCD would have likely prevented the confusion. The deaerator vendor changed the previous vendor’s design, which used a deaerator storage tank with the deaerating heater mounted on top of the tank. The latest vendor’s standard design is functionally the same, but the deaerating heater is mounted inside the deaerator storage tank.

The applicant further stated that the differences between the two vendors' deaerator designs do not have any effect on the design or function of the feedwater and condensate systems. The applicant changed DCD Figure 10.4.7-1 (Sheets 2 and 3), as well as the text in DCD Section 10.4.7.2.2, to more accurately reflect the current vendor's deaerator.

In evaluating the applicant's response, the staff notes that the applicant indicated that the deaerator vendor had changed and that in the new vendor's design, the deaerator heater is mounted inside the deaerator storage tank, as opposed to being located on top of the tank, as was the case in the DCD revisions before Revision 16. However, while the new design is shown in the revised Figure 10.4.7-1, the deaerator description in Section 10.4.7.2.2 continues to describe the dearator as "a tray type horizontal shell, direct contact heater located on top of a horizontal storage tank." The deareator described in Section 10.4.7.2.2 of the DCD, and that shown in Figure 10.4.7-1, are not consistent, despite the applicant's claim in its response that it changed the text in DCD Section 10.4.7.2.2 to more accurately reflect the current vendor's deaerator. The staff found the DCD information on the deaerator to be inconsistent and the applicant's response to be insufficient, since the response includes an inaccuracy regarding the text in Section 10.4.7.2.2. In DCD Revision 17, Section 10.4.7.2.2, the applicant revised the text that describes the deaerator to reflect what is shown in Figure 10.4.7-1 of the DCD. On the basis of its review, the staff finds RAI-SRP10.4.7-SBPA-01 to be resolved.

10.4.7.3 Conclusion

The staff reviewed the applicant's proposed changes to the CFS. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 CFS to meet the applicable acceptance criteria. The staff also finds that the applicant properly incorporated the design changes into the appropriate sections of the AP1000 DCD. Because the AP1000 CFS design continues to meet all of the applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 CFS are acceptable.

10.4.8 Steam Generator Blowdown System

10.4.8.1 Summary of Technical Information

The staff approved Section 10.4.8, "Steam Generator Blowdown System," of the AP1000 DCD, Revision 15, in the certified design. In the proposed changes to the AP1000 DCD, the applicant revised Section 10.4.8 of the certified design.

- In DCD Section 10.4.8.2.2, the system flow rates were increased by approximately 8 to 9 percent. In a letter dated July 30, 2009, the applicant explained that the flow rates were changed in order to reflect updated system design calculations. In addition, the words, "at standard conditions" were added after the flow rates.

- Three wording corrections were made for consistency with other parts of the DCD.

- A note was added to Figure 10.4.8-1. The note identifies the drawing as a functional arrangement that could have different internal details due to procurement requirements.

10.4.8.2 Evaluation

The staff reviewed the changes to the steam generator blowdown system identified in Section 10.4.8 of the AP1000 DCD. The staff did not review descriptions and evaluations of Section 10.4.8 that were previously approved and that are not affected by the new changes. The specific criteria that apply to the steam and generator blowdown system related changes include 10 CFR 52.63(a)(1)(vi) in that the changes substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.8 of the AP1000 DCD. The staff reviewed the proposed changes to the AP1000 DCD against the applicable acceptance criteria of NUREG-0800 Section 10.4.8, "Steam Generator Blowdown System," and other applicable criteria.

The staff finds the changes in flow rate values acceptable because they do not affect the amount of blowdown as a percentage of steam flow. Hence, the updated flow rates support the level of secondary-side water purification in the certified AP1000 design. Adding the phrase, "at standard conditions" is acceptable because it provides clarification to the flow-rate values.

The other wording changes were made for consistency with the design as documented in other parts of the DCD and are, thus, acceptable. The note added to Figure 10.4.8-1 is acceptable because it clarifies that there is flexibility in the as-built system without affecting the system functionality.

10.4.8.3 Conclusion

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

The staff reviewed the applicant's proposed changes to the AP1000 steam generator blowdown system. The staff finds that the applicant's proposed changes do not affect the ability of the steam generator blowdown system to meet the applicable acceptance criteria. Therefore, the staff finds that the proposed changes to AP1000 DCD, Section 10.4.8, are acceptable.

10.4.10 Auxiliary Steam System

The staff approved Section 10.4.10, "Auxiliary Steam System," of the AP1000 DCD, Revision 15, in the certified design. In the proposed changes to the AP1000 DCD, the applicant proposed the following change to Section 10.4.10 of the certified design.

The applicant replaced the oil-fired boiler with an electric boiler. TR-114, "AP1000 Auxiliary Boiler Sizing and Design" (APP-GW-GLR-114), dated June 14, 2007, documents the basis for this change. The applicant deleted detailed design information for the oil-fired boiler and proposed some changes throughout Section 10.4.10 to reflect the changes addressed in TR-114.

The staff confirmed that all other changes in DCD Section 10.4.10 are editorial and do not require an evaluation.

10.4.10.1 Evaluation

The staff reviewed all changes identified in the AP1000 DCD. The staff did not review descriptions and evaluations of the auxiliary steam system in the AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. All technical changes in the DCD are supported by information presented in Westinghouse TRs. The specific criteria that apply to the auxiliary steam system related changes include 10 CFR 52.63(a)(1)(vi) in that the changes substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.10 of the AP1000 DCD. In its previous evaluations of this section, the staff found that the current NUREG-0800 does not include a section specifically addressing the auxiliary steam system. The staff has determined that the acceptability of this system will be based on meeting the requirements of GDC 4. In other words, failure of the auxiliary steam system, as a result of a pipe break or malfunction of the system, should not adversely affect safety-related SSCs. The staff has reviewed the proposed changes to DCD Section 10.4.10 and discusses the results below.

The applicant replaced the oil-fired boiler with an electric boiler and submitted TR-114 to address the basis for this change. The applicant proposed a series of changes to DCD Section 10.4.10 to be consistent with the new electric boiler. These changes include the following:

- The applicant added text to DCD Section 10.4.10.1.2 to indicate that auxiliary steam supplements the main steam system during startup. This change provides additional information on system operation, has no impact on the auxiliary steam system and is, therefore, acceptable.

- The applicant deleted reference to "oil-fired boiler" and replaced it with "electric package boiler" in DCD Section 10.4.10.2.2. This is a conforming change to reflect the design change and is, therefore, acceptable.

- The applicant changed the nominal net output capacity from "at least 110,000 pounds per hour (49,900 kilograms per hour)" to "approximately 100,000 pounds per hour (45,360 kilograms per hour)" in DCD Section 10.4.10.2.2. This change has no impact on any safety-related SSC and, therefore, is acceptable.

- The applicant deleted the component description for the auxiliary boiler fuel-oil components from DCD Section 10.4.10.2.2. This is a conforming change to reflect the design change and, therefore, is acceptable.

10.4.10.2 Conclusion

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

The staff reviewed the applicant's proposed changes to the AP1000 auxiliary steam system against GDC 4. The staff finds that the applicant's proposed changes do not affect the ability of the auxiliary steam system to meet the applicable acceptance criteria. The staff also finds that the applicant properly incorporated the design changes into the appropriate sections of the AP1000 DCD. The staff concludes that the AP1000 auxiliary steam system continues to meet all applicable acceptance criteria, and the proposed changes are properly documented in the updated AP1000 DCD; therefore, the staff finds that the proposed changes to AP1000 DCD, Section 10.4.10, are acceptable.

11. RADIOACTIVE WASTE MANAGEMENT

The AP1000 radioactive waste (radwaste) management systems control the handling and treatment of liquid, gaseous, and solid radwaste. These systems include the liquid radwaste system (WLS), the gaseous radwaste system (WGS), and the solid radwaste system (WSS). The WLS is designed to control, collect, process, store, and dispose of liquid radioactive wastes. The WLS is discussed in Section 11.2 of this report. The WLS contains holdup tanks, process pumps, and other processing equipment, including monitor tanks and appropriate instrumentation and controls. Ion exchange is the principal waste treatment process in the WLS.

The WGS collects, processes, and monitors gaseous releases. The WGS is discussed in Section 11.3 of this report. The WGS collects gaseous wastes that are potentially radioactive or hydrogen-bearing (i.e., those wastes resulting from degassing the reactor coolant and the contents of the reactor coolant drain tank (RCDT)), stores them for decay in charcoal delay beds, and subsequently releases them into the environment via the plant vent. The WSS controls the processing of solid wastes generated during reactor operation, as well as the packaging and storage of such processed wastes before shipment to a licensed disposal facility. The WSS is discussed in Section 11.4 of this report. The process and effluent radiological monitoring instrumentation and sampling systems, which are discussed in Section 11.5 of this report, detect and measure the radioactive materials in plant liquid and gaseous processes and effluent streams.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the additional and amended information provided by the applicant using the guidance in Chapter 11 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 3, issued March 2007. The NRC developed the original NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," using the guidance from regulatory guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3, issued November 1978. Therefore, this supplement follows the format of the original NUREG-1793.

The scope of this review includes Chapter 11, Sections 11.2 through 11.5 of Revision 16 and 17 of the AP1000 design certification amendment, Tier 2 and Tier 1, Chapter 2, Sections 2.3.10, "Liquid Radwaste System"; 2.3.11, "Gaseous Radwaste System"; and 3.5, "Radwaste Monitoring," which includes the associated inspections, tests, analyses, and acceptance criteria (ITAAC). Section 11.1 did not include any technical changes in Revisions 16 and 17; therefore, the staff did not include Section 11.1 in this supplement.

This section describes the staff's evaluation and findings of the AP1000 Design Control, Document (DCD), Revisions 16 and 17. The staff reviewed Revision 16 upon receipt; however, after review, the staff issued several requests for additional information (RAIs) and the applicant issued Revision 17 prior to the resolution of these RAIs. Hence, this review and evaluation encompasses both revisions.

11.2 Liquid Waste Management System

11.2.1 Summary of Technical Information

In the AP1000 DCD, the applicant proposed to make five technical changes as described below:

1) In AP1000 DCD Section 11.2, the applicant proposed to increase overall liquid waste holdup capacity and improve operational flexibility by adding three additional liquid waste monitor tanks (and associated pumps, piping, instruments, and valves). The applicant proposed to house the new tanks 56,781 liters (15,000 gallons) each in the radwaste buildings. The new tanks are identical to the three existing monitor tanks, which are housed in the auxiliary building. The applicant has documented these changes in Westinghouse TR-116, "Additional Liquid Radwaste Monitor Tanks and Radwaste Building Extension," APP-GW-GLN-116, Revision 0.

2) In AP1000 DCD Sections 11.2.1.2.4 and 11.2.1.3, the applicant added statements certifying compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1406, "Minimization of contamination." These changes are documented in Westinghouse TR-98, "Compliance with 10 CFR 20.1406," APP-GW-GLN-098, Revision 0.

3) In AP1000 DCD Section 11.2, Figure 11.2-2, "Liquid Radwaste System Piping and Instrumentation Diagram, Sheet 1, the applicant proposed changing the figure to show the addition of a flow element to monitor letdown flow from the chemical and volume control system to the liquid waste system. The applicant documented this chemical and volume control system post accident monitoring system (PAMS) instrument modification in TR-118, "Chemical and Volume Control System PAMS Instrument Modifications," APP-GW-GLN-118, Revision 0.

4) In AP1000 DCD Sections 11.2.5.3 and 11.2.5.4, the applicant proposed updates to document the closure of combined license (COL) Information Items 11.2-3 and 11.2-4, respectively and as described in TR-48, "Identification of Ion Exchange and Adsorbent Media," APP-GW-GLR-008, Revision 0 and in TR-73, "Dilution and Control of Boric Acid Discharge," APP-GW-GLR-014, Revision 0.

5) In AP1000 DCD Section 11.2.3.3, 11.2.3.5 and 11.2.5.2, the applicant proposed changes related to the liquid effluent release requirements in 10 CFR Part 20, "Standards for protection against radiation," and 10 CFR Part 50, "Domestic licensing of production and utilization facilities."

The changes in Revision 17 to the AP1000 DCD reflect the responses to the RAIs from Revision 16.

The evaluation below discusses these changes and other information pertaining to NUREG-0800 acceptance criteria, provides an overview of the staff's RAIs, the applicant's responses, and the staff's evaluation of the responses.

11.2.2 Evaluation

The staff reviewed all technical changes and ITAAC to the liquid waste management system in accordance with NUREG-0800 Section 11.2, "Liquid Waste Management System" and all

changes identified by change marks in the AP1000 DCD. In addition, the staff reviewed all changes to the liquid waste management system-based design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.10.

11.2.2.1 Addition of Three Liquid Waste Monitor Tanks

In TR-116, the applicant proposed to add three additional liquid waste monitor tanks (and associated pumps, piping, instruments, and valves). The applicant proposed to house the new tanks, 56,781 liters (15,000 gallons) each in the radwaste building.

In AP1000 DCD Tier 2, Section 11.2, the staff noted that a potential exists for the quantity of radionuclides in the radwaste building portion of the liquid waste management system to exceed the A_1 value, thus requiring these structures, systems, and components (SSCs) to be designated as RW-IIa in accordance with RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 2. However, these SSCs were designated as RW-IIc in AP1000 DCD Tier 2, Section 1, Appendix 1A. In a September 9, 2008 letter, the applicant responded to the staff's concern and provided an analysis to show that the concentrations will be less than the A_1 and A_2 levels. The applicant then modified Revision 17 of the AP1000 DCD to state that the contents of each monitor tank in the non-seismic radwaste building will be less that the A_1 and A_2 levels of 10 CFR Part 71, "Packaging and transportation of radioactive material," Appendix A, "Determination of A_1 and A_2," Table A-1, "A_1 and A_2 Values for Radionuclides." The staff finds this response acceptable and RAI-SRP11.2-CHPB-06 is closed.

In AP1000 DCD Section 3.2, Table 3.2-3, Sheets 63 and 64, the staff noted that the additional equipment for the WLS was not included in the table, nor was the location of the added WLS components included in the Radwaste Building. Moreover, in Table 3.2-3, the applicant did not identify the new WLS piping interconnecting the Auxiliary and Radwaste Buildings and its classification. The applicant added the necessary information to the diagrams and table in Revision 17. The staff reviewed these changes and found them acceptable. RAI-SRP11.2-CHPB-07 is closed.

Based on the evaluation of AP1000 DCD Section 11.2 and the response to the RAI, the staff concludes that the applicant properly identified all design information related to the three additional liquid waste monitor tanks and associated equipment, and provided an adequate demonstration that design objectives for equipment necessary to control releases of radioactive effluents to the environment have been met in accordance with 10 CFR Part 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors."

11.2.2.2 Documentation of Compliance with 10 CFR 20.1406

In TR-98, the applicant proposed to comply with the regulation by selection of design technology. Table TR98-1, "AP1000 Features Applicable to 10 CFR 20.1406" in TR-98 lists specific examples of how the AP1000 design complies with 10 CFR 20.1406 (Items 19, 22, 23, 24, 25, and 26). The staff reviewed the items listed in Table TR98-1 pertaining to the WLS and found that the applicant addressed the minimization of waste generation in 10 CFR 20.1406.

The staff issued regulatory guidance for 10 CFR 20.1406 in RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," after Revision 16 but before the issuance of Revision 17. As such, the staff documented its review according to

RG 4.21 in Chapter 12. The staff finds that this change does not affect the design and performance aspects of the WLS.

11.2.2.3 Chemical and Volume Control System Post Accident Monitoring System Instrument Modifications

In AP1000 DCD Section 11.2, Figure 11.2-2, "Liquid Radwaste System Piping and Instrumentation Diagram" Sheet 1, the applicant changed the figure to include a flow element to monitor letdown flow from the chemical and volume control system to the WLS. The staff concluded that this change does not affect the design and performance aspects of the WLS and provides additional capabilities to monitor the flow to the WLS.

11.2.2.4 Closure of COL Information Item 11.2-3 and 11.2-4

In AP1000 DCD Sections 11.2.5.3 and 11.2.5.4, the applicant proposed updates to document the closure of COL Information Items 11.2-3 and 11.2-4, respectively as described in TR-48 and in TR-73. The applicant proposed that media selection should be a matter for the plant operator, and should not be identified at the COL application stage. The two reports, TR-48 and TR-73, describe this process and give justification to show that COL Information Items 11.2-3 and 11.2-4 should be deleted. For both of these applications, the staff considered that this media will be replaced multiple times throughout the operating life of the plant. Since these changes are operational in nature and do not affect the design and performance aspects of the WLS, the staff finds these changes acceptable.

11.2.2.5 Changes Related to the Effluent Release Requirements in Parts 20 and 50

Section 11.2.3.3, "Dilution Factor" in the AP1000 DCD, Revision 16 omitted compliance with the annual offsite dose limits of 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," Section II.A. The DCD stated that the plant operator selected dilution flow rates to ensure compliance with the effluent concentration limits of 10 CFR Part 20 and any local requirements. Plant operators must also ensure that the annual releases are within the dose limits of 10 CFR Part 50, Appendix I, which in some situations may be the limiting case. Because of this omission, the staff issued an RAI requesting the applicant to incorporate Section II.A requirements. The applicant added this requirement to Revision 17. The staff reviewed this change and concluded that it was acceptable and closed RAI-SRP11.2-CHPB-01.

Section 11.2.3.5 in the AP1000 DCD, Revision 16 stated that the estimated doses from liquid effluents are site-specific and discussed in DCD Section 11.2.5. Section 11.2.5 only stated that the COL applicant will provide a site-specific cost-benefit analysis to address the requirements of 10 CFR Part 50, Appendix I, Section II.D regarding population doses due to liquid effluents. The COL applicant must also comply with the individual dose limits to members of the public in 10 CFR Part 50, Appendix I and 10 CFR 20.1301(e). Because of this omission, the staff issued an RAI requesting the applicant to incorporate the individual dose requirements of 10 CFR Part 20 and 10 CFR Part 50. The applicant added these requirements to Revision 17. The staff reviewed this change and concluded that it was acceptable and closed RAI-SRP11.2-CHPB-02.

The staff found an inconsistency between the estimated dose sections in Sections 11.2, "Liquid Waste Management System," and 11.3, "Gaseous Waste Management System." No individual doses were calculated for liquid effluents, but doses were calculated for gaseous releases. The staff issued an RAI asking the applicant to address this inconsistency. Revision 17 to the AP1000 DCD addressed this inconsistency by adding text that requires compliance with Section II.A of Appendix I to 10 CFR Part 50. The staff reviewed this change and concluded that it was acceptable and closed RAI-SRP11.2-CHPB-03.

11.2.2.6 Preoperational Testing Information

Section 11.2.4, "Preoperational Testing," of the AP1000 DCD, Revisions 16 and 17 did not address the testing and inspection of ion exchange resin. The initial performance of the liquid radioactive waste system depends on the existence and performance of ion exchange resin in the ion exchange vessels. The applicant based the annual liquid effluent release of radioactivity estimated in Section 11.2 on assuming that the media provided a specific level of decontamination as listed in Table 11.2-5, "Decontamination Factors." The applicant did not specify any preoperational testing and inspections to ensure that the resin is properly installed and performing to assumed levels at initial start up.

The staff issued RAI-SRP11.2-CHPB-05 requesting a statement in the DCD indicating that the applicant will confirm the presence of the correct amount of resin in the liquid radwaste system ion exchange vessels. On April 3, 2009, the applicant submitted a proposed revision to DCD Section 11.2.4.3. This revision stated that an inspection of the system would confirm that the applicant installed the proper volume of media into the appropriate components. The staff finds this confirmation acceptable. In a subsequent revision to the DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

11.2.2.7 Reactor Coolant System Effluents

In the AP1000 DCD, Revision 17, the applicant included additional operator actions when returning the contents of the monitoring tanks for further processing. When radioactivity exceeds operational targets, the operator is instructed to recirculate and sample the contents of the monitoring tanks. The staff reviewed this change and found it to be more specific and appropriate. Thus, the staff concluded that this change was acceptable.

11.2.2.8 Tier 1 Section 2.3.10, Liquid Radwaste System

The applicant added three additional WLS monitoring tanks and a letdown flow monitor to reflect the current design changes in Tier 1, Section 2.3.10. These are conforming changes and have no impact on the staff's conclusion about the acceptability of the design of the WLS.

11.2.3 Conclusion

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

In the previous evaluation of AP1000 DCD, Section 11.2, "Liquid Waste Management System," the staff identified acceptance criteria based on the design's meeting relevant requirements in

10 CFR 20.1302, "Compliance with dose limits for individual members of the public," as it relates to limits on doses to persons in unrestricted areas; 10 CFR 50.34a, as it relates to the inclusion of sufficient design information to demonstrate the design objectives for equipment necessary to control releases of radioactive effluents to the environment; 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" General Design Criteria (GDC) 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the design of liquid waste management systems to control releases of liquid radioactive effluents; and GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to the design of liquid waste management systems to ensure adequate safety under normal and postulated accident conditions.

The staff reviewed the applicant's proposed changes to the AP1000 liquid waste management system. The staff concluded that the applicant's proposed changes do not affect the ability of the AP1000 liquid waste management system to meet the applicable acceptance criteria in NUREG-0800 Section 11.2. The staff also concluded that the applicant had properly incorporated design changes into the appropriate sections of the AP1000 DCD. The staff determined that the AP1000 liquid waste management system continued to meet all applicable acceptance criteria and the applicant documented the changes in the updated AP1000 DCD. The staff also concluded that all of the changes related to the liquid waste management system design description and ITAAC as described in AP1000 DCD Tier 1, Section 2.3.10 are acceptable.

11.3 Gaseous Waste Management System

11.3.1 Summary of Technical Information

In the AP1000 DCD, Revision 17, the applicant proposed ten technical changes to the gaseous waste management system as supported by information presented in TR-48, TR-98, and TR-103, "Fluid System Changes" APP-GW-GLN-019, Revision 2. A description of each Revision 16 technical change is provided below:

1) The applicant demonstrated compliance with 10 CFR 20.1406 through the selection of design technology. This change is documented in TR-98. The applicant identified this change in AP1000 DCD, Revision 16, Tier 2, page 11.3-4.

2) The applicant completed COL Information Item 11.3.5.2 by providing the requested information in TR-48. The applicant has identified this change in AP1000 DCD, Revision 16, Tier 2, pages 11.3-4 and 11.3-11.

3) The applicant changed Figure 11.3-1, "Gaseous Radwaste System Piping and Instrumentation Diagram," to update the drawing nomenclature. This change was documented TR-103. The applicant identified this change in AP1000 DCD, Revision 16, Tier 2, page 11.3-20.

4) On page 11.3-2 in the DCD section on water incursion (Section 11.3.1.2.2.2), the applicant added the automatic isolation of the guard bed inlet on high moisture separator level.

5) In the General Description (Section 11.3.2.1) on page 11.3-5, the applicant changed the temperature of the influent gas from 7 degrees Celsius (C) (45 degrees Fahrenheit (F)) to 4 degrees C (40 degrees F).

6) In the General Description (Section 11.3.2.1) on page 11.3-5 and in Section 11.3.2.3.3 on pages 11.3-7 and 11.3-8, the applicant reduced the capacity of each of the two activated carbon delay beds from 100 percent system capacity to 50 percent capacity.

7) In order to maintain a slight positive pressure in the WGS when the system is inactive, the applicant originally proposed to inject a small nitrogen flow into the system. In Revision 17, the applicant eliminated the nitrogen injection and relied on a closed discharge isolation valve to maintain positive pressure. The applicant documented this change in Section 11.3.2.2.1, "Normal Operation," on page 11.3-5.

8) The applicant previously stated that the gas leaving the moisture separator would be monitored for moisture content and a high alarm would alert the operator to a condition requiring attention. The applicant revised the design and operation of the system to monitor temperature and not moisture. The applicant documented this change in Section 11.3.2.2.1, "Normal Operation," on page 11.3-6 and in Table 11.3-2, "Component Data (Nominal) – Gaseous Radwaste System."

9) The applicant removed the Xenon and Krypton dynamic adsorption coefficients and holdup times from Table 11.3-1, "Gaseous Radwaste System Parameters," on page 11.3-12.

10) The applicant removed some of the parameter data and revised some of the remaining parameter values in Table 11.3-2, "Component Data (Nominal) – Gaseous Radwaste System" on page 11.3-13.

The evaluation below discusses these changes and other information pertaining to the NUREG-0800 acceptance criteria, provides an overview of the staff's RAIs, the applicant's responses, and the staff's evaluation of the responses.

11.3.2 Evaluation

The staff reviewed all technical changes to the WGS in accordance with NUREG-0800 Section 11.3, "Gaseous Waste Management System" and all changes identified by change marks in the AP1000 DCD. In addition, the staff reviewed all changes to the gaseous waste management system design description and ITAAC in AP1000 DCD, Tier 1, Section 2.3.11.

11.3.2.1 Compliance with 20.1406

In TR-98, the applicant proposed to comply with the regulation by the selection of design technology. Table TR-98-1 in TR-98 lists specific examples of how the AP1000 design complies with 10 CFR 20.1406 (Items 19, 22, 23, 24, 25, and 26). The staff reviewed the items listed in Table TR-98-1 pertaining to the WGS and found that the applicant addressed the minimization of waste generation in 10 CFR 20.1406.

The NRC staff issued regulatory guidance for 10 CFR 20.1406 in RG 4.21 after Revision 16 but before the issuance of Revision 17. As such, the staff documented its review according to RG 4.21 in Chapter 12. The staff finds that this change does not affect the design and performance aspects of the gaseous waste management system.

11.3.2.2 Completion of COL Item 11.3.5.2

The applicant addressed COL Information Item 11.3.5.2 by providing the requested information in TR-48. The applicant proposed that media selection should be a matter for the plant operator, and should not be identified at the COL application stage. The two reports, TR-48 and TR-73, describe this process and justify the deletion of COL Information Item 11.3.5.2. For both of these applications, the staff considered that this media was a consumable item, designed for replacement when expended. The media will be replaced multiple times throughout the operating life of the plant. For these reasons, the staff concluded that these changes are operational in nature and do not affect the design and performance aspects of the WGS.

11.3.2.3 Revision of Figure 11.3-1 Piping and Instrumentation Diagram

The applicant proposed a change to Figure 11.3-1, "Gaseous Radwaste System Piping and Instrumentation Diagram," to update the drawing nomenclature. The staff concluded that the updated nomenclature did not affect the design or performance of the system and considered the change in nomenclature editorial.

11.3.2.4 Addition of Automatic Isolation of the Guard Bed Inlet on High Moisture Separator Level

In Section 11.3.1.2.2.2, "Water Incursion," the applicant removed an automatic isolation of the guard bed inlet on high moisture separator level. The staff concluded that this change was acceptable because the system has an automatic isolation on temperature, a moisture separator, drain traps, and the guard bed. These all provide protection from wetting of the activated carbon delay beds. Activated carbon loses its retention properties when wet.

11.3.2.5 Temperature of the Influent Gas Changed From 7 °C (45 °F) to 4 °C (40 °F)

In Section 11.3.2.1, "General Description," the applicant changed the outlet gas temperature from the gas cooler from about 7 °C (45 °F) to about 4 °C (40 °F). This change is small and conservative, and does not affect the conclusions in NUREG-1793. The staff found this change acceptable.

11.3.2.6 Reduced Capacity of Each of the Two Activated Carbon Delay Beds from 100 Percent System Capacity to 50 Percent Capacity

In Section 11.3.2.3.3, "Gaseous Radwaste System Tanks," the applicant proposed to reduce each activated carbon delay bed capacity from 100 percent of the system capacity to 50 percent of the system capacity during design basis conditions. A single bed still provided adequate performance under normal conditions. The staff issued RAI-SRP11.3-CHPB-04 requesting additional information to verify compliance with NUREG-0800 Section 11.3, "SRP Acceptance Criteria," Item 2 regarding the capacity of the system to meet the anticipated processing requirements of the plant.

On April 3, 2009, the applicant responded to RAI-SRP11.3-CHPB-04 and provided an analysis to determine the effects of operation with one charcoal delay bed out of service. The applicant used the PWR-GALE code to determine gaseous effluent releases assuming delay times for noble gases with only one delay bed in service. The applicant then compared the expected airborne concentrations and doses at the site boundary to those using the holdup times for two delay beds. The results of the analysis for the one delay bed showed that the effluent

concentrations and doses at the site boundary increased by negligible amounts and were still well within the limits of 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," Table 2, and the design limits of Appendix I to 10 CFR Part 50.

The staff reviewed this analysis and concluded that the applicant used reasonable delay times and an approved computer code to simulate the performance of one delay bed. The staff found the response acceptable and RAI-SRP11.3-CHPB-04 is closed.

11.3.2.7 Elimination of the Nitrogen Injection and Reliance on a Closed Discharge Isolation Valve to Maintain Positive Pressure in WGS

In Section 11.3.2.2.1, the applicant used the discharge isolation valve to maintain the WGS at a positive pressure. In AP1000 DCD Section 11.3.2.3.4, "Remotely Operated Valves," the applicant provided information regarding the nitrogen purge pressure control valve. The applicant stated that this valve maintains a small positive pressure in the WGS to prevent ingress of air during periods of low flow. Section 11.3.1.2.3.1, "Prevention of Hydrogen Ignition," stated that the discharge isolation valve of the WGS is continuously pressurized with nitrogen to prevent ingress of air into the system from the discharge path. The staff concluded that this change is acceptable as a provision to preclude the ingress of air and does not affect the conclusions in NUREG-1793.

11.3.2.8 Monitoring Temperature Instead of Moisture of the Gas Leaving the Moisture Separator

In Section 11.3.2.2.1, "Normal Operation," and Table 11.3-2, "Component Data (Nominal) – Gaseous Radwaste System of Instrument Indication and Alarms," the applicant changed the monitored parameter of the gas leaving the moisture separator to temperature instead of moisture. The high temperature alarm indicates a reduced performance of the moisture separators and alerts the operator of an abnormal situation. Since this change performs the same function as the moisture alarm in the previous revisions, the staff finds this change acceptable.

11.3.2.9 Removal of the Xenon and Krypton Dynamic Adsorption Coefficients and Holdup Times from Table 11.3-1

In AP1000 DCD Table 11.3-1, "Gaseous Radwaste System Parameters," the applicant removed the dynamic adsorption coefficients and holdup times for noble gases. With the removal of this information, the staff questioned the justification of the holdup times used to calculate the Krypton and Xenon releases in the gaseous effluents. In addition, the staff used these holdup times to conclude that no alteration in system operation would be necessary due to adverse meteorological conditions. The staff asked the applicant in RAI-SRP11.3-CHPB-04 to explain the reason for removing the coefficients and holdup times from the table.

On April 3, 2009, the applicant responded to the staff's concern and stated that the dynamic adsorption coefficients and holdup times were removed from the table to avoid confusion. The applicant did not use these data in any analysis. Furthermore, these coefficients and holdup times are less conservative than what the applicant used to calculate the release of noble gases using the GALE code.

The staff confirmed that the applicant did not use these data in any analysis; therefore, any conclusions regarding effluent releases remain unchanged. Even using the more conservative dynamic coefficients and holdup times from the GALE code, the resulting concentrations and doses are low enough to provide an ample margin of safety should adverse meteorological conditions occur. The staff found the response acceptable and closed RAI-SRP11.3-CHPB-04.

11.3.2.10 Removal of Some of the Parameter Data and Revising Some of the Remaining Parameter Values in Table 11.3-2

The applicant revised AP1000 DCD Table 11.3-2, "Component Data (Nominal) – Gaseous Radwaste System" to reflect the performance of the current design. The changes include pump operating pressure, heat exchanger type, and heat exchanger operating temperature, pressure, and flow. These changes have been reviewed by the staff and have been found to be minor, do not affect the performance of the system to meet the design objectives and the conclusions in NUREG-1793, and are, therefore, acceptable.

11.3.2.11 Additional NUREG-0800 Section 11.3 Acceptance Criteria

The NRC staff noted that Section 11.3.3 was missing the consequence evaluation of a gaseous waste system leak or failure. The acceptance criteria in NUREG-0800 Section 11.3 are based on the availability of this information as part of the evaluation and relies on the approach specified in Branch Technical Position (BTP) 11-5. Based on the safety evaluation report (SER) for Revision 15, the applicant performed this analysis in response to an RAI, but the description of the analysis and results were not included in the AP1000 DCD. The staff issued RAI-SRP11.3-CHPB-02 requesting the inclusion of this analysis in the DCD.

On January 13, 2009, the applicant responded to RAI-SRP11.3-CHPB-02 stating that Section 11.2.2 will be revised to include the consequence analysis. The applicant committed to base its analysis on 1 percent fuel defects, 1-hour bypass of charcoal beds, 30 minute decay prior to release to the environment, and updated atmospheric dispersion factors. At that time, the applicant had not submitted the consequence analysis. This issue was tracked as an open item in the SER until the staff received and evaluated the resulting analysis.

On July 9, 2009, the applicant responded to RAI-SRP11.3-CHPB-02 Revision 1, with the requested analysis and proposed revisions to the AP1000 DCD. The applicant based its analysis on BTP 11-5, which included 1 percent fuel defects, 1-hour bypass of the charcoal beds, 30 minute decay prior to release to the environment, and updated atmospheric dispersion factors. The staff reviewed this response and determined that the applicant properly followed BTP 11-5. The staff independently verified the calculations. Based on the facts that the applicant's analysis properly followed BTP 11-5, the staff verified the results, and the results meet the acceptance criteria in BTP 11-5, 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.

11.3.2.12 Tier 1 Section 2.3.11, Gaseous Radwaste System

The applicant revised this section to indicate that the activated carbon delay beds will be designed to one-half seismic Category I instead of full seismic Category I. Since the WGS is not a safety-related system, the staff finds that this change is in accordance with RG 1.143 and is acceptable.

Additionally, in a letter dated January 20, 2010, the applicant also proposed Change Number 23 to AP1000 DCD, Tier 1 Section 2.3.11 removing all ITAAC concerning the seismic design loads of the equipment in the gaseous radioactive waste management system, specifically the carbon delay beds and the discharge isolation valve. In RAI-SRP11.3-CHPB-06, the staff questioned the basis for removing the ITAAC because these components have seismic design criteria that the applicant must inspect and test before operation.

In response to this RAI, the applicant reinstated the ITAAC for the gaseous waste management system and provided seismic design criteria. The staff reviewed this response and found that the ITAAC meet the inspection and testing recommendations in RG 1.143 and finds the response acceptable.

In a subsequent revision to the DCD, the applicant revised DCD Tier 1, Section 2.3.11, and Table 2.3.11-1 to include the appropriate ITAAC, which resolves this issue.

11.3.3 Conclusion

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

In the previous evaluation of the AP1000 DCD, Section 11.3, "Gaseous Waste Management System," the staff identified acceptance criteria based on the design's meeting relevant requirements in 10 CFR 20.1302, as it relates to limits on doses to persons in unrestricted areas; 10 CFR 50.34a, as it relates to the inclusion of sufficient design information to demonstrate the design objectives for equipment necessary to control releases of radioactive effluents to the environment; GDC 60, as it relates to the design of the gaseous waste management system to control releases of radioactive effluents; and GDC 61 as it relates to the design of the gaseous waste management systems to ensure adequate safety under normal and postulated accident conditions.

The staff reviewed the applicant's proposed changes to the AP1000 gaseous waste management system as documented in the AP1000 DCD. The staff concluded that the applicant's proposed changes do not affect the ability of the AP1000 gaseous waste management system to meet the applicable acceptance criteria in NUREG-0800 Section 11.3. The staff also concluded that the applicant had properly incorporated design changes into the appropriate sections of AP1000 DCD. The staff determined that the AP1000 gaseous waste management system continued to meet all applicable acceptance criteria and the applicant documented these changes in the updated AP1000 DCD. The staff also concluded that all of the changes related to the gaseous waste management system design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.11 are acceptable.

11.4 Solid Waste Management System

11.4.1 Summary of Technical Information

In the AP1000 DCD, Revision 16, the applicant proposed to make changes to Section 11.4. The applicant proposed two technical changes that were supported by information presented in TR-98 and TR-103. The description of each technical change is provided below:

1) The applicant proposed to demonstrate compliance with 10 CFR 20.1406 through the selection of design technology. This change is documented in TR-98. The applicant has identified this change in the AP1000 DCD, Revision 16, Tier 2, on pages 11.4-3 and 11.4-14.

2) The applicant proposed to change the type of pump used for spent resin transfer. This change is documented in TR-103. The applicant has identified this change in the AP1000 DCD, Revision 16, Tier 2, Table 11.4-10, and on page 11.4-32.

The applicant made no additional technical changes in subsequent revisions to the AP1000 DCD. The evaluation below discusses technical changes provided in updated revisions to the AP1000 DCD, and provides an overview of the staff's RAIs, the applicant's responses, and the staff's evaluation of the responses.

11.4.2 Evaluation

The staff reviewed all changes to the solid waste management system as described in the AP1000 DCD in accordance with the guidance in NUREG-0800 Section 11.4, "Solid Waste Management System." The regulatory basis for Section 11.4 of the AP1000 DCD is documented in NUREG-1793. The following evaluation discusses the results of the staff's review. In addition, the staff reviewed the solid waste management system design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.11 and identified no changes to this section.

11.4.2.1 Documentation of Compliance with 10 CFR 20.1406

In TR-98, the applicant proposed to comply with the regulation by the selection of design technology. Table TR98-1 in TR-98 lists specific examples of how the AP1000 design complies with 10 CFR 20.1406 (Items 19, 22, 23, 24, 25, and 26). The staff reviewed the items listed in Table TR98-1 pertaining to the solid waste management system and found that the applicant addressed the minimization of waste generation in 10 CFR 20.1406.

The NRC staff issued regulatory guidance for 10 CFR 20.1406 in RG 4.21 after the receipt of DCD Revision 16, but before the issuance of Revision 17. As such, the staff documented its review according to RG 4.21 in Chapter 12. The staff concluded that this change does not affect the design and performance aspects of the solid waste management system.

11.4.2.2 Spent Resin Transfer Pump

In TR-103, the applicant stated that utilities have reported operational problems due to wear of progressive cavity pumps, and the grinding of resin beads. The crushed beads are difficult to dewater and could create a storage problem. To remedy this situation, the applicant proposed to replace the progressive cavity pump with a material handling positive displacement pump. The staff concluded that this change does not affect the design of the WSS. The staff finds that the conclusions of NUREG-1793 regarding the acceptability of the WSS remain valid.

11.4.3 Conclusion

In NUREG-1793, Supplement 1, the staff concluded that the AP1000 design was acceptable and that the applicant's application for the design certification met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In the previous evaluation of AP1000 DCD, Section 11.4, "Solid Waste Management System," the staff identified acceptance criteria based on the design's meeting the relevant requirements in 10 CFR 20.1302, as it relates to limits on dose to persons in unrestricted areas; 10 CFR 50.34a, as it relates to the inclusion of sufficient design information to demonstrate the design objectives for equipment necessary to control releases of radioactive effluents to the environment; GDC 60, as it relates to the design of waste management systems to control releases of radioactive effluents; and GDC 61, as it relates to the design of waste management systems to ensure adequate safety under normal and postulated accident conditions.

The staff reviewed the applicant's proposed changes to the AP1000 solid waste management system as documented in the latest revision of the AP1000 DCD. The staff concluded that the applicant's proposed changes do not affect the ability of the AP1000 solid waste management system to meet the applicable acceptance criteria in NUREG-0800 Section 11.4. The staff also concluded that the applicant had properly incorporated design changes into the appropriate sections of the AP1000 DCD. The staff determined that the AP1000 solid waste management system continued to meet all applicable acceptance criteria and the applicant documented the changes in the updated AP1000 DCD. The staff also concluded that all of the changes related to the gaseous waste management system design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.11 were acceptable. In addition, the staff reviewed the solid waste management design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.11 and found no changes to this section.

11.5 Radiation Monitoring

This section describes the staff's evaluation and findings of the AP1000 DCD, Revisions 16 and 17. The staff reviewed Revision 16 upon receipt and after review issued several RAIs. Prior to resolving these RAIs, the applicant issued Revision 17 to the DCD. Hence, this review and evaluation encompasses both revisions.

11.5.1 Summary of Technical Information

In the AP1000 DCD, Revision 16, the applicant made changes to Section 11.5. In Revision 16, the applicant only made editorial changes by renaming the Technical Support Center to the Control Support Area. The staff reviewed these changes and found them to be editorial in nature.

In Revision 17, the applicant proposed the following five technical changes:

1) The applicant switched the radiation monitor for the service water blowdown from an offline monitor to an inline monitor. The applicant documented this change on pages 11.5-5 in Section 11.5.2.3.1.

2) The applicant eliminated the statement that it will follow the design guidelines of American National Standards Institute (ANSI) N13.1, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities" for the discharge radiation monitor of the turbine island vent. The applicant documented this change in Section 11.5.2.3.3 on page 11.5-11.

3) The applicant switched the radiation monitor for the liquid radwaste discharge from an offline monitor to an inline monitor. The applicant documented this change on pages 11.5-11 and 11.5-12 in Section 11.5.2.3.3.

4) The applicant made several changes to the monitors and specifications listed in Table 11.5-1. These changes included the following:

   - addition of two main steam line monitors for N-16
   - lowering the minimum and maximum of the nominal range of the steam generator blowdown monitors by an order of magnitude
   - adding the detection of Ar-51 and N-13 to the containment atmosphere monitors
   - lowering the minimum and maximum of the nominal range for the fuel handling building, auxiliary building, and annex building exhaust vents by an order of magnitude
   - changing the nominal range for the monitors of the main control room air supply duct, containment air filtration exhaust, health physics (HP) and hot machine shop exhaust, radwaste building exhaust, and the gaseous and liquid radwaste discharge

5) The applicant made several changes to the monitors and specifications listed in Table 11.5-2. These changes included the following:

   - the addition of a liquid and gaseous monitor for radwaste area 2
   - the addition of a containment area personal hatch monitor at maintenance elevation level 30.48 meters (m) (100.0 feet (ft))
   - specifying the location of the containment area monitor at operating deck – 41.22 m (135 ft, 3 inches (in))
   - removing the rail car bay area as a service area for monitor RMS-JE-RE013

The evaluation below discusses these changes and other information, and provides an overview of the staff's RAIs, the applicant's responses, and the staff's evaluation of the responses.

11.5.2 Evaluation

The staff reviewed the entire section and all technical changes to the radiation monitoring systems in accordance with NUREG-0800 Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems." The staff reviewed the entire section and all changes identified by change marks in the AP1000 DCD.

11.5.2.1 Offline to Inline Monitors for Service Water and Radwaste Liquid Discharges

The change from offline to inline monitoring for the service water blowdown, liquid radwaste discharge, and the waste water discharge has the advantage of reducing the potential for liquid leaks, reducing the likelihood of equipment failure, and reducing areas within the system where

radioactivity can accumulate and present exposure problems. For these reasons, the staff concluded that the change from offline to inline monitoring is acceptable.

11.5.2.2 Removal of Commitment to ANSI N13.1 for Turbine Vent Monitor

The applicant removed the commitment to follow ANSI N13.1 for the design of the monitoring system for the turbine island vent. This action raises a question about what design guidance the applicant would follow for this vent. NUREG-0800 Section 11.3 states that ANSI/HPS N13.1-1999 should be used. The staff issued RAI-SRP11.5-CHPB-02 and RAI-SRP11.5-CHPB-03 requesting that the applicant specify what design guidance it will follow.

On January 13, 2009 and March 23, 2009, the applicant responded to the RAIs by stating that since the turbine island vent monitor is an inline noble gas monitor and not a sample extraction monitor, ANSI N13.1 does not apply. The applicant then further stated that the monitor complies with ANSI N42.18-1980, "Specification and Performance of On-Site Instrumentation for Continuous Monitoring Radioactivity in Effluents."

The staff concluded that this response was acceptable given the fact that the 1980 version of ANSI N42.18 does not differ appreciably from the current version. on this basis, the staff closed RAI-SRP11.5-CHPB-02 and RAI-SRP11.5-CHPB-03.

11.5.2.3 Offline to Inline Monitors for Wastewater Discharge

The applicant changed the radiation monitor from an offline monitor to an inline monitor for the wastewater discharge. The staff concluded that the use of an inline monitor is preferable to an offline type. In addition, the applicant found an inconsistency between the descriptions of the action of the radiation monitor for the wastewater discharge between AP1000 DCD Sections 9.2.9 and 11.5.2. To make Section 11.5.2 consistent with the design description in Section 9.2.9, the applicant eliminated the statement that this monitor controls the basin transfer pumps. The staff concluded that this change is acceptable because Section 11.5 now reflects the actual design and function of the monitor originally described in Section 9.2.9.

11.5.2.4 Changes to Table 11.5-1

The staff reviewed the changes to Table 11.5-1. The staff found the changes acceptable for the following reasons:

- the additional monitors provide improved monitoring capabilities for the main steam line
- the addition of argon (Ar-51) and nitrogen (N-13 and N-16) monitoring in the containment atmosphere improve the monitoring capabilities of this system
- the steam generator blowdown monitors have improved sensitivity
- the fuel handling, auxiliary building, and annex building exhaust vent monitors have improved sensitivity
- the changes to the nominal ranges for main control room air supply duct, containment air filtration exhaust, HP and hot machine shop exhaust, radwaste building exhaust, and the

gaseous and liquid radwaste discharge do not change the overall ability of the monitors to perform their intended function

11.5.2.5 Changes to Table 11.5-2

The staff reviewed the changes to Table 11.5-2. The staff concluded that the additional monitoring and location specificity improves the overall area monitoring and is thus acceptable. However, the staff required clarification on the change that removed the rail car bay area as a service area for monitor RMS-JE-RE013. This action seemed to leave the rail car bay area unmonitored. The staff issued RAI-SRP11.5-CHPB-04 to obtain clarification.

The applicant responded that it did not remove the monitor from the rail car bay area but renamed the area. The applicant proposed to revise the title of the area monitor to be consistent with the title of the area. This response is acceptable since the rail car area remains monitored.

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

11.5.2.6 Additional NUREG-0800 Section 11.5 Acceptance Criteria

Section 11.5 states that the radiation monitoring system is designed in accordance with ANSI N13.1-1969. This standard was withdrawn and replaced in 1999 because the approach taken in the 1969 standard does not provide assurance that the sample in the effluent vent would be representative. The 1999 revision to ANSI N13.1 differs significantly from the earlier version in that it is now performance based. NUREG-0800 Section 11.5 (2007) uses the 1999 standard as acceptance criteria.

The 1969 standard does not provide assurance that the sample from the effluent vent is representative of the particulate matter and reactive vapors passing through the vent. The ability to obtain a representative sample is important since it supports the ability of the licensee to determine public and occupational exposure to radioactivity. The staff concluded that the applicant should use the new standard to ensure that the measurements are accurate. The staff issued RAI-SRP11.5-CHPB-01 to the applicant requesting a change to the newer standard.

On May 22, 2009, the applicant responded to the RAI by stating that committing to the 1999 standard would introduce an excessive degree of uncertainty into the detailed design and construction of the AP1000. To address the limitations of the 1969 standard the applicant proposed an alternative approach relying on best design practices and performance testing and criteria similar to the 1999 standard.

The staff reviewed the response and found that the applicant adopted all relevant performance tests and criteria in the 1999 standard. Once built, the applicant will test the system to ensure that it meets the performance objectives of the 1999 standard. 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 staff also reviewed Change Number 45 to AP1000 DCD Tier 2 (proposed DCD Revision 18), Appendix 1A, Sections 3.1.4, 3.6.3.3, 5.2.5.3.3, and 11.5.2.3.1, and LCO 3.4.9

Sections B.3.4.7 and 3.4.9 and as a result issued RAI-SRP11.5-CHPB-05. The applicant subsequently provided sufficient information to demonstrate that the newly proposed Fluorine-18 (F-18) particulate radiation monitor (PSS-JE-RE027) is capable of detecting the AP1000 technical specifications for the reactor coolant system (RCS) leak rate. The discussion of the staff's review and analysis is included in Chapter 23 of this report, Section 23.F,"Changes to Reactor Coolant Pressure Boundary Leakage Detection."

Based on its review and independent verification, the staff concludes that the proposed monitor is sufficiently sensitive to detect the technical specification leak rate. In Revision 18 to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue. On this basis, RAI-SRP11.5-CHPB-05 is resolved.

11.5.3 Conclusion

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

In its previous evaluations of AP1000 DCD, Section 11.5, "Radiation Monitoring," the staff identified acceptance criteria based on NUREG-0800 Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems." The staff reviewed the AP1000 radiation monitoring and sampling systems for compliance with these requirements, as referenced in NUREG-0800 Section 11.5 and determined that the monitoring systems, as documented in AP1000 DCD, Revision 15, were acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to AP1000 DCD Section 11.5. The staff concluded that the applicant's proposed changes did not affect the ability of the AP1000 radiation monitoring instrumentation and sampling systems to meet the applicable acceptance criteria in NUREG-0800 Section 11.5. The staff also concluded that the design changes have been properly incorporated into the appropriate sections of the AP1000 DCD. On the basis that the AP1000 monitoring system for process and effluents continues to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff concludes that all of the changes related to the system design of the AP1000 process and effluent radiation monitoring are acceptable.

12. RADIATION PROTECTION

12.1 Introduction

The AP1000 design control document (DCD) Tier 2, Chapter 12, "Radiation Protection," Revisions 16 and 17 include changes to the descriptions of the commitments pertaining to the radiation protection measures and programs of the AP1000 design, as described in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004. As such, this supplemental document to NUREG-1793 must be used in concert with the original issue of NUREG-1793 to completely understand the full evaluation of the AP1000 standard design.

In the AP1000 design certification amendment, the applicant provided additional information related to the Radiation Protection Program and the design features that will ensure that occupational radiation exposures are as low as is reasonably achievable (ALARA). It also provided information on related facility design changes submitted in various technical reports (TRs) potentially affecting the internal and external radiation exposures to station personnel, contractors, and the general population, resulting from plant conditions, including anticipated operational occurrences that will be within regulatory criteria. The staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed the additional and amended information provided by the applicant, using the guidance in Chapter 12 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 3, issued March 2007. The NRC developed the original NUREG-1793 using the guidance from regulatory guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3, issued November 1978. Therefore, this supplement follows the format of the original NUREG-1793.

The scope of this review includes Chapter 12, Sections 12.1 through 12.5, Revision 17 of the AP1000 DCD, Tier 2 and Tier 1, and Section 3.5, "Radiation Monitoring," which covers the associated inspections, tests, analyses, and acceptance criteria (ITAAC).

Each section of this report describes the staff's evaluation and review results of the changes proposed in Chapter 12 of the AP1000 DCD, Revision 17. The staff reviewed Revision 16 upon receipt and after review issued several requests for additional information (RAIs); however, the applicant submitted Revision 17 prior to the resolution of these RAIs.

12.1.1 Compliance with Title 10 of the *Code of Federal Regulations* Part 20, "Standards for Protection Against Radiation"

The applicant has submitted several TRs with radiation protection implications that include references to industry standards and other regulatory guidance. The applicant's document, TR-98, "AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-098, Revision 0, 'Compliance with 10 CFR 20.1406' " (APP-GW-GLR-017), references Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1406, "Minimization of contamination." The applicant has documented changes to the radwaste building in TR-116, "Additional Liquid Radwaste Monitor Tanks and Radwaste Building Extension" (APP-GW-GLN-116), Revision 0. The applicant documented structural changes in TR-54, "Spent Fuel Storage Racks Structure/Seismic Analysis" (APP-GW-GLR-033), Revision 0, and a redesign of the reactor vessel (RV) head in

TR APP-GW-GLE-016, "Impact of In-Core Instrumentation Grid, Quicklocs and Changes to Integrated Head Package," Revision 0.

12.1.2 Compliance with 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"

Based upon the discussion in Section 12.1 of NUREG-1793, the standardized power plant designer or combined license (COL) applicant will satisfactorily demonstrate that the radiation protection measures incorporated in the AP1000 program, as documented in the DCD, will offer reasonable assurance that, during all plant operations, the occupational doses will be maintained ALARA and within the limits of 10 CFR Part 20. The following sections present the basis for the staff's conclusions.

12.2 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable

12.2.1 Summary of Technical Information

This section addresses the design, construction, and operations policies to maximize the incorporation of both design and construction engineering practices and industry lessons learned to achieve the desired ALARA objectives.

The applicant revised two areas discussing the COL applicant's management commitment and compliance with 10 CFR Part 20 and RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants"; RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable"; and RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," to reflect the plant's staffing and organizational differences. These changes are editorial in nature and are incorporated in AP1000 DCD Section 12.1.3 as COL actions requiring the COL applicant to provide such information.

The staff determined that all changes in Revision 17 to DCD Section 12.1 are editorial, with the exception of the following item:

> In DCD Section 12.1.2.4, the applicant added statements certifying compliance with 10 CFR 20.1406. These changes are documented in Westinghouse TR-98, "Compliance with 10 CFR 20.1406" (APP-GW-GLN-098), Revision 0.

During its evaluation and confirmation, the staff identified insufficient information relating to the description of design features concerning the compliance with 10 CFR 20.1406. In the process of the staff developing RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," issued June 2008 (formerly DG-4012), the applicant submitted TR-98. This report was intended to identify and justify standard changes to be incorporated in the DCD. Revision 17 to the AP1000 DCD has incorporated the information in the various sections of the DCD, including Chapters 9, 11, and 12. The implementation of the regulations in 10 CFR 20.1406 affects other structures, systems, and components (SSCs) described in the DCD. Even though this information was not described in the other DCD chapters, it was referenced in the response to RAI-SRP12.1-CHPB-01.

The evaluation below provides an overview of RAI-SRP12.1-CHPB-01, and discusses the applicant's response and the staff's evaluation of the response.

12.2.2 Evaluation

This section of the DCD lists specific equipment, as well as facility layout and general design considerations, for 10 CFR 20.1406. The description is substantial and provides examples of design features or considerations for meeting 10 CFR 20.1406. These design features appeared to be based on the draft guidance issued for public comment in the development of DG-4012. The staff has since published the guidance as RG 4.21, Revision 0.

The information presented in AP1000 DCD Tier 2, Sections 12.1 through 12.5, Revision 17, identifies some AP1000 general design features that would minimize the contamination of the facility and environment, as well as the generation of radioactive waste. Specifically, DCD Section 12.1.2.4 describes piping and fuel pool design features to comply with 10 CFR 20.1406. However, this information did not address design features that are unique to system designs or their locations in the plant, warranting more technical details. The applicant did not identify those that should be considered as COL action items. The staff asked the applicant to provide this information in RAI-SRP12.1-CHPB-01. In a letter dated September 9, 2008, the applicant described specific features that are incorporated into the AP1000 design to comply with the requirements of 10 CFR 20.1406 or referenced in TR-98, Revision 0, for the systems that were listed in RAI-SRP12.1-CHPB-01 (and for any other plant systems that may generate or transfer radioactive materials or waste).

The staff determined that the analysis provided by the applicant was insufficient because the applicant's response to RAI-SRP12.1-CHPB-01 and TR-98 failed to address specific information regarding compliance with 10 CFR 20.1406. Section 9.4, "Air-Conditioning, Heating, Cooling, and Ventilation System," covers design features for heating, ventilation, and air conditioning systems (HVAC). Additional information was needed concerning design features provided for these systems to prevent or minimize contamination of the environment. For example, whether there are provisions to monitor and collect condensate that may form at coolers or in HVAC ducts that may contain or potentially contain contamination. The staff issued RAI-SRP12.1-SPCV-01 and RAI-SRP12.1-SPCV-02 to request this information from the applicant. This was identified as an open item in the SER with open items and was tracked as Open Item OI-SRP12.1-CHPB-01.

In a letter dated August 19, 2009, the applicant submitted its response to RAI-SRP12.1-SPCV-01 (R1) and RAI-SRP12.1-SPCV-02 (R1). The applicant provided clarifying information regarding the AP1000 design features used to prevent and mitigate the spread of contamination due to HVAC operation or abnormal conditions. Information was provided in the applicant's response on the features to prevent water entry into HVAC ducting from the liquid radwaste system (WLS) and radioactive waste drain system (WRS). The applicant proposed to revise the following DCD sections: 9.4.3.2.1.1, "Auxiliary/Annex Building Ventilation Subsystem"; and 11.4.2.2.1, "Spent Resin Tanks." The staff concludes that the information provided describes the design features that will prevent and mitigate the spread of contamination through ventilation subsystems and is, therefore, appropriate. In a subsequent revision to the DCD, the applicant made an appropriate change to the DCD text; therefore, RAI-SRP12.1-SPCV-01 and RAI-SRP12.1-SPCV-02 and the associated Open Item OI-SRP12.1-CHPB-01 are resolved.

The staff also asked the applicant to provide information in RAI-TR98-CHPB-01, RAI-TR98-CHPB-02, and RAI-TR98-CHPB-03. In a letter dated December 19, 2007, the applicant described other specific design features that are incorporated into the AP1000 design

to comply with the requirements of 10 CFR 20.1406 and that were referenced in TR-98, Revision 0. The applicant has not committed to RG 4.21, Revision 0, but has described the design features in the licensing basis, the referenced TR-98, and in the response to the aforementioned RAIs.

AP1000 DCD Tier 2, Revision 17, Chapter 12, identified some AP1000 general design features that would minimize the contamination of the facility and environment and would minimize the generation of radioactive waste. However, this information did not address design features that are unique to the auxiliary steam and condensate systems. Current generation plant operating experience has demonstrated that normal pressurized-water reactor (PWR) operation will likely result in detectable levels of tritium as well as other radionuclides in the condensate, auxiliary steam system, and boilers. This was identified as RAI-SRP12.1-CHPB-02. The applicant was requested to describe the design features and operating objectives related to the auxiliary steam and condensate transfer system leaks.

In a letter dated March 4, 2010, the applicant submitted its response to RAI-SRP12.1-CHPB-02. In its response, the applicant described that the outdoor piping from the hotwell to the condensate storage tank will be located above grade, such that leakage from the pipe could be detected in the conduct of routine site activities. When describing the interconnected steam and power systems located in the turbine building, the applicant described the collection of leakage via the waste water system and turbine building drains, which are directed to the turbine building sump. This sump requires routine sampling and analysis prior to the discharge into the environment. This is a requirement for the licensed facility operator to demonstrate compliance with the radiological effluent technical specifications (RETS) and Radiological Environmental Monitoring Program (REMP).

Additionally, the applicant described the plant hot water heating system heat exchangers as providing a barrier to the potential spread of contamination, in the event that the auxiliary steam system was to become contaminated by low levels of primary system leakage to the secondary side of the steam generators. The applicant proposed a revision to AP1000 DCD Section 12.1.2.4.1, "Piping," which describes the configuration of the outdoor piping and collection of turbine building drains.

The staff reviewed the response provided by the applicant and found it to be acceptable in that the response provided a description of the design features provided to address minimization of contamination of the facility that were specifically not addressed in TR-98. The response provides adequate information, which would allow for timely identification of low activity leakage from these sources to preclude contamination of the site, facility or environment where an AP1000 site is located. The description provided addresses operating experience from the operating nuclear power plants, as discussed in DC/COL ISG-006, "Office of New Reactors Interim Staff Guidance on Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications." In a subsequent revision to the DCD, the applicant made an appropriate change to Tier 2, Section 12.1.2.4.1 of the DCD, which resolves this issue; therefore, RAI-SRP12.1-CHPB-02 is resolved.

12.2.3 Conclusions

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

design. Based on the information supplied by the applicant, as described above, the staff concludes that the AP1000 design features met the criteria of NUREG-0800 Section 12.1. These design features are intended to maintain individual doses (person-sievert (roentgen equivalent man) (rem)) and total doses to plant workers and to members of the public ALARA, while maintaining individual doses within the regulatory limits of 10 CFR Part 20.

Adding more detailed information to the DCD about the features for minimizing contamination increases standardization of the AP1000 design. Thus, these changes meet the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

The staff reviewed the applicant's proposed changes to the AP1000 radiation protection section describing activities that ensure that occupational radiation exposures are ALARA, as documented in DCD Section 12.1. The staff finds the applicant's proposed changes to DCD Section 12.1 to be acceptable.

12.3 Radiation Sources

12.3.1 Summary of Technical Information

The staff approved AP1000 DCD, Section 12.2, "Radiation Sources," Revision 15, in the certified design. This review addresses the compliance with 10 CFR Part 20 and General Design Criterion (GDC) 61, "Fuel Storage and Handling and Radioactivity Control," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities."

In the AP1000 DCD, Revision 17, the applicant proposed the following two technical changes with associated TR changes:

(1) In AP1000 DCD Section 11.2, the applicant proposed to increase the overall liquid waste holdup capacity and improve operational flexibility by adding three liquid waste monitor tanks (and associated pumps, piping, instruments, and valves). The applicant proposed to house the additional tanks, 56,775 liters (15,000 gallons) each, in the radwaste building. The new tanks are identical to the three existing monitor tanks, which are housed in the auxiliary building. The applicant documented these changes in TR-116, Revision 0.

(2) In AP1000 DCD Section 9.1.2.1, the applicant proposed to increase the overall capacity of the spent fuel pool (SFP) from storage locations for 619 fuel assemblies to locations for 884 fuel assemblies. The applicant documented these structural changes in TR-54, Revision 0. The applicant documented the heat loading analysis changes in TR-103, "Fluid System Changes" (APP-GW-GLR-019), Revision 2.

The evaluation below discusses these changes and missing information and provides an overview of the staff's RAI-SRP12.2-CHPB-01 and RAI-SRP12.2-CHPB-02, the applicant's response, and the staff's evaluation of the response.

12.3.2 Evaluation

Liquid Waste Hold-up Tank

The staff reviewed all technical changes to the radiation sources identified in the AP1000 DCD, Revision 17, in accordance with NUREG-0800 Section 12.2, "Radioactive Sources." In addition, the staff reviewed the entire section to ensure that there was no missing information critical to providing adequate protection to public health and safety.

NUREG-1793 documents the regulatory basis for DCD Section 12.2, Revision 15. The staff has reviewed the proposed changes to DCD Section 12.2 against the applicable acceptance criteria in NUREG-0800 Section 12.2. The following evaluation discusses the results of the staff's review.

In TR-116, the applicant proposed to add three liquid waste monitor tanks (and associated pumps, piping, instruments, and valves) and to house the additional tanks, 56,775 liters (15,000 gallons) each, in the radwaste building.

In its review of AP1000 DCD Section 12.2, the staff identified areas in which additional information was necessary to complete its evaluation of the applicant's change (RAI-SRP12.2-CHPB-01). In DCD Tier 2, Section 11.2, the staff noted that a potential exists for the quantity of the radionuclides in the radwaste building portion of the liquid waste management system to exceed the A_1 value. The DCD states that, "The monitor tanks in the non-seismic radwaste building are used to store processed water. The radioactivity content of processed water in each tank will be less than the A_1 and A_2 levels of 10 CFR 71, Appendix A, Table A-1 ("A_1 and A_2 Values for Radionuclides")."

In 10 CFR 71.4, "Definitions," defines A_1 as the maximum activity of special form radioactive material permitted in a Type A package. A_2 is defined as the maximum activity of radioactive material, other than special form materials, low specific activity, and surface contaminated object material permitted in a Type A package.

The description of the radioactive sources listed for liquid waste tanks included in AP1000 DCD Section 12.2 did not indicate an increase in volume to approximately 170,325 liters (45,000 gallons) and hence an increase in the overall radioactivity that would thereby be a much larger source of occupational radiation to personnel in the radwaste building.

In Tier 2, Figure 12.3-1 (Sheet 14 of 16), "Radiation Zones, Normal Operation/Shutdown Radwaste Building EL 100'-0" indicates that the room (Room No. 50355) that the tanks will be located in is Plant Radiation Zone 1, which is defined by Figure 12.3-1 (Sheet 1 of 16) in Tier 2, as very low or no radiation sources: "Inside Controlled Area" and Outside "Restricted Area." This is an area that will result in a dose rate of less than or equal to 2.5 microSieverts per hour (μSv/h) (0.25 millirem per hour (mrem/h)).

The applicant responded to the staff's concerns in a letter dated September 9, 2008. The applicant agreed that the liquid waste tanks could alter the plant radiation zone assignment and that the issue should be re-analyzed. The affected area will be reclassified to a Plant Radiation Zone III. Plant Radiation Zone III is defined in AP1000 DCD Tier 2, Section 12.3, Figure 12.3-1, as being an area of low-to-moderate radiation sources; limited worker occupancy, with maximum design dose rates less than or equal to 150 μSv/h (15.0 mrem/h). The revised radiation zone reflects the potential increase in volume and hence the increased radioactivity

stored in the liquid waste monitor tanks. Given the building design, tank thickness and potential radioactivity stored in the liquid waste monitor tanks, assigning the adjacent area as a Plant Radiation Zone III is appropriate. Based on the evaluation of the DCD information and the applicant's response to the RAI, the staff concludes that the applicant properly identified all design information related to the three additional liquid waste monitor tanks and associated equipment and provided an adequate demonstration that design objectives have been met for the contained source terms described in the DCD as the basis for radiation design shielding calculations and personnel dose assessment. Therefore, RAI-SRP12.2-CHPB-01 is resolved.

Spent Fuel Pool Capacity

In AP1000 DCD Section 9.1.2.1, "Design Basis," the applicant increased the overall capacity of the spent fuel storage from the proposed storage locations for 619 fuel assemblies to storage locations for 884 fuel assemblies. The staff noted that the additional fuel assemblies were not addressed in DCD Section 12.2.1.2.3, "Spent Fuel," nor included in Table 12.2-25, "Fuel Handling Area Airborne Radioactivity Concentrations." The potential addition of 265 fuel assemblies with 0.25 percent fuel defects would increase the airborne radioactivity. Moreover, Table 12.2-25, did not identify the basis of the parameters included in Table 12.2-24 for the number of fuel assemblies used in its calculations. The staff presented these concerns in RAI-SRP12.2-CHPB-02 and the applicant responded in a letter dated September 9, 2008. The air activity was based on a full core offload, with RV design head removal at 100 hours after shutdown; completion of core offload was determined to be 10 days (that is, 240 hours after shutdown); and the SFP purification system was assumed to be operating at 946 liters per minute (250 gallons per minute). Table 12.2-25 was based only on the core from the recent full offload and thus not affected by the increase in the number of fuel assemblies in the SFP. The applicant committed to completing a detailed review and revising the response, if necessary. This was tracked as Open Item OI-SRP12.2-CHPB-02 in the SER with open items.

In a letter dated September 25, 2009, the applicant submitted its response to Open Item OI-SRP12.2-CHPB-02, which provided a discussion that defined the basis and calculations made to assess the maximum airborne activity in the fuel handling area. Following the review of the calculations and additional information provided in the response, the staff concludes that the information in the applicant's response on the airborne activity based on a full core off-load is conservative and, therefore, appropriate. RAI-SRP12.2-CHPB-02 and the associated Open Item OI-SRP-12.2CHPB-02 are resolved.

12.3.3 Conclusions

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

In its previous evaluations of the AP1000 DCD Section 12.2, the staff identified acceptance criteria based on the ability of the design to meet the relevant requirements in 10 CFR Part 20, as it relates to limits on doses to occupationally exposed persons in restricted areas, and in the requirements of GDC 61, as it relates to the information about radiation sources provided by the applicant.

The staff considers RAI-SRP12.2-CHPB-01 to be resolved, and has verified that the appropriate change was incorporated in Revision 17 to AP1000 DCD Tier 2, Figure 12.3-1, Sheet 14 of 16, room number 50355. Additionally, the staff considers Open Item OI-SRP12.2-CHPB-02 to be resolved and concludes that the applicant has committed to follow the guidelines of the RGs and staff positions set forth in NUREG-0800 Section 12.3.

12.4 Radiation Protection Design Features

12.4.1 Summary of Technical Information

The staff approved AP1000 DCD Section 12.3, "Radiation Protection Design Features," Revision 15, in the certified design. This review addresses system design, performance aspects, and ITAAC only of the radiation protection design features.

The staff's assessment of the revisions listed in Tier 2, Section 12.3, "Radiation Protection Design Features," includes changes in Tier 1, Section 3.5, "Radiation Monitoring" (ITAAC) and Tier 2, Section 11.5, "Radiation Monitoring." These changes do not affect the ITAAC scope and acceptance criteria.

In the AP1000 DCD, Revision 17, the applicant proposed to make the following three technical changes. The staff also identified additional changes related to the density of concrete used for shielding purposes and a description of the computer codes used for shielding calculations. These changes are described below:

(1) In AP1000 DCD Section 12.3.2.2.4, "Fuel Handling Area Shielding Design," the applicant decreased the minimum water depth above the active fuel portion of the assembly. Spent fuel removal and transfer operations are performed under borated water to provide radiation protection and maintain subcriticality. Revision 15 of the AP1000 DCD stated that the minimum allowable water depths above active fuel in a fuel assembly during fuel handling were 3.05 meters (m) (10 feet (ft)) in the reactor cavity and in the SFP. This limits the dose to personnel on the spent fuel pool handling machine (SFHM) to less than 25 μSv/h (2.5 mrem/h) for an assembly in a vertical position. Normal water depth above the stored assemblies is about 7 m (23 ft), and for this depth the exposure to plant workers is insignificant. In DCD Revision 16, the minimum allowable water depth above active fuel in a fuel assembly during fuel handling was decreased to 2.89 m (9.5 ft). During the review of Revision 17 to DCD Tier 2, Section 12.3.2.2.4, the staff observed that the applicant decreased the overall minimum allowable water depth above active fuel in the reactor cavity and SFP to 2.67 m (8.75 ft) during fuel movement.

(2) In AP1000 DCD Section 12.3.1.1.2, "Common Facility and Layout Designs for ALARA," and Section 12.3.5.1, "Administrative Controls for Radiological Protection," the applicant added statements certifying compliance with 10 CFR 20.1406. TR-98, Revision 0, documented these changes. In the DCD sections, the applicant described general practices, such as to minimize the use of embedded pipes to the extent possible, consistent with maintaining radiation doses ALARA. In addition, to the extent possible, pipes will be routed in accessible areas, such as dedicated pipe routing tunnels or pipe trenches, which provide good conditions for decommissioning, and the number of passageways (doors) between the radiological controlled area and the environment has been minimized. When such doors are incorporated, systems of drains and floor and exterior concrete sloping are used to prevent (potentially radioactive) fluid from exiting the buildings, as well as to prevent surface water from entering the buildings. Because

of the potential for adsorption of contaminated fluids, another feature included minimizing the use of concrete block walls in the radiologically controlled areas of the plant. Where such walls are used, they are fully sealed at the ceiling or top of the block to prevent liquid incursion. The applicant added two COL information items where the COL applicant will, in accordance with 10 CFR 20.1406, establish a groundwater monitoring program beyond the normal radioactive effluent monitoring program and will establish a program to ensure documentation of operational events deemed to be of interest for decommissioning, beyond that required by 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning." This or another program will include remediation of any leaks that have the potential to contaminate groundwater.

(3) In AP1000 DCD Tier 2, Section 12.3.1.1.1, the applicant proposed changes to the integrated RV head package and quick-loc connectors. The staff noted that the applicant's description included an integrated head package that combines the head lifting rig, control and gray rod drive mechanisms, lift columns, control rod drive mechanism cooling system, and power and instrumentation. In its review of DCD Sections 12.3 and 12.4, the staff identified areas that needed additional information to complete its evaluation of the applicant's change including Figure 12.3-1 (Sheet 8 of 16), "Radiation Zones, Normal Operation/Shutdown Nuclear Island," EL 135'-3" and Table 12-4-12, "Dose Estimate for Refueling Activities." The description of the change lacked sufficient detail to determine the radiological impact on occupational exposure (RAI-SRP12.3-CHPB-01).

(4) In AP1000 DCD Section 12.3.2.2.9, the applicant decreased the overall assumed concrete density used for shielding design purposes in the spent fuel transfer canal and tube shielding from 147 pounds per cubic foot (lb/ft^3) to 140 lb/ft^3, without an analysis or description of the potential radiological effects.

(5) In AP1000 DCD Section 12.3.2.3, the applicant described computer codes used to determine design and operational dose rates. The DCD Reference 22 identified the point kernel code Microshield 4 as being a personal computer version with a menu guided user interface of the mainframe calculation code QAD. The staff requested additional information to support verification and validation of this computer code as it is used in the AP1000 DCD dose rate calculations.

12.4.2 Evaluation

The staff reviewed all technical changes to the radiation protection design features identified by change marks in the AP1000 DCD, Revision 17, in accordance with NUREG-0800 Section 12.3 - 12.4, "Radiation Protection Design Features." Descriptions and evaluations of the radiation protection design features in the AP1000 DCD, Revision 15, that were previously approved are not affected by the new changes and were not re-reviewed by the staff. Information presented in the TRs support all technical changes in this section of the DCD.

The staff reviewed the Tier 1, Section 3.5, "Radiation Monitoring," ITAAC. This section remained substantially unchanged, but the AP1000 DCD, Revision 17 enhanced the airborne radioactivity and area radiation monitors by adding monitors with multiple detectors and revising the title of selected area monitors. The staff's evaluation of these Tier 1 changes is in Section 11.5 of this report. No additional technical evaluation was required for Section 12.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation."

NUREG-1793 documents the regulatory basis for AP1000 DCD Section 12.3, Revision 15. The staff has reviewed the proposed changes to DCD Section 12.3 against the applicable acceptance criteria of NUREG-0800 Section 12.3-12.4. The following evaluation discusses the results of the staff's review.

12.4.2.1 Fuel Handling Area Shielding Design

In AP1000 DCD Section 12.3.2.2.4, spent fuel removal and transfer operations are performed under borated water to provide radiation protection and maintain subcriticality. According to Revision 15 of DCD Tier 2, Section 12.3.2.2.4, minimum allowable water depths above active fuel in a fuel assembly during fuel handling are 3.05 m (10 ft) in the reactor cavity and SFP. This limits the dose to personnel on the SFHM to less than 25 μSv/h (2.5 mrem/h) for an assembly in a vertical position. Normal SFP water depth above the stored assemblies is approximately 7 m (23 ft); and, for this depth, the exposure to plant workers is insignificant. TR-121, Revision 0, documented several changes and, as a result, the NRC staff questioned the accuracy of the calculated exposure to workers adjacent to the fuel handling area. The response to RAI-SRP12.2-CHPB-02 clarified the issue raised by the NRC staff to show that it was not possible to maintain the original value of 3.05 m (10 ft) of water over the fuel assembly, given the design of the SFP and its handling equipment. The design drawings showed an actual value of 2.59 m (8.5 ft) of water above the active fuel portion of the fuel assembly. The changes submitted in TR-121 changed the water level in the SFP by 0.3 m (12 inches) to 2.89 m (9.5 ft), and dose rate calculations for this water depth showed that the dose to the SFHM operators would be 25 μSv/h (2.5 mrem/h) or less when moving an irradiated fuel assembly while standing on the SFHM. The applicant documented these changes in TR-121, Revision 0. The applicant also responded to the RAI related to TR-121 in a letter dated October 4, 2007, and described assumptions used in its calculations for the exposure of workers adjacent to the fuel handling areas (RAI-TR121-CHPB-01, RAI-TR121-CHPB-03, and RAI-TR121-CHPB-04). The applicant described the potential radiological effects and dose estimates associated with the change in the minimum water level over active fuel in the refueling area and the SFP.

During the review of Revision 17 to AP1000 DCD Tier 2, Section 12.3.2.2.4, the staff observed that the applicant again changed the overall minimum allowable water depth above active fuel in the reactor cavity and SFP from 2.89 m (9.5 ft) to 2.67 m (8.75 ft) during fuel movement. The applicant did not identify the basis of its parameters in Section 12.3.2.2.4 or the reason for this change. The staff issued RAI-SRP12.3-CHPB-02 to request an explanation for the Revision 17 changes to the minimum refueling and SFP water depth.

The staff reviewed the shielding calculation, APP-GW-N2C-006, Revision 2, "Spent Fuel Shielding Evaluation" (Alt Doc # CN-REA-05-55) referenced in the response to the staff's RAI-SRP12.3-CHPB-02. On the basis of this review, the staff had additional questions related to calculations performed for both 259 cm (102 inches(in)) and 267 cm (105 in) above the active portion of the fuel assembly and how the heights of 259 cm (102 in) and 267 cm (105 in) above the fuel assembly correspond to the minimum required water level in the SFP.

APP-GW-N2C-006 also provided the worst-case source, Case D (4.16E+12 γ–cc/sec) from a single fuel assembly at 259 cm (102 in) and 267 cm (105 in) below the water's surface. It appears from the information supplied in the referenced calculation that the source described in Case A (4.26E+12 γ–cc/sec) is more conservative based solely on the dose contribution from the single elevated fuel assembly in the SFP. The staff requested a clarification to determine the basis for selection of Case D versus Case A, as well as a description of design

features/access controls to ensure that the dose to the refueling personnel on the fuel handling bridge deck are maintained ALARA during refueling operations. The staff identified this as Open Item OI-SRP12.3-CHPB-02 in the SER with open items.

In a letter dated July 20, 2009, the applicant submitted its revised response to RAI-SRP12.3-CHPB-02. The applicant indicated that the SFP bridge deck, with 2.54 centimeters (cm) (1 inch) of steel shield, would provide adequate shielding during irradiated fuel movement to maintain whole body exposures to personnel on the SFHM to less than 25 μSv/h (2.5 mrem/h). Calculations referenced by the applicant were reviewed and verified to demonstrate the adequacy of the design to maintain exposures ALARA and were found to be conservative. The staff concludes that the combination of shielding, configuration and SFP water level will maintain exposures ALARA, and is, therefore, acceptable. Therefore, both RAI-SRP12.3-CHPB-02 and the associated Open Item OI-SRP12.3-CHPB-02 are resolved.

12.4.2.2 Compliance with 10 CFR 20.1406

During its evaluation of Revision 17 to the AP1000 DCD, the staff identified insufficient information available on the description of design features concerning its compliance with 10 CFR 20.1406 (RAI-SRP12.1-CHPB-01). The staff was in the process of developing RG 4.21, when the DCD applicant submitted TR-98, Revision 0. The staff intended for its report to identify and justify standard changes to be incorporated into the DCD. Revisions 16 and 17 incorporated the information into the various sections of the DCD, including Chapters 9, 11, and 12. The implementation of 10 CFR 20.1406 affects other SSCs described in the DCD, but that information was not sufficiently described in the other DCD chapters. The design features discussed in DCD Section 12.3 and the COL information items, added as a result of TR-98, clarify some aspects of the applicant's compliance with 10 CFR 20.1406 but do not provide the description of the program consistent with the guidance in RG 4.21.

In TR-98, Revision 0, the applicant proposed to comply with the regulation by the selection of design technology. Table TR-98-1, "AP1000 Features Applicable to 10 CFR 20.1406," in TR-98 lists specific examples (Items 19, 22, 23, 24, 25 and 26 in the table) showing how the AP1000 design complies with the portions of 10 CFR 20.1406 dealing with minimizing the generation of waste. The staff has reviewed the items listed in Table TR-98-1 pertaining to the liquid radwaste system and finds that the applicant addressed the issue of minimization of waste generation in 10 CFR 20.1406. The applicant has not committed to RG 4.21, Revision 0, but has described the design features in the licensing basis for the AP1000 to meet 10 CFR 20.1406 requirements. The staff implemented 10 CFR 20.1406 and issued RG 4.21 after Revision 16 to the DCD but before Revision 17. Therefore, the staff is providing documentation requirements for following this RG in correspondence related to Chapter 12 for COL applicants.

In a letter dated September 9, 2008, the applicant described specific features that are incorporated into the AP1000 design to comply with the requirements of 10 CFR 20.1406 or are referenced in TR-98, Revision 0, for the systems that were listed in RAI-SRP12.1-CHPB-01 (and for any other plant systems that may generate or transfer radioactive materials or waste). The staff determined the analysis is sufficient because the responses to RAI-SRP12.1-CHPB-01, as well as TR-98, provide specific information about compliance with 10 CFR 20.1406, and the staff has determined that the applicant considered the applicable design criteria. (See also Section 12.2.2 of this report).

12.4.2.3 Addition of Integrated RV Head Package and Quick-Loc Connectors

In its review of AP1000 DCD Section 12.3, the staff identified areas that needed additional information to complete an evaluation of the applicant's proposed change (RAI-SRP12.3-CHPB-01). In Tier 2, DCD Section 12.3.1.1.1, the staff noted that the applicant's description includes an integrated RV head package that combines the head lifting rig, control and gray rod drive mechanisms, lift columns, control rod drive mechanism cooling system, and power and instrumentation cabling. The applicant also replaced the conventional top-mounted instrumentation ports/conoseal thermocouple arrangement with a combination thermocouple/incore detector system.

In Revision 17 to AP1000 DCD Tier 2, Figure 12.3-1 (Sheet 8 of 16), "Radiation Zones, Normal Operation/Shutdown Nuclear Island," EL 135′-3" indicates that the RV head stand area may be a Plant Radiation Zone V [less than or equal to 10 mSv/h (1 rem/h)] when the RV head is in the stand, which is defined by Figure 12.3-1 (Sheet 1 of 16). In Revision 15, the same drawing indicated that the area for the RV head stand would be a Plant Radiation Zone II [(less than or equal to 25 μSv/h (2.5 mrem/h)]. There were no supporting calculations to show that the integrated RV head package will result in a dose rate of less than or equal to the original RV head configuration, or to show how this change is ALARA.

Table 12.4-12, "Dose Estimate for Refueling Activities," did not change as a result of the addition of the design change implementing the integrated RV head package. The use of an integrated RV head, which has been installed at several current generation facilities, minimizes the time necessary to perform disassembly and reassembly of the RV during refueling outages.

The description of the change to include the integrated RV head package lacked sufficient information to determine if the containment area radiation zones are affected or if the implementation results in an increase or decrease in the refueling dose estimates. The staff identified this as Open Item OI-SRP12.3-CHPB-01 and was tracked in the SER with open items.

In a letter dated January 21, 2010, the applicant submitted its response to Open Item OI-SRP12.3-CHPB-01.

The applicant indicated that the integrated head package would result in several changes to the radiation zones associated with the upper level of containment and the head stand area. In addition, the applicant revised Table 12.4-12 to reflect a net positive effect concerning the reduction of total dose due to refueling processes. The staff concludes that revised exposure rates and total exposure are consistent with industry data and are appropriate. In a subsequent revision to the DCD, the applicant made an appropriate change to the DCD text; therefore, both RAI-SRP12.3-CHPB-01 and the associated Open Item OI-SRP12.3-CHPB-01 are resolved.

12.4.2.4 Concrete Density for Shielding Design

During the review of Revision 17 to AP1000 DCD Tier 2, Section 2.3.2.2.9, "Spent Fuel Transfer Canal and Tube Shielding," the staff observed that the applicant decreased the assumed overall concrete density for shielding design purposes from 2354.7 kilograms per cubic meter (kg/m^3)(147 lb/ft^3) to 2242.58 kg/m^3 (140 lb/ft^3). The applicant provided no discussion in DCD Chapter 12 (in Revision 17) describing the effect of an approximate 5-percent decrease in the assumed shielding density of the transfer tube on area radiation levels during fuel movement. With the reduction in the concrete density, the applicant did not identify the basis of the parameters included in Section 12.3.2.2.9 or the reason for the change.

The applicant did not describe the radiological exposure consequences for occupationally exposed personnel nor discuss the effect on radiation zoning. The staff asked the applicant to address these concerns in RAI-SRP12.3-CHPB-03. The staff identified this as Open Item OI-SRP12.3-CHPB-03 and was tracked in the SER with open items.

In a letter dated December 15, 2009, the applicant submitted its response to RAI-SRP12.3-CHPB-03.

The applicant indicated that additional dose calculations were performed and they resulted in redesignation of several radiation zones and two rooms being designated as High Radiation Areas. An additional room now will require access controls consistent with an area greater than 1 rem per hour. Regarding the information in the applicant's response on the changes in radiation zones and controls, the staff concludes that the information provided is consistent with regulatory requirements and guidance and is appropriate. In a subsequent revision to the DCD, the applicant made an appropriate change to the DCD text, which resolves this issue; therefore, both RAI-SRP12.3-CHPB-03 and the associated Open Item OI-SRP12.3-CHPB-03 are resolved.

12.4.2.5 Computer Codes Used for Shielding Calculations

In AP1000 DCD Section 12.3.2.3, the applicant described computer codes used to determine design and operational dose rates for the plant areas located on the AP1000 nuclear island. DCD Reference 22 in Section 12.3 identified the point kernel code, Microshield 4, as being a personal computer version with a menu-driven use interface of the mainframe calculation code QAD. The staff issued RAI-SRP12.3-CHPB-04 requesting additional information to support verification and validation of this computer code as it is used in the AP1000 DCD dose rate calculations.

In its December 15, 2009, response to the staff's RAI, the applicant stated that they had reviewed the AP1000 shielding calculations and updated the DCD to reflect the software used in the most recent shielding analyses of the plant. The applicant revised DCD Reference 22 to cite use of the updated Version 6.20 of the Microshield code. Microshield Version 6.20 is based upon data from the American National Standards Institute (ANSI) standard for gamma radiation attenuation and buildup (ANSI/American Nuclear Society (ANS)-6.4.3-1991, "Gamma-Ray Attenuation Coefficients and Buildup Factors for Engineering Materials") and is widely used in the nuclear industry. The applicant stated that they have verified this software code by comparing the software results from this code with the results for reference problems provided in ANSI/ANS-6.6.1-1979, "American National Standard for Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants." While Microshield Version 6.20 is used for non-complex geometries, the applicant modified the DCD to state that they use Monte Carlo or discrete ordinate methods for radiation analysis for complex geometries. The staff finds the applicant's response to RAI-SRP12.3-CHPB-04 to be acceptable because the applicant modified the DCD to clarify and better describe the type of computer code software referenced to perform simple gamma radiation shielding calculations for the AP1000 design and also provided verification and validation of this code's output. Based on the applicant's response, RAI-SRP12.3-CHPB-04 is resolved.

12.4.3 Conclusions

Based on the evaluation of the AP1000 DCD information and response to RAI-SRP12.1-CHPB-01, and the resolution of Open Item OI-SRP12.1-CHPB-01 (see

Section 12.2.2), the staff concludes that the applicant properly identified design information related to its compliance with 10 CFR 20.1406 and provided an adequate demonstration that design objectives for equipment necessary to minimize contamination to the environment have been met, in accordance with 10 CFR Part 20.

Based on the evaluation of the DCD information in the applicant's response to RAI-SRP12.3-CHPB-02 and RAI-SRP12.3-CHPB-03, the staff concludes that the applicant has properly identified all design information related to its compliance with 10 CFR Part 20 and with GDC 61, and has provided an adequate demonstration that design objectives have been met for the spent fuel handling equipment, spent fuel transfer canal, and tube shielding necessary to minimize exposures and maintain personnel doses ALARA, in accordance with 10 CFR Part 20.

Based on the evaluation of the DCD information in the applicant's response to RAI-SRP12.3-CHPB-01, the staff concludes that the applicant provided a complete description of how the placement of the integrated RV head package and the revised and associated equipment in the containment building meet the acceptance criteria of NUREG-0800 Section 12.3 - 12.4.

Based on the evaluation of the DCD information in the applicant's response to RAI-SRP12.3-CHPB-04, the staff concludes that the applicant provided an adequate description of the computer codes used in the shielding calculations for the AP1000 DCD and revised the associated DCD Section 12.3.2.3 and Section 12.3.6 and as a result, meet the acceptance criteria of NUREG-0800 Section 12.3 - 12.4.

The staff's evaluation of the proposed changes (the addition of monitors with multiple detectors and the revision of the title of selected area monitors) in Tier 1, Section 3.5, "Radiation Monitoring," in Revision 17 of the DCD is in Section 11.5 of this report.

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

In its previous evaluation of AP1000 DCD Section 12.3, the staff identified acceptance criteria based on the ability of the applicant's design to meet the relevant requirements in 10 CFR Part 20, as it relates to limits on doses to persons in restricted areas; 10 CFR Part 50, as it relates to the inclusion of sufficient design information to demonstrate the objectives for equipment facility design features, shielding, ventilation, area radiation, and airborne radioactivity monitoring instrumentation; and 10 CFR Part 70, "Domestic licensing of special nuclear material," as it relates to the design of radiation protection features to ensure adequate safety under normal and abnormal operating conditions. The staff reviewed the AP1000 design for compliance with these requirements, as referenced in NUREG-0800 Section 12.3 - 12.4, and determined that the design of the radiation protection features, as documented in the AP1000 DCD, Revision 15, was acceptable because it conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 radiation protection design features as documented in the AP1000 DCD, Revision 17. The staff finds that the applicant's proposed changes do not adversely affect the ability of the AP1000 design features to meet the applicable acceptance criteria. The staff also concludes that the applicant has properly

incorporated the design changes into the appropriate sections of the AP1000 DCD. On the basis that the AP1000 radiation protection design features continue to meet all applicable acceptance criteria and that the updated AP1000 DCD properly documents the changes, the staff concludes that all of the changes related to the radiation protection features in the AP1000 system design are acceptable.

In addition, the radiation protection pertaining to the IHP and Quickloc connectors contribute to the increased standardization of this aspect of the AP1000 design. Therefore, these changes meet the finality criterion in 10 CFR 52.63(a)(1)(vii).

12.5 Dose Assessment

12.5.1 Summary of Technical Information

The staff approved DCD Section 12.4, "Dose Assessment," in Revision 15 to the certified design. In Revision 17 to the AP1000 DCD there were no technical changes. This review addresses the anticipated occupational radiation exposure from normal operation and anticipated inspections and maintenance.

12.5.2 Evaluation

The staff reviewed all changes to the "Dose Assessment" section in accordance with NUREG-0800 Section 12.3 - 12.4. The staff also reviewed all changes in the AP1000 DCD, Revision 17. Information presented in the TRs support all changes in the DCD.

NUREG-1793 documents the regulatory basis for AP1000 DCD Section 12.4, Revision 15. The staff reviewed the proposed changes to DCD Section 12.4 against the applicable acceptance criteria in NUREG-0800 Section 12.3 - 12.4. The following evaluation discusses the results of the staff's review.

12.5.2.1 Summary of Changes

The staff reviewed the applicant's supporting documentation and determined that there were no supporting calculations that showed the integrated RV head package would result in a dose rate of less than or equal to the original RV head configuration, or that the change was ALARA. Table 12.4-12 did not reflect increased dose (person-sievert (person-rem)) as a result of the addition of the design change implementing the integrated RV head package. The use of an integrated head, which has been installed at several current generation facilities, minimizes the time necessary to perform disassembly and reassembly of the RV during refueling outages. The staff asked the applicant to address these issues in RAI-SRP12.3-CHPB-01. This was addressed as part of the applicant's response to Open Item OI-SRP12.3-CHPB-01 and was found to be acceptable. (See also Section 12.4.2.3 of this report).

The staff concludes that COL Action Item 12.5-1 was not technical in nature and has not changed from the AP1000 DCD, Revision 15.

12.5.3 Conclusions

As described in Section 12.2 of this report, the response to Open Item OI-SRP12.3-CHPB-01 was found to be acceptable and the staff considers Open Item OI-SRP12.3-CHPB-01 to be resolved. In NUREG-1793 and its Supplement 1, the staff documented its conclusion that the

AP1000 design and DCD (up to and including Revision 15) were acceptable and that the applicant's application for design certification met the requirements in Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design. With the exception of the changes addressed in Open Item OI-SRP12.3-CHPB-01, none of the changes described alter or affect the dose assessment in Revision 15 to the AP1000 DCD. Therefore, the staff requires no technical evaluation of the changes described in DCD Tier 2, Section 12.4.

12.6 Health Physics Facilities Design

This section describes the staff's evaluation and findings of the AP1000 DCD, Revisions 16 and 17. The staff originally reviewed Revision 16 and issued several RAIs, but the applicant issued Revision 17 before the staff could close out the RAIs for the previous revision. This review and evaluation, therefore, encompasses both revisions.

12.6.1 Summary of Technical Information

The staff approved Section 12.6, "Health Physics Facilities Design," of the AP1000 DCD, Revision 15, in the certified design. In Revision 17 to the AP1000 DCD, the applicant proposed one change, described below, with associated TR changes. This review addressed system design and performance aspects only of the health physics facilities design.

In DCD Tier 2, Section 9.1.2, "Spent Fuel Storage," and Section 9.1.4, "Light Load Handling System," the applicant proposed to increase the minimum allowable water depth above the active fuel region in a fuel assembly to 2.89 m (9.5 ft) when an assembly is being transferred in a SFHM. The applicant stated that the proposed increase in minimum water depth was sufficient to ensure that the personnel radiation exposures would be limited to less than or equal to 25 μSv/h (2.5 mrem/h) on the SFHM. The applicant has documented these changes in TR-121, Revision 0.

12.6.2 Evaluation

The staff reviewed the technical changes to the health physics facilities design, in accordance with NUREG-0800 Section 12.3 - 12.4. The staff reviewed all changes identified by change marks in the AP1000 DCD, Revision 17. The information presented in the TRs support all of the technical changes in the DCD.

NUREG-1793 documents the regulatory basis for AO1000 DCD Section 12.5, Revision 15. The staff has reviewed the proposed changes to DCD Section 12.5 against the applicable acceptance criteria of NUREG-0800 Sections 12.3 - 12.4 and 12.5. The following evaluation discusses the results of the staff's review.

12.6.2.1 The Results of Spent Fuel Water Level and Dose

In TR-121, the applicant proposed to change the minimum required depth of water above the active fuel region in a fuel assembly to minimize the exposure from direct radiation to personnel operating equipment on the SFHM. In its review, the staff requested additional information providing the dose rate analysis that was based on an actual increase in the water level from approximately 2.59 m (8.5 ft) to 2.89 m (9.5 ft) above the actual fuel in a fuel assembly when in a SFHM. The applicant responded to the RAI related to TR-121 in a letter dated October 4, 2007, and described the assumptions used in its calculations for the exposure of

workers adjacent to the fuel handling areas (RAI-TR121-CHPB-01, RAI-TR121-CHPB-03, and RAI-TR121-CHPB-04). The applicant described the potential radiological effects and dose estimates associated with the reduction of the minimum water level over active fuel in the refueling area and the SFP. Revision 16 to the AP1000 DCD incorporated these changes.

In its review of DCD Section 12.5, the staff identified areas in which the additional information provided by the applicant in a letter dated October 4, 2007, determined that the initial depth of water was in error. The SFHM design actually provided 2.59 m (8.5 ft) depth of water over the active fuel in a fuel assembly when initially proposed. In the applicant's response provided, the elevation of the top of the active fuel was unchanged since the SFP water level was increased by 0.3 m (12 inches). This was necessary to ensure that the exposure rates on the bridge deck (where operating personnel would normally be located) were less than 25 μSv/h (2.5 mrem/h). The information provided, and a review of the requisite guidance in NUREG-0800 Sections 12.3 - 12.4 and 12.5, and in RG 1.13, "Spent Fuel Storage Facility Design Basis," Revision 2, allowed the staff to complete its evaluation of the applicant's change. Further discussion on the water lever change is included in Section 12.4.2.1 of this report.

In DCD Tier 2, Section 9.1.4, the staff noted that a potential exists for the movement of active fuel above the required minimum water depth, if the applicant uses an auxiliary hoist in conjunction with a specialized spent fuel handling tool (SFHT) to reach the approximately 25 percent of the SFP rack spaces that are not accessible using the SFHM. The applicant's response to TR-121, RAI-SRP9.1.4-SBPB-04 describes this activity. The applicant's responses to the staff's RAI are discussed below.

Section 9.1.4.3.7 of the AP1000 DCD states that:

> The three fuel handling devices used to lift spent fuel assemblies are the refueling machine, fuel handling machine, and the spent fuel handling tool. Both the refueling machine and fuel handling machine contain positive stops which prevent the fuel assembly from being raised above a safe shielding height.

DCD Section 9.1.4.3.3 invokes the design of the refueling machine for the SFHM; DCD Section 9.1.4.3.1 states that, because of "mechanical or failure tolerant electrical interlocks or redundant electrical interlocks," the "refueling machine is restricted to raising a fuel assembly or core component to a height at which the water provides a safe radiation shield."

The latter statements imply that, when using the SFHT, there are no positive stops to prevent the fuel assembly from being raised above a safe shielding height. The SFHT with an auxiliary hoist will apparently be used for at least 25 percent of the SFP storage cells, based on the information in TR-121. In TR-121, Revision 0, the applicant stated the following:

> ...due to the radius of the FHM manipulator mast and the proximity to the SFP walls, approximately 25 percent of the SFP storage cells cannot be serviced by the mast crane. Also, there are instances where fuel inspection and/or fuel repair require the fuel to be moved from the SFP storage racks to the designated fuel inspection or fuel repair workstation. These non-normal fuel transfer operations are performed using the Spent Fuel Handling Tool (SFHT). The SFHT is a long handled tool which latches onto the fuel assembly top nozzle via manually actuated grippers. Lifting of the SFHT and attached fuel assembly is performed using an auxiliary hoist on the FHM.

The DCD does not describe any interlocks related to the movement of fuel assemblies when using the auxiliary hoist.

In a letter dated June 26, 2008, the applicant submitted a response to RAI-SRP9.1.4-SBPB-04 on fuel handling equipment. The response stated that the refueling machine and the SFHM will contain positive stops to prevent the fuel assembly from being raised above a safe shielding point. The SFHT will only be used in conjunction with the refueling machine and the SFHM. The applicant's response also stated that it would revise AP1000 DCD Tier 1, Section 2.1.1, page 2.1.1-1, to limit the lift height of the refueling machine mast and SFHM mast to maintain the minimum required depth of water shielding. DCD Tier 1, Table 2.1.1-1, includes Item 5 of the ITAAC to describe the acceptance criteria for this design commitment. The staff's evaluation of this Tier 1 change is in Section 9.1.4 of this report.

12.6.2.2 Documentation of Compliance with 10 CFR Part 20

The staff finds that these changes do not affect the design and performance aspects of the health physics facilities, as previously reviewed in NUREG-1793, Section 12.6.

12.6.3 Conclusions

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

In its previous evaluations of AP1000 DCD Section 12.5, regarding the health physics facilities design, the staff identified acceptance criteria based on the ability of the design to meet the relevant requirements in 10 CFR 20.1302, "Compliance with dose limits for individual members of the public," as it relates to limits on doses to persons in occupied areas, and GDC 61, as it relates to the design of spent fuel storage and handling to ensure adequate safety under normal and postulated accident conditions. The staff reviewed the AP1000 health physics facilities design for compliance with these requirements, as referenced in NUREG-0800 Section 12.5, and determined that the design of the health physics facilities, as documented in the AP1000 DCD, Revision 15, was acceptable because it conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 health physics facilities design as documented in the AP1000 DCD, Revision 17. The staff concludes that the applicant's proposed changes do not affect the ability of the AP1000 health physics facilities design to meet the applicable acceptance criteria. The staff also concludes that the applicant has properly incorporated the design changes into the appropriate sections of the AP1000 DCD, Revision 17. On the basis that the AP1000 health physics facilities design continues to meet all applicable acceptance criteria and the updated AP1000 DCD properly documents the changes, the staff concludes that all of the changes related to the system design of the AP1000 health physics facilities are acceptable.

13. CONDUCT OF OPERATIONS

13.3 Emergency Planning

13.3.1 Introduction

In Revision 17 to the AP1000 design control document (DCD), the applicant proposed changes to the annex building Technical Support Center (TSC). The specific DCD changes include: (1) renaming the TSC area in the annex building from the Main TSC Operations Area to the Control Support Area (CSA); (2) removing the identification of the specific TSC location from Tier 1 DCD information; (3) identifying the TSC location in the CSA as Tier 2; and (4) providing additional Tier 1 and Tier 2 DCD conforming changes to reflect the new TSC and CSA designations. The technical justification for the proposed changes is provided in technical report (TR)-107, "AP1000 Technical Support Center," APP-GW-GLR-107, Revision 1, dated June 14, 2007.

13.3.2 Regulatory Basis

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 52.48, "Standards for review of applications," the Nuclear Regulatory Commission (NRC) staff reviewed the AP1000 DCD for compliance with the standards set out in 10 CFR 50.47(b)(8), "Emergency plans," and 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV.E, as those standards are technically relevant to the proposed generic DCD changes for the TSC area. Associated TSC guidance is in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Sections 13.3, "Emergency Planning," and 14.3.10, "Emergency Planning – Inspections, Tests, Analyses, and Acceptance Criteria," 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," NUREG-0696, "Functional Criteria for Emergency Response Facilities," and Supplement 1 to NUREG-0737, "Requirements for Emergency Facilities." The specific criterion that applies to the changes evaluated in this section is 10 CFR 52.63(a)(1)(iii), in that the proposed changes reduce unnecessary regulatory burden and maintain protection to public health and safety and the common defense and security.

13.3.3 General Description of Facilities

The TSC provides an area and resources for use by personnel providing plant management and technical support to the plant operating staff during emergency evolutions. In addition, the TSC relieves operators of peripheral duties and communications not directly related to reactor system manipulations and prevents congestion in the control room. Revision 17 of the AP1000 DCD identifies the TSC as the Main TSC Operations Area (Room 40403) in the annex building at Elevation 117'-6", adjacent to the passage from the annex building to the nuclear island control room.

In Section 13.3, "Emergency Planning," of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," the staff evaluated the TSC information in Revision 15 of the AP1000 DCD and found that it meets the applicable regulatory requirements in 10 CFR 50.47(b)(8) and 10 CFR Part 50, Appendix E, Section IV.E.

10 CFR Part 52, "License, certifications, and approvals for nuclear power plants," Appendix D, "Design Certification Rule for the AP1000 Design," constitutes the standard design certification (DC) for the AP1000 design and incorporates by reference the associated generic DCD, which includes the Tier 1 and Tier 2 information.

As described in TR-107, the name of the AP1000 annex building TSC (i.e., Main TSC Operations Area) is changed to the CSA, and the DCD is modified to reflect the change. In addition, the identification of the specific TSC location is removed from Tier 1 DCD Section 3.1, "Emergency Response Facilities," and Table 3.1-1, "Inspections, Tests, Analyses, and Acceptance Criteria" (ITAAC). However, TR-107 does not affect the TSC functional requirements and criteria, which are retained and must be met regardless of its actual location. The changes are intended to facilitate a combined license (COL) applicant describing a site-specific location for its TSC, independent of the AP1000 annex building, without the need for an exemption to a Tier 1 AP1000 TSC location. These changes will also ease the NRC's review burden by eliminating the need for such individual COL applicants to apply for exemptions from the standardized DCD wording.

The changes in TR-107 provide flexibility, in that they will allow COL applicants who reference the AP1000 certified design to either use the CSA as the TSC (without departing from the DCD), or designate an alternative TSC location in accordance with the change process in 10 CFR Part 52, Appendix D, Section VIII. The CSA is designed so that it may be used as a TSC, if desired. The generic DCD Revision 17 location for the TSC is maintained in the annex building CSA, and was designated as Tier 2* information in DCD Tier 2, Section 18.8.3.5, "Technical Support Center Mission and Major Tasks," rather than Tier 1 information in DCD Section 3.1. The other DCD Tier 1 requirements associated with the TSC are unaffected by this change, and will be subject to the applicable Tier 1 change control process.

The staff's position is that the DCD Section 18.8.3.5 change should be to Tier 2, rather than Tier 2*. (A change from Tier 1 to Tier 2 would be governed by the same regulatory basis, described above, as a change from Tier 1 to Tier 2*.) This is because the NRC has previously used the Tier 2* designation for DCD information where there is a reasonable expectation of a change over the lifetime of the facility (e.g., a fuel change). The nature of the information is such that the NRC must review and approve the proposed change prior to the change being made. As another example, the Tier 2* designation would be appropriate for information relating to detailed design methodologies and evaluation criteria. Such examples could result in design changes where the safety of the completed design may not be readily apparent. In regard to the AP1000 DCD, once the TSC is built, it is unlikely that it will be moved. Thus, the staff concludes that the Tier 2 designation for the TSC location is more appropriate than Tier 2*. For a Tier 2 TSC DCD location designation, a proposed change to the TSC location (i.e., to a location other than the CSA) by an applicant (or licensee) would require a departure from the certified design.

The staff determined that the applicant must change the TSC location designations in the DCD from Tier 2* to Tier 2. The other DCD Tier 1 requirements associated with the TSC are unaffected by this change, and will be subject to the applicable Tier 1 change control process. The staff identified resolution of this issue as Open Item OI-TR107-NSIR-07. In its January 27, 2010, response to Open Item OI-TR107-NSIR-07, the applicant changed the TSC location designation in DCD Section 18.8.3.5 from Tier 2* to Tier 2, and reflected that change in Revision 19 of the AP1000 DCD. For the reasons discussed above, the staff finds the change acceptable, and Open Item OI-TR107-NSIR-07 is, therefore, resolved.

13.3.4 Conclusion

Based on the above evaluation, the staff concludes that the design changes in TR-107, which are reflected in Revision 19 of the AP1000 DCD, are acceptable because the TSC continues to meet the requirements in 10 CFR 50.47(b)(8) and Section IV.E of Appendix E to 10 CFR Part 50.

13.5 Plant Procedures

13.5.1 Summary of Technical Information

In Revision 19 to the AP1000 DCD, the applicant proposed to partially resolve COL Information Item 13.5-1 in TR-70, "Plant Operations, Surveillance, and Maintenance Procedures" (APP-GW-GLR-040), Revision 1 and in the DCD by addressing; normal operating, abnormal operating, emergency operating, refueling and outage planning, alarm response, administrative, maintenance, inspection, test, and surveillance procedures, as well as the procedures that address the operation of post-72-hour equipment. The COL applicant will address operational and maintenance programmatic issues to resolve COL Information Item 13.5-1.

13.5.2 Evaluation

NUREG-0800 Section 13.5.2.1 states that the applicant should describe its program for developing the operating procedures and that the staff will review the applicant's program for developing and implementing the operating procedures. The staff reviewed TR-70, Revision 1, and its associated references, and determined that it described a process to manage the development, review, and approval of these procedures, but did not clearly provide the requested description. The staff submitted request for additional information (RAI)-SRP13-COLP-01 to ask that the applicant clarify the program description for developing and implementing the operating procedures. In a letter dated July 29, 2008, the applicant stated that TR-70, Revision 1, described the program. Subsequent to the submission of TR-70, Revision 0, the staff met with the applicant to discuss procedure development issues related to the AP1000 design and to allow the staff an opportunity to audit a variety of AP1000 operations procedures. TR-70, Revision 1, addresses issues discussed in this meeting, as well as the concerns regarding NUREG-0711, "Human Factors Engineering Program Review Model," issued February 2004, that were discussed and later addressed in the staff's letter, "Summary of the April 11 and 12, 2007, Meeting to Discuss AP1000 Plant Operating Procedures," dated May 11, 2007. Along with TR-70, Revision 1, the applicant submitted the "AP1000 Writer's Guidelines for the Normal and Two-Column Format Procedures" (APP-GW-GJP-100 and 200). After reviewing the AP1000 Writer's Guidelines, which clarified procedure development, the staff finds TR-70, Revision 1 acceptable because the applicant's program description for developing and implementing the operating procedures meets the guidance in NUREG-0800 Section 13.5.2.1.

Similarly, with respect to the development of emergency operating procedures (EOPs), NUREG-0800 Section 13.5.2.1 states that the applicant should describe its program for developing EOPs, as well as the required content of the EOPs, and that the staff will review the applicant's program for developing and implementing the EOPs. The staff reviewed TR-70 and its associated references and determined that it described a process to manage the development, review, and approval of these procedures, but did not clearly provide the requested description. The staff submitted RAI-SRP13-COLP-02 to ask the applicant to describe the program for developing and implementing the EOPs. In a letter dated

July 29, 2008, the applicant stated that TR-70, Revision 1, described the program. Subsequent to the submission of TR-70, Revision 0, the staff met with the applicant to discuss procedure development issues related to the AP1000 design and to allow the staff an opportunity to audit a variety of the AP1000 operations procedures. TR-70, Revision 1, addresses issues discussed in this meeting, as well as the concerns regarding NUREG-0711 that were discussed and later addressed in the staff's letter dated May 11, 2007. Along with TR-70, Revision 1, the applicant submitted the "AP1000 Writer's Guidelines for the Normal and Two-Column Format Procedures" (APP-GW-GJP-100 and 200). After reviewing the AP1000 Writer's Guidelines, which clarified procedure development, the staff finds TR-70, Revision 1 acceptable because the applicant's program description for developing and implementing the EOPs meets the guidance in NUREG-0800 Section 13.5.2.1.

The staff reviewed DCD Section 13.5.1, which references TR-70, Revision 1, and its associated references, which the applicant submitted as a basis for closing this COL action item. During this review, the staff noted that the DCD addressed safety-related logic circuitry but did not specify which organization had responsibility for it. The staff submitted RAI-SRP13-COLP-03 and RAI-SRP13-COLP-04 to ask the applicant to specify which organization had responsibility for safety-related logic circuitry and freeze seals respectively. In a letter dated July 29, 2008, the applicant stated that it had responsibility for these issues, as described in TR-70, Revision 1. The staff finds this acceptable because the applicant's program for developing and implementing the operating procedures includes a complete description of the safety-related logic circuitry and freeze seals and thus meets the guidance in NUREG-0800 Section 13.5.2.1.

The staff reviewed DCD Section 13.5.1, which states that TR-70 partially addresses the requested COL information, which includes normal operating, abnormal operating, emergency operating, refueling and outage planning, alarm response, administrative, maintenance, inspection, test, and surveillance procedures, as well as the procedures that address the operation of post-72-hour equipment, and that the COL applicant will address operational and maintenance issues. The staff was not clear as to which operational and maintenance issues the COL applicant is responsible for or how they differ from those addressed in TR-70. The staff submitted RAI-SRP13-COLP-05 to ask the applicant to clarify these issues. In a letter dated July 29, 2008, the applicant stated that it was responsible for the development, review and approval of normal operating, abnormal operating, emergency operating, refueling and outage planning, alarm response, administrative, maintenance, inspection, test, and surveillance procedures, as well as the procedures that address the operation of post-72-hour equipment, and that the COL applicant was responsible for maintaining those procedures after their approval and acceptance, as well as for operator training in those procedures. The procedures developed and approved by the applicant are generic procedures. Site-specific procedures will be approved and maintained by the COL applicant. The staff finds this acceptable because the allocation for responsibility for developing and implementing the operating procedures meets the guidance in NUREG-0800 Section 13.5.2.1.

The staff reviewed DCD Section 13.5.1, which states that TR-70 submitted several reports to the staff. To the staff, the word "submit" means "docketed"; thus, this is not an appropriate word to use when referring to documents that have not been docketed. The staff submitted RAI-SRP13-COLP-06 to inform the applicant of this meaning. In a letter dated July 29, 2008, the applicant acknowledged the incorrect language and provided corrected text with amended language. The staff finds this acceptable because the applicant's program description now uses language that is clear and consistent with the staff's understanding.

13.5.3 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design changes are acceptable because they meet the guidance in NUREG-0800 Section 13.5.2.1. The proposed DCD changes are acceptable, pursuant to 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information.

13.6 Physical Security

13.6.1 Summary of Technical Information

This section of the AP1000 safety evaluation report (SER) documents the staff's review of the physical security aspects of the AP1000 DC application submitted to the NRC by the applicant.

In Revision 19 of the AP1000 DCD Tier 2, Section 13.6, the applicant describes the plant's physical security program, including those elements of physical protection and mitigative measures identified as being within the scope of the applicant's design. The description includes the required physical security elements of a DC application and references TRs that are part of the DC application, on physical protection and mitigative measures. It describes the design for protecting the plant against acts of radiological sabotage; specifically, the plant layout and protection of vital equipment are in accordance with 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," and applicable regulatory guidance. This safety evaluation (SE) incorporates the staff reviews of DCD Tier 2, Revision 19, Section 13.6; DCD Tier 1, Revision 19, Section 2.6.9, and Table 2.6.9-1; the applicant's RAI responses and the ITAAC for the physical security hardware and referenced safeguards TRs.

The applicant responded to several RAIs. The staff found all responses regarding regulatory requirements to be acceptable, including the applicant's responses to the RAIs associated with physical protection and the open items. All RAIs and open items are resolved and discussed in the SE that is designated as including safeguards information (SGI). The DCD and TRs identify vital equipment and vital areas; describe armed responder positions, physical security attributes (e.g., delay barrier(s) within the AP1000 design scope), their characteristics; and analyze adversarial scenarios for design-basis threats (DBTs). Because this information is security sensitive, the comprehensive physical protection SE includes SGI and is not available for public disclosure. Those persons with the correct access authorization and need-to-know may view the SGI version of the physical security SE, hereafter referred to as the "SGI SER of the AP1000," which is located in the NRC's Secure Local Area Network.

The applicant provided the design description and information related to physical protection in the following parts of the DCD: Section 13.6, Section 2.6.9, and referenced safeguards TRs.

In the AP1000 DCD Tier 1, Revision 19 and applicant RAI responses, Section 2.6.9 and Table 2.6.9-1, the applicant describes the design features and ITAAC for physical security hardware, and the design commitments for physical security hardware ITAAC within the scope of the AP1000 design.

In DCD Tier 2, Revision 19, Section 13.6, the applicant states that the "AP1000 Interim Compensatory Measures Report," the "AP1000 Enhancement Report," and the "AP1000 Safeguards Assessment Report" were separately submitted to establish the design of the AP1000 security systems.

In DCD Tier 2, Revision 19, Section 13.6.1, the applicant states that COL applicants referencing the AP1000 design will address site-specific information related to the "Physical Security Plan, Training and Qualification Plan and the Safeguards Contingency Plan," which are the responsibility of the COL applicant.

13.6.1.1 Summary of Technical Information - ITAAC

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1 information, as described in DCD Tier 2, Section 14.3, to support ITAAC for the AP1000 structures, systems, and components (SSCs).

The applicant organized the DCD Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1 Table of Contents. Site-specific structures that are not within the scope of the certified design are to be addressed by the COL applicant that references the AP1000 certified design. Along with design and program descriptions of site-specific physical protection features, the COL applicant is required to address site security provisions during construction of new reactor(s) that is either inside or co-located to an existing PA. In addition, the COL applicant would address, as applicable, the controls and measures necessary for transitioning between new and existing physical protection systems and continue to maintain in effect the site security program and controls required for implementing the protective strategy for operating power reactors. ITAAC are addressed by both the DC and by the COL applicants incorporating its site-specific ITAAC to meet NRC requirements.

The design bases or supporting security analyses and assumptions related to the design descriptions of security-related features incorporated as AP1000 standard design are provided in SGI TR-94, "AP1000 Safeguards Assessment Report," APP-GW-GLR-066. The staff reviewed the DCD Tier 1 information provided by the applicant in accordance with NUREG-0800 Section 14.3.12, "Physical Security Hardware – Inspections, Tests, Analyses, and Acceptance Criteria" Revision 1, January 2010.

The applicant provided design descriptions and information related to physical protection systems or features in the following portions of the DCD and referenced TRs:

AP1000 DCD Tier 1, Chapter 2, "System Based Design Descriptions and ITAAC," Section 2.6.9, "Plant Security System," describes the design features and ITAAC for security hardware for the AP1000 design. Table 2.6.9-1, "Inspections, Tests, Analyses, and Acceptance Criteria" describes the design commitments for security hardware that are within the scope of the AP1000 design.

Tier 2, Chapter 1, Section 1.2, "General Plant Description," and Section 1.2.1, "Design Criteria, Operating Characteristics, and Safety Considerations," provide descriptions of the scope of the AP1000 design.

Tier 2, Chapter 13, "Conduct of Operations," Section 13.6, "Security," of the AP1000 describes physical protection systems or features incorporated as a part of the AP1000 standard design. Elements of a site-specific security program such as organization structure, training, operational program implementations, plant procedures, site-specific target sets, protective strategy, design features for security, and fitness for duty program are to be described by the COL applicant, along with an implementing schedule.

Section 14.2.9, "Preoperational Test Descriptions," identifies tests to be completed prior to operating conditions. Section 14.2.9.1.14, "Class IE DC Power and Uninterruptible Power Supply Testing" and Section 14.2.9.4.13, "Plant Communications System Testing" addresses security components of plant's lighting and intra-plant communications.

The applicant submitted TR-94, which describes the security measures credited in defending the AP1000 against a DBT in support of the application.

The applicant submitted TR-96, APP-GW-GLR-067, "AP1000 Interim Compensatory Measures Report," that include information on compliance with the various sections of the Commission Order on Interim Compensatory Measures that was issued to NRC power reactor licensees on February 25, 2002.

The applicant submitted TR-49, APP-GW-GLR-062, "AP1000 Enhancement Report," which describes design areas as physical security enhancements that will enhance the ability of a COL applicant to meet the general performance objective of 10 CFR 73.55.

The information found in these referenced reports is considered SGI and Official Use Only Security-Related Information and is protected in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," and 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements."

The applicant submitted, in Revision 19 of the AP1000 DCD and RAI responses, proposed changes to physical security hardware ITAAC in DCD Tier 1, Section 2.6.9, "Plant Security System." As a result of the staff's review, the applicant submitted additional proposed changes to DCD Tier 1, Section 2.6.9 in its responses to RAI-SRP14.3.12-NSIR-06 and RAI-SRP14.3.12-NSIR-07. The final proposed AP1000 physical security hardware ITAAC was provided in the response to RAI-SRP14.3.12-NSIR-07, Revision 1.

13.6.2 Regulatory Basis – Physical Security

The NRC's regulations for protecting nuclear power reactors in 10 CFR Part 73, "Physical protection of plants and materials," include specific security and performance requirements that, when implemented correctly, are designed to protect nuclear power reactors against acts of radiological sabotage, prevent the theft or diversion of special nuclear material, and protect SGI against unauthorized release.

Regulations in 10 CFR 73.1(a)(1), "Purpose and scope," require the establishment of physical protection systems to protect special nuclear material against the DBT for radiological sabotage, and 10 CFR 73.55 describes the required physical protection for licensed activities. Pursuant to 10 CFR 50.34(c)(2), "Contents of applications; technical information," and 10 CFR 52.79(a)(35)(i), "Contents of applications; technical information in final safety analysis report," applicants must prepare and maintain security plans that describe the security-related actions they will take to protect their facilities against acts of radiological sabotage.

Subpart B of 10 CFR 52.47, "Contents of applications; technical information," requires that information submitted for a DC include performance requirements and design information sufficiently detailed to permit an applicant to prepare procurement specifications and construction and installation specifications. According to 10 CFR Part 52.48, the NRC will

review applications filed under 10 CFR Part 52 for compliance with the standards set forth in 10 CFR Part 73.

The AP1000 design descriptions, commitments, and acceptance criteria for the security features, including the plant's layout and protection of vital equipment, as described in the DC application, are based on meeting the relevant requirements of the following Commission regulations:

- 10 CFR Part 50
- 10 CFR Part 52
- 10 CFR Part 73, Appendix B, "General Criteria for Security Personnel"; Appendix C, "Nuclear Power Plant Safeguards Contingency Plans"; Appendix G, "Reportable Safeguards Events"; and Appendix H, "Weapons Qualification Criteria"
- 10 CFR 73.1(a)(1)
- 10 CFR 73.70(f), "Records"
- 10 CFR Part 74, "Material control and accounting of special nuclear material"
- 10 CFR 100.21(f), "Non-seismic site criteria"

In its review, the staff used NUREG-0800 Section 13.6.2 to complete its AP1000 physical security DC review. The following paragraphs in 10 CFR 73.55 include acceptance criteria related to the staff's review in accordance with NUREG-0800 Section 13.6.2:

- Section (e) – Physical barriers: The licensee shall locate vital equipment only within a vital area, which, in turn, shall be located within a protected area (PA), such that access to vital equipment requires passage through at least two physical barriers (as defined in 10 CFR 73.2, "Definitions") that perform their required function in support of the licensee's physical protection program. The physical barriers at the perimeter shall be separated from any other barrier designated as a physical barrier for a vital area within the PA. Isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the PA permit observation. An intrusion detection system detects penetration or attempted penetration of the PA barrier. Isolation zones and appropriate exterior areas within the PA are illuminated. The main control room has bullet-resistant external walls, doors, ceiling, and floors. Vehicle control measures, which include vehicle barrier systems, protect against the threat of assault by land vehicles.

- Section (g) – Access control: The licensee shall control all points of personnel and vehicle access into a PA; this includes providing equipment capable of detecting firearms, explosives, incendiary devices, or other items that could be used to commit radiological sabotage, or a visual and physical search, or both. Unoccupied vital areas are locked and alarmed with activated detection systems that annunciate in both the central alarm station (CAS) and secondary alarm station (SAS) upon intrusion into a vital area. The individual responsible for the last access control function (controlling admission to the PA) must be isolated within a bullet-resisting structure.

- Section (i) – Detection and assessment systems: All alarms required pursuant to this part must annunciate and display concurrently in at least two continuously staffed onsite alarm stations, at least one of which must be protected in accordance with the requirements of the CAS. The CAS must be inside the PA, and the interior must not be visible from the perimeter of the PA. The applicant must design and equip the continuously staffed CAS and SAS so that a single act cannot disable both. At least one alarm station must maintain the ability to detect and assess alarms, initiate and coordinate an adequate response to an alarm, summon offsite assistance, and provide command and control. The CAS shall be considered a vital area and be bullet resistant, and associated onsite secondary power supplies for alarm annunciators and nonportable communication equipment must be located within vital areas. Alarm devices and transmission lines must be tamper indicating and be self-checking. Alarm annunciation on CAS/SAS computer monitoring stations shall indicate the type of alarm and its location. All emergency exits from protected and vital areas shall be alarmed and secured by locking devices.

- Section (j) – Communication requirements: Each security officer or armed-response individual shall be capable of maintaining constant communications with an individual in each continuously manned alarm station. Conventional telephone and radio- or microwave-transmitted two-way voice communications shall be established with local law enforcement authorities.

- Section (n) – Maintenance, testing, and calibration: Each applicant shall develop test and maintenance provisions for intrusion alarms, emergency alarms, communications equipment, access-control equipment, physical barriers, and other security-related devices or equipment.

The staff, in its review, used the following regulatory guidance documents:

- Regulatory guide (RG) 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," issued February 1978

- RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations," issued April 1998

- RG 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials," issued November 1973

- RG 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," issued September 1986

- RG 5.7, "Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas," Revision 1, issued May 1980

- RG 5.44, "Perimeter Intrusion Alarm Systems," Revision 3, issued October 1997

- Information Notice 86-83, "Underground Pathways into Protected Vital Areas, Material Access Areas, and Controlled Access Areas," dated September 19, 1986

- Regulatory Issue Summary 2005-04, "Guidance on Protection of Unattended Openings that Intersect a Security Boundary or Area," April 14, 2005

- "Nuclear Power Plant Security Assessment Format and Content Guide," Information Systems Laboratories, issued September 2007 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML07030054)

- SAND 2007- 5591, "Nuclear Power Plant Security Assessment Technical Manual," Sandia National Laboratories, issued September 2007 (ADAMS Accession Number ML072620172)

The staff evaluates the following specific acceptance criteria for ITAAC:

- 10 CFR 73.1, as it relates to the prescribed requirements for the establishment and maintenance of a physical protection system and for protection against the DBT of radiological sabotage

- 10 CFR 73.55, as it relates to the requirements for physical protection against radiological sabotage of licensed activities in nuclear power reactors

- 10 CFR 73.70(f), as it relates to the requirements specific to alarm annunciation records

- 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC will be built and operated in accordance with the DC; the provisions of the Atomic Energy Act of 1954, as amended (the Act); and NRC regulations

The COL applicant referencing a certified design is responsible for the site-specific security operational programs to meet the requirements in 10 CFR 50.34(c)(2) or 10 CFR 52.79(a)(35)(i) and 10 CFR 52.79(a)(36)(i), (ii), and (iii). This is satisfied, in part, by describing a physical protection system and administrative programs and procedures for implementing a site-specific protective strategy that demonstrates high assurance that the plant is protected against the DBT. The site-specific physical protection system must be reliable and available and must implement defense-in-depth to provide a high assurance of protection. The following specific and performance requirements describe the security operational programs and the physical protection system: 10 CFR Part 26, "Fitness for duty programs"; 10 CFR 73.55; 10 CFR 73.56, "Personnel access authorization requirements for nuclear power plants"; 10 CFR 73.57, "Requirements for criminal history records checks of individuals granted unescorted access to a nuclear power facility or access to Safeguards Information"; 10 CFR 73.70; 10 CFR 73.58, "Safety/security interface requirements for nuclear power reactors"; and 10 CFR Part 74. Regulations in 10 CFR 52.79(a)(36)(i) or 10 CFR 50.34(d) and Appendix C to 10 CFR Part 73 require COL applicants to submit the security program and planning for a safeguards contingency. The performance and specific requirements in Appendix B to 10 CFR Part 73 requires COL applicants to submit a training and qualification plan and implement the training and qualification requirements for readiness of security personnel and responders.

Within this context, the DC applicant must address those elements or portions of physical protection systems that are considered within the scope of the design. However, the DC applicant may include descriptions of security systems or hardware, with supporting technical

bases that are beyond the physical configuration for the scope of the design, provided that it is clearly stated that they are within the scope of the DC.

The staff used NUREG-0800 Section 14.3.12, "Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria," to review the applicant's ITAAC submittal. Section 13.6.3.1 of this SER documents the staff's evaluation of these DCD changes.

13.6.2.1 Regulatory Basis - ITAAC

The NRC regulation for protecting nuclear power reactors is provided in 10 CFR Part 73. The regulation includes specific security and performance requirements that, when adequately implemented, are designed to protect nuclear power reactors against acts of radiological sabotage, prevent the theft or diversion of special nuclear material, and protect safeguards information against unauthorized release.

The performance requirements for the physical protection of nuclear power reactors are provided in 10 CFR 73.1(a)(1), which bounds the adversarial characteristics of the DBT, and 10 CFR 73.55. Pursuant to 10 CFR 50.34(c)(2), 10 CFR 50.34(d), 10 CFR 50.54(p)(1), "Conditions of licenses," and 10 CFR 50.54(p)(2), 10 CFR 73.55(c)(4), and as referenced in 10 CFR Part 52, applicants are required to prepare and maintain security plans that describe the security-related actions that they will take to protect their facilities against acts of radiological sabotage.

10 CFR 52.47(b)(1) requires a DC applicant to contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a plant that incorporates the DC is built and will operate in accordance with the DC.

Regulatory requirements and acceptance criteria related to physical protection systems or hardware are, in part, applicable to the DC (i.e., within scope of the design) or may only be applicable to a COL applicant (outside of a DC design scope) and are identified as follows, as specified in NUREG-0800 Section 14.3.12, "Physical Security Hardware – Inspections, Tests, Analyses, and Acceptance Criteria" Revision 1, January 2010.

The COL applicant is required to describe commitments for establishing and maintaining a physical protection system (engineered and administrative controls), organization, programs, and procedures for implementing a site-specific strategy that, if adequately implemented, provides a high assurance of protection of the plant against the DBT. The site-specific physical protection system described must be reliable and available and implement the concept of defense-in-depth protection in order to provide a high assurance of protection. The security operational programs and the physical protection system are required to meet specific and performance requirements of 10 CFR Part 26; 10 CFR Part 74; 10 CFR 73.55; 10 CFR 73.56; 10 CFR 73.57; and 10 CFR 73.70. The COL applicant's security program and planning for safeguards contingency are required to meet 10 CFR 50.34(d) and 10 CFR Part 73, Appendix C. The training and qualification program for readiness of security personnel and responders are required to meet performance and specific requirements of 10 CFR Part 73, Appendix B. Within this context, the DC applicant must address those elements or portion of physical protection systems or features that are considered within the scope of the certified portion of the design. The technical basis for physical protection hardware within the scope of the certified portion of the design provides the basis for ITAAC verification and closures. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD

text, which resolves the list of ITAAC numbers. Therefore, all ITAAC numbers are consistent with NUREG-0800 Section 14.3.12.

13.6.3 Evaluation – Physical Security

The staff reviewed AP1000 DCD Tier 1, Revision 19, Section 2.6.9 and Table 2.6.9-1; and DCD Tier 2, Revision 19, Section 13.6, applicant RAI responses, and referenced safeguards TRs.

In its review of the referenced safeguards TRs, the staff identified areas in which it needed additional information to complete the review of the applicant's physical security design. The applicant responded to the staff's RAIs as discussed below.

The staff reviewed the applicant's submittals to determine if its consideration of physical security in the AP1000 design was acceptable.

The staff identified several RAIs relating to target sets for the purpose of reviewing the applicant's physical protection program. The applicant provided design details as background information to assist a licensee with the development of site-specific target sets analyses. The staff evaluated the applicant's responses, and found them to be acceptable for the DC review of the AP1000 physical protection program. The applicant stated in TR-94 that target sets were created to aid in the development of the AP1000 physical security system, and that final target sets will be developed by the COL applicant. Upon the completion of its review, the staff determined that the applicant adequately addressed regulations and the NUREG-0800 acceptance criteria that were identified as within the scope of their design.

Combined License Information Items

The staff reviewed the AP1000 description and commitment for the COL information item that COL applicants referencing the AP1000 certified design must address.

Acceptance Criteria

Pursuant to 10 CFR 52.47(b)(1), a DC applicant is required to submit the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC will be built and operated in accordance with the DC, the provisions of the Act, and the NRC's regulations.

In addition to ITAAC, the staff also reviewed the following information that was submitted by an applicant for the physical security design. The following information was provided by an applicant to meet the acceptance criteria identified in Section 13.6.2, "Regulatory Basis," for the physical security DC.

As required by 10 CFR 73.55(e)(9)(i), a DC applicant shall identify vital areas and a list of vital equipment, by location.

As required by 10 CFR 73.55(e)(9)(v) and (vi), a DC applicant shall identify the control room as a vital area and secondary power supply (for alarm annunciator equipment and nonportable communications) as within a vital area.

As required by 10 CFR 73.55(e)(9)(iii), a DC applicant shall provide the design of the locks and alarms of all unoccupied vital areas.

As required by 10 CFR 73.55(e)(5), a DC applicant shall provide the design describing the bullet resistance of the control room and the CAS.

As required by 10 CFR 73.55(g)(1)(i)(B), a DC applicant should identify locks used to protect the facility and special nuclear material as manipulative resistant.

Evaluation of COL Information

The staff evaluated the COL information item identified in its review of the AP1000 DC application and included in DCD Tier 2, Revision 19, Section 13.6.1. COL information items and applicant responses are those physical security requirements from the above six acceptance criteria that are either met partially or are not addressed by the DC applicant. The staff's evaluation determines whether the DC applicant adequately describes those physical security requirements so that a COL applicant would be able to address them during the COL licensing process. The DC applicant need not identify as COL information items those physical security elements required by regulation. However, for physical security elements partially met in the DC application, the DC applicant should explicitly identify which part of the requirement was met and which part the COL applicant referencing the design will be required to meet.

In the COL information item in DCD Section 13.6.1, the applicant states: Combined License applicants referencing the AP1000 certified design will address site-specific information related to the physical security, contingency, and training and qualification plans.

On the basis of the staff review, the COL information item appropriately addresses interface requirements between the referenced AP1000 physical protection system design and the COL applicant.

13.6.3.1 Evaluation – ITAAC

The applicant submitted the following ITAAC for detection and assessment hardware in Revision 17 of the AP1000 DCD Tier 1, Section 2.6.9, "Plant Security System," which addresses ITAAC consistent with NUREG-0800 Section 14.3.12. The numbering system below corresponds to the applicable elements of NUREG-0800 Section 14.3.12.

2. Physical barriers for the PA perimeter are not part of vital area barriers.

3. Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the PA that allows 20 feet of observation on either side of the barrier. Where permanent buildings do not allow a 20-foot observation distance on the inside of the PA, the building walls are immediately adjacent to, or an integral part of, the PA barrier.

4. An intrusion detection system can detect penetration or attempted penetration of the PA barrier.

5. Isolation zones and exterior areas within the PA are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.

6. The external walls, doors, ceiling, and floors in the main control room, the CAS, and the last access control function for access to the PA are bullet resistant.

9. An access control system with numbered picture badges is installed for use by individuals who are authorized access to PAs without escort.

10. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the CAS and SAS upon intrusion into a vital area.

11. Security alarm annunciation occurs in the CAS and in at least one other continuously manned station not necessarily onsite.

14. Equipment exists to record onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.

After the review of the ITAAC for detection and assessment hardware, the staff determined that the applicant submitted ITAAC within the DCD that are not within the scope of the DC, and that should be submitted as part of a COL application. In order to complete its review, in RAI-SRP14.3.12-NSIR-06, the staff requested that the applicant revise the physical security hardware ITAAC in Tier 1 of the DCD consistent with NUREG-0800 Section 14.3.12.

In its response to RAI-SRP14.3.12-NSIR-06, the applicant proposed to revise DCD Tier 1, Section 2.6.9 to delete any items that are outside the scope of the certified design. The applicant removed ITAAC Items 3, 4, and 9, which will be submitted by COL applicants. The applicant also removed ITAAC Item 2, as the PA barrier will be addressed by the COL applicant in an ITAAC that will be provided for the site-specific design elements of plant security. In addition, the applicant also revised Item 6.

As a result of changes to regulations for "Power Reactor Security Requirements," effective May 26, 2009, in RAI-SRP14.3.12-NSIR-07, the staff requested that the applicant submit revised AP1000 ITAAC that conform to the 10 CFR Part 73 Power Reactor Security Requirements Final Rule.

In its response to RAI-SRP14.3.12-NSIR-07, dated October 15, 2009, the applicant revised the AP1000 physical security hardware ITAAC to be consistent with the Power Reactor Security Requirements Final Rule of May 26, 2009. ITAAC Item 6 was revised to include the SAS. The applicant also revised ITAAC Item 11 to Item 11(a) and included the addition of video assessment capability and the SAS. In addition, the applicant added ITAAC Items 11(b) and 11(c) as follows:

11(b) The central and secondary alarm stations are located inside a protected area, and the interior of both alarm stations is not visible from the perimeter of the protected area.

11(c) The central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to assess alarms and communicate with onsite and offsite response personnel.

Based on discussions at an RAI closed meeting on November 24, 2009 between the applicant and the staff, the applicant submitted revised AP1000 physical security hardware ITAAC in its Revision 1 response to RAI-SRP14.3.12-NSIR-07, dated December 16, 2009. The applicant revised ITAAC Item 6 to add the minimum bullet resistance for the Main Control Room, CAS, and SAS. In addition, the applicant revised ITAAC Item 11(a) to add that alarm annunciation and video assessment is displayed concurrently, and that video recording with real time playback capability can provide assessment of activities before and after each alarm. This revision of ITAAC Item 11(a) also reads on video image recording in ITAAC Item 4(b).

On the basis of its review, the staff finds the revised ITAAC for detection and assessment hardware to be acceptable because it is in conformance with the staff's definition of physical security hardware ITAAC that is within the scope of the DC, and the ITAAC are sufficient to verify that the hardware, as finally installed and constructed, will function as designed.

The applicant submitted the following ITAAC for delay or barrier design features in Revision 17 of the AP1000 DCD Tier 1, Section 2.69, "Plant Security System," which addresses ITAAC consistent with NUREG-0800 Section 14.3.12:

1.
a) Vital equipment is located only within a vital area.
b) Access to vital equipment requires passage through at least two physical barriers.

7. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.

8. Access control points are established to:
a) Control vehicle and personnel access into the PA.
b) Detect firearms, explosives, and incendiary devices at the PA personnel access points.

13. Security alarm devices including transmission lines to annunciators are tamper-indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms and emergency exit alarm) and location.

As a result of its review of the ITAAC for delay or barrier design features, the staff determined that the applicant submitted ITAAC within the AP1000 DCD that are not within the scope of the DC and that should be submitted as part of a COL application. In order to complete its review, in RAI-SRP14.3.12-NSIR-06, the staff requested that the applicant revise the physical security hardware ITAAC in Tier 1 of the DCD in accordance with NUREG-0800 Section 14.3.12.

In its response to RAI-SRP14.3.12-NSIR-06, the applicant proposed to revise the AP1000 DCD Tier 1, Section 2.6.9 by deleting ITAAC Items 8(a) and 8(b). The applicant indicated that ITAAC Items 8(a) and 8(b), access control points, would be submitted by the COL applicants.

As noted above, as a result of changes to regulations for "Power Reactor Security Requirements," effective May 26, 2009, in RAI-SRP14.3.12-NSIR-07, the staff requested that

the applicant submit revised AP1000 ITAAC that conform to the 10 CFR Part 73 Power Reactor Security Requirements Final Rule.

In its response to RAI-SRP14.3.12-NSIR-07, the applicant revised the AP1000 physical security hardware ITAAC to be consistent with the Power Reactor Security Requirements Final Rule of May 26, 2009. The applicant revised ITAAC Item 13 to Item 13(a) and added ITAAC Item 13(b) to include the requirement for intrusion detection and assessment systems to provide visual display and audible annunciation of the alarm in both the CAS and SAS.

Based on discussions at an RAI closed meeting on November 24, 2009 between the applicant and the staff, the applicant submitted revised AP1000 physical security hardware ITAAC in its Revision 1 response to RAI-SRP14.3.12-NSIR-07. The applicant revised ITAAC Item 1(b) to state "Access to vital equipment requires passage through the vital area barrier." Because the applicant considered the PA barrier outside of the design scope, it indicated that the requirement in 10 CFR 73.55(e)(9)(i) for two physical barriers would be the responsibility of COL applicants. The applicant added ITAAC Item 13(b) as follows:

13(b) Intrusion detection and assessment systems provide visual displays and audible annunciation of alarms in the central and secondary alarm station.

On the basis of its review, the staff finds the revised ITAAC for delay or barrier design features to be acceptable, because it is in conformance with the staff's definition of physical security hardware ITAAC that is within the scope of the DC, and sufficient to verify that the hardware, as finally installed and constructed, will function as designed.

The applicant submitted the following ITAAC systems, hardware, or features facilitating security response and neutralization in Revision 17 to the AP1000 DCD Tier 1, Section 2.6.9, "Plant Security System," which addresses ITAAC consistent with NUREG-0800 Section 14.3.12:

12. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

15. Emergency exits through the PA perimeter and the vital area boundaries are alarmed.

16. The CAS and SAS:

    a) have conventional (landline) telephone service and other communication capabilities with local law enforcement authorities, and

    b) are capable of continuous communications with security personnel.

As a result of its review of the ITAAC systems, hardware, and changes to regulations for "Power Reactor Security Requirements," effective May 26, 2009, the staff determined that the applicant submitted ITAAC within the AP1000 DCD that are not within the scope of the DC and that should be submitted as part of a COL application. In order to complete its review, in RAI-SRP14.3.12-NSIR-07, the staff requested that the applicant submit revised AP1000 ITAAC that address the 10 CFR Part 73 Power Reactor Security Requirements Final Rule.

In its original and Revision 1 responses to RAI-SRP14.3.12-NSIR-07, the applicant revised ITAAC Item 15 to include the requirement for emergency exits through the vital area boundaries to be equipped with a crash bar to allow for emergency egress.

The applicant also removed reference to emergency egress through the PA perimeter in ITAAC Item 15. As the applicant considers the PA barrier outside the design scope, it will be the responsibility of the COL applicant to complete the requirement in 10 CFR 73.55(e)(8)(iii) for emergency exits through the PA perimeter.

The applicant revised ITAAC Item 16(a) to include communication with the main control room, removed the phrase "other communication capabilities," and added ITAAC Item 16(c) to include the requirement for continued operability of non-portable communications equipment in the CAS and SAS in the event of the loss of normal power, by independent power sources.

On the basis of its review, the staff finds the revised ITAAC for systems, hardware, and features to be acceptable, because it is in conformance with the staff's definition of physical security hardware ITAAC that is within the scope of the DC, and sufficient to verify that the hardware, as finally installed and constructed, will function as designed.

The staff identified in its review of the proposed changes to the physical security hardware ITAAC in AP1000 DCD Tier 1 and Tier 2 as described below:

1. The external walls, doors, ceiling, and floors in the main control room, the CAS, and the SAS are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.

3. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

4. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the CAS and SAS upon intrusion into a vital area.

5.
   a) Security alarm annunciation and video assessment information is displayed concurrently in the CAS and the SAS, and the video image recording with real time playback capability can provide assessment of activities before and after each alarm annunciation within the perimeter barrier.

   b) The CAS and SAS are located inside the PA, and the interior of each alarm station is not visible from the perimeter of the PA.

   c) The CAS and SAS are designed and equipped such that, in the event of a single act, in accordance with the DBT of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to detect and assess alarms and communicate with onsite and offsite response personnel.

6. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.

7.
   a) Vital equipment is located only within a vital area.
   b) Access to vital equipment requires passage through the vital area barrier.

8. Isolation zones and exterior areas within the PA area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.

9. Emergency exits through the vital area boundaries are locked, alarmed and equipped with a crash bar to allow for emergency egress.

13.

a) The CAS and SAS have conventional (landline) telephone service with the main control room and local law enforcement authorities.

b) The CAS and SAS are capable of continuous communication with security personnel.

c) Non-portable communication equipment in the CAS and SAS remains operable from an independent power source in the event of loss of normal power.

15.

a) Security alarm devices including transmission lines to annunciators are tamper-indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms and emergency exit alarm) and location.

b) Intrusion detection and assessment systems concurrently provide visual displays and audible annunciation of alarms in the CAS and SAS.

16. Equipment exists to record onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.

In DCD Tier 1, Revision 19, Table 2.6.9-1, Table 3.3-6, and Tier 2, Section 13.6, "Security," the applicant made appropriate changes that reflect the changes identified in RAI-SRP14.3.12-NSIR-07, Revision 1.

The staff concludes that the applicant has adequately described the Tier 1 physical security hardware ITAAC to be incorporated as part of the standard design. The applicant adequately described the plant layout and protection of vital equipment in accordance with the requirements of 10 CFR 73.55 and provided the technical bases for establishing a physical protection system for the protection against acts of radiological sabotage. The applicant has adequately described requirements specific to design for alarm annunciation records in accordance with 10 CFR 73.70(f). The applicant has provided adequate descriptions of objectives, prerequisites, test methods, data required, and acceptance criteria for security-related ITAAC for the certification the AP1000 design. Therefore, the staff concludes that the AP1000 ITAAC within the scope of NUREG-0800 Section 14.3.12 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, analyses are performed and the acceptance criteria met, a facility referencing the certified AP1000 design has been constructed and will be operated in compliance with the DC and applicable regulations.

ITAAC Combined License Information Items

The staff reviewed the applicant's descriptions and commitments for COL action items for physical security hardware ITAAC that must be addressed by a COL applicant referencing the certified design.

In response to discussions with the staff at an RAI closed meeting on November 24, 2009, the applicant, in its Revision 1 response to RAI-SRP14.3.12-NSIR-07, proposed to include a COL action item in the next revision of the AP1000 DCD, Tier 2, Section 13.6.1, for the COL applicant to address the physical security hardware ITAAC listed above. The staff finds this to be acceptable and in conformance with the physical security hardware ITAAC requirements to protect against the DBT of radiological sabotage as stated in 10 CFR 73.1(a).

Based on its review of AP1000 DCD Tier 2, Section 13.6.2, the staff determined that the applicant has submitted an appropriate COL action item for physical security hardware ITAAC. The ITAAC that the applicant identified as the responsibility of the COL applicant meet the requirements of 10 CFR 52.47(b)(1) for a COL holder referencing the AP1000 design to build and operate in accordance with the DC, the provisions in the Atomic Energy Act and NRC regulations.

13.6.4 Conclusion

The staff finds that the applicant considered and provided descriptions of physical security systems or features in the standard AP1000 design to provide or facilitate the implementation of a physical protection system to protect against acts of radiological sabotage. The details of this information are provided in the SGI SER for the AP1000, which is stored in the automated database of the NRC's Secure Local Area Network Electronic Safe. The applicant adequately described the plant layout for physical protection and identifying vital equipment and areas, in accordance with the requirements of 10 CFR 73.55. The staff evaluated the technical bases and assumptions related to ITAAC for physical security hardware and found them to be adequate.

The applicant identified the generic issues in the following documents as being outside the scope of the AP1000 design: Generic letter (GL) 89-007, "Power Reactors Safeguards Contingency Planning for Surface Vehicle Bombs," dated April 28, 1989; GL 91-010, "Explosives Searches at Protected Area Portals," dated August 27, 1991; and GL 91-003, "Reporting of Safeguards Events," dated March 6, 1991. The staff has identified the generic issues in the above documents as outside the scope of the design and finds the applicant's approach acceptable. The staff's review of the design found that the applicant addressed Task Action Plan Item A-29, "Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage," in an acceptable method.

The staff finds that the applicant has provided reasonable assurance that the standard AP1000 design, if implemented correctly, will ensure adequate protection against acts of radiological sabotage. The applicant has provided sufficient physical security design information to support the issuance of an amendment to the AP1000 DC.

The staff reviewed AP1000 DCD, Tier 1, Section 2.6.9 and Tier 2, Section 13.6 and the applicant's responses to RAIs issued on Tier 1 and Tier 2 material, performed in accordance with NUREG-0800 Section 13.6.2, and NUREG-0800 Section 14.3.12 Revision 1 of January 2010. Based on its review, the staff determined that the applicant's selection criteria

and methodology for the development of Tier 1 information are acceptable, that the implementation of this selection criteria and methodology is appropriate, and that the resultant ITAAC are adequate for verification that a facility referencing the AP1000 design has been constructed and will be operated in compliance with the DC and applicable regulations.

The staff finds that the applicant has adequately described the objectives, prerequisites, test methods, data required, and acceptance criteria for physical security hardware ITAAC for the certification of the AP1000 design.