23 – DESIGN CHANGES PROPOSED IN ACCORDANCE WITH ISG-11

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

[ ]

The complete text of this chapter, containing proprietary information can be found at ADAMS Accession Number ML102920584 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

23 - Introduction

This chapter addresses new design changes proposed by Westinghouse that are scheduled for inclusion in Revision 18 of the AP1000 design control document (DCD). The design changes that are evaluated in this chapter do not constitute all of the changes that Westinghouse proposes to include in the DCD, Revision 18. Rather, the design changes evaluated in this chapter are in addition to those that Westinghouse has submitted to the U.S. Nuclear Regulatory Commission (NRC) as responses to requests for additional information or safety evaluation report open items.

Westinghouse proposes that the DCD be changed by adding new, more detailed design information that expands upon the design information already included in the DCD. This information would be used by every combined license application that references the certified AP1000 design. The regulations of 10 CFR 52.63(a)(1) require that the Commission may not modify a design certification, whether on its own motion or in response to a petition from any person, unless the Commission determines that the change meets one of seven criteria. The staff finds that each of the changes evaluated in this chapter meets one or more of the seven criteria in 10 CFR 52.63(a)(1). Further, the staff finds that many of the design changes evaluated in this chapter contribute to the increased standardization of the certification information because the design amendment would be applied to all COL applicants that reference the design certification rule. Therefore, these changes enhance standardization, and meet the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

Interim Staff Guidance, DC/COL-ISG-011, "Interim Staff Guidance Finalizing Licensing-basis Information," describes the NRC staff position regarding the control of licensing-basis information during and following the initial review of applications for design certifications. In part, ISG-11 describes the categories of design changes that applicants should not defer until after the issuance of the design certification rule. Categories of those changes that should not be deferred include:

- the correction of significant errors in an application,
- changes needed to ensure compliance with NRC regulations,
- changes needed to support other licensing-basis documents (e.g., conforming changes to information in the FSAR supporting technical specifications),
- significant technical corrections associated with the design or program described in the licensing document (i.e., if not changed, would preclude operation within the bounds of the licensing basis, as opposed to proposed alternatives to the described design or program), and
- changes needed to address a significant vulnerability identified by probabilistic risk assessments (PRAs) or other studies (e.g., a change in a PRA insight).

This chapter includes 37 confirmatory items, which will be resolved upon NRC staff confirmation that Westinghouse has properly included these proposed design changes in the DCD.

23.A Changes to Component Cooling Water System

23.A.1 Description of Proposed Changes

In letters dated April 26, 2010 (ML101180111), and July 29, 2010, (ML102150199), Westinghouse proposed changes to the design of the component cooling water system. The normal residual heat removal system (RNS), component cooling system (CCS) relief valves increase in size from 1 inch x 1 inch to 3 inches x 4 inches to meet required flow capacity requirements. In addition, the CCS surge tank vent-line increases in size from 2 inches to 3 inches nominal pipe size (NPS). Tier 2 text in the DCD is modified to include Sections 3.4.1.2.2.2, “Auxiliary Building Flooding Events,” 9.2.2.3.4, “Component Cooling Water System Valves”, 9.2.2.4.5.2, “Leakage into the Component Cooling Water System from a High Pressure Source,” 19E.2.5, “Component Cooling and Service Water Systems,” and Figure 9.2.2-2, “Component Cooling Water System Piping and Instrumentation Diagram”.

23.A.2 Regulatory Basis

The regulatory basis for evaluating the component cooling water system (CCS) is documented in Section 9.2.2 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.” While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the reactor coolant system is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to regulatory treatment of nonsafety systems (RTNSS) in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of inspections, tests, analyses and acceptance criteria (ITAAC), test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Section 9.2.2, "Reactor Auxiliary Cooling Water System," Revision 4, March 2007, as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified by SRP Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.A.3 Technical Evaluation

During the staff's evaluation of the proposed CCS design changes, it was determined that additional information was required from the applicant. The staff generated RAI DCP-CN-06-SBP-01 addressing three issues.

1. Information previously supplied in a letter dated January 20, 2010, for Section 15.6.2, "Failure of Small Lines Carrying Primary Coolant Outside Containment," was omitted from the April 26, 2010 submittal. This information appeared to be relevant but was removed without an explanation. Additional information is needed to justify that a Chapter 15 change is no longer needed.
2. The basis for the increased CCS surge tank vent line from 2" to 3" (overflow protection due to normal residual heat removal system leakage into CCS) was not described in the DCD, Section 9.2.2. The applicant should consider adding this information to the DCD.
3. The flow rate (GPM) of the CCS/RNS relief valve(s) through the WWS was not described as related to the capacity of the auxiliary building sump pumps (concerns for potential building flooding if the relief valve flow rate exceeds the sump pump flow rate). The applicant should consider adding this information to the DCD.

The applicant's response to RAI DCP-CN-06-SBP-01 provided the following:

1. The information was deleted from this section in the final proposed change because it describes a non-limiting case (i.e., it is bounded by the sample line break). Discussion of the RNS heat exchanger tube leak has been incorporated into Appendix 19E, Section 19E.2.5.
2. The CCS surge tank line vent / overflow line was increased in size from 2" to 3" to eliminate the potential for over-pressurizing the surge tank (designed as an atmospheric tank) in the event of a large RNS heat exchanger tube leak that causes a significant increase in liquid volume in the CCS. The basis for the increase in surge tank vent line size is described in the proposed revision to DCD Section 9.2.2.3.3.

3. The maximum flow rate possible as a result of a double-ended break of one RNS heat exchanger tube is approximately 520 gpm. The large relief valve on the RNS heat exchanger cooling water line discharges to the radioactive waste drain system (WRS) auxiliary building equipment and floor drain sump at elevation 66'-6". This sump is pumped to the waste holdup tank by two air-driven sump pumps, each of which has a nominal capacity of 125 gpm. In the event that the relief valve discharges continuously for an extended period of time, the WRS floor drain sump may overflow into the 66'-6" level of the auxiliary building. Section 3.4.1.2.2.2 in Tier 2 of the DCD describes auxiliary building flooding events. In the radiologically controlled area of the first level (elevation 66'-6") there are no safe shutdown components and the maximum flood elevation has been determined to be 12 inches or less, assuming that the flooding is identified and isolated within 30 minutes.

Based on the staff's review, the applicant's response for item 1 was determined to be acceptable since the Tier 2 DCD text that was deleted for Section 15.6.2 (between the January and the April letters) was not a bounding case and did not need to be included as part of DCD Section 15.6.2. The information that was added to Section 19E.2.5 was evaluated by the staff and was found acceptable since the DCD markup added relevant information associated with the RNS heat exchanger and overpressure protection. Inclusion of the DCD markups in a future DCD revision is being tracked as **Confirmatory Item 23-01.**

For Item 2, the staff finds the applicant's response to be acceptable since the increase in relief valve size assures that the CCS surge tank does not over-pressurize due to an RNS heat exchanger tube leak. The existing 2" NPS was too small to pass the approximate 520 GPM leakage from the RNS tube rupture. DCD markup was provided to add this RNS leak-rate value to Tier 2 DCD Section 9.2.2.3.3. Inclusion of the DCD markups in a future DCD revision is being tracked as **Confirmatory Item 23-02.**

For Item 3, the staff finds the applicant's response to be acceptable since an overflow of the WRS floor drain sump does not affect safe shutdown components of a RNS heat exchanger tube break. The maximum flooding level on this floor (auxiliary building, elevation 66'- 6") was determined to be 12 inches or less above the floor, assuming flooding is identified and isolated within 30 minutes.

23.A.4 Conclusion

Pending resolution of the **Confirmatory Item 23-01** and **Confirmatory Item 23-02** noted above, the staff's review concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related structures, systems, and components (SSCs) and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. These design changes were evaluated with respect to the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable. RAI-DCP-CN-06-SBP-01 is resolved.

23.B Changes to Component Cooling Water System

23.B.1 Description of Proposed Change

In letters dated April 26, 2010 (ML101180111), and July 29, 2010, (ML102150199), Westinghouse proposed changes to the design of the component cooling water system. Four piping header systems were added for the RNS/CCS relief valves. Tier 2, DCD Figure 9.2.2-2, was modified to include these piping header systems.

23.B.2 Regulatory Basis

The regulatory basis for evaluating the component cooling water system (CCS) is documented in Section 9.2.2 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the reactor coolant system is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to regulatory treatment of nonsafety systems (RTNSS) in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of inspections, tests, analyses and acceptance criteria (ITAAC), test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Section 9.2.2, "Reactor Auxiliary Cooling Water System – Revision 4, March 2007," as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified by SRP Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.B.3 Technical Evaluation

During the staff's evaluation of the proposed CCS design changes, , it was determined that additional clarification to the description of the design changes was required from the applicant; therefore, the staff generated RAI DCP-CN-09-SBP-01.

The applicant's response to RAI DCP-CN-09-SBP-01 is as follows:

> The "Eliminate the RNS …" statement should be replaced with "Larger pressure relief valves have been added to the CCW system." The reason for this design change is further clarified by:

The original (1” x 1”) thermal relief valves provided for the RNS and SFS heat exchangers were intended to discharge to the floor of the CCS valve room, where the discharge would be collected by a nearby floor drain. Now that much higher capacity relief valves are needed for the RNS heat exchanger cooling water lines to meet ASME VIII equipment overpressure protection requirements, the larger valves needed for the RNS heat exchanger (V302A/B) are piped to a dedicated drain collection funnel located in the valve room to minimize the potential release of large quantities of vapor and water spray to the room in the event that a high capacity discharge occurred. The smaller valves (V342A/B) are also provided with their own collection header that discharges near the existing room floor drain. These changes are shown in the revised sheet 3 of Tier 2 DCD Figure 9.2.2-2.

Based on the staff’s review, the applicant’s response was determined to be acceptable since the design includes a collection piping header and dedicated drain funnel to handle the potential release of large quantities of vapor and water spray to the room in the event of a high capacity relief valve discharge. This discharge connects to the main drain header in the auxiliary building. The previous design allowed relief valve discharge to the floor with water collection to the nearest floor drain. DCD markup was provided to add this design improvement to Tier 2 DCD Figure 9.2.2-2. Inclusion of the DCD markups in a future DCD revision is being tracked as **Confirmatory Item 23-03.**

23.B.4 Conclusion

Pending resolution of **Confirmatory Item 23-03** noted above, the staff’s review concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related structures, systems, and components (SSCs) and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. These design changes were evaluated with respect to the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable. RAI-DCP-CN-09-SBP-01 is resolved.

23.C Changes to Component Cooling Water System

23.C.1 Description of Proposed Changes

In a letter dated June 18, 2010 (ML101720647), Westinghouse proposed changes to the design of the component cooling water system. The proposed changes increase the size of the relief valves on four (4) cooling water lines associated with the reactor coolant pumps (RCPs), and one (1) chemical and volume control system (CVS). Specifically, the RCP cooling water line relief valves V253/A/B/C/D are changed from 3" x 4" to 4" x 6" with the associated branch lines L253A/B/C/D changed from 3" to 4". Also the CVS letdown cooling water line relief valve V222 is changed from 2" x 3" to 3" x 4" with the associated branch line size of L222 changed from 2" to 3". The basis for the change is to prevent over-pressurization of the CCS piping systems. Tier 2, DCD Figure 9.2.2-2 was modified to include these changes.

23.C.2 Regulatory Basis

The regulatory basis for evaluating the component cooling water system (CCS) is documented in Section 9.2.2 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the reactor coolant system is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to regulatory treatment of nonsafety systems (RTNSS) in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of inspections, tests, analyses and acceptance criteria (ITAAC), test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Section 9.2.2, "Reactor Auxiliary Cooling Water System," Revision 4, March 2007, as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified by SRP Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.C.3 Technical Evaluation

Based on the staff's review, these design changes were determined to be acceptable. The changes to the five relief valves and associated branch piping are necessary to prevent over-pressurization of the CCS due to a postulated tube leak in the RCP and CVS heat exchangers with the cooling water lines isolated. The larger relief valves, 4" x 6" for the RCPs and 3" x 4" for CVS, and associated piping have been adequately sized for the required relief flow, thus preventing possible damage to the CCS. DCD markup was provided to address these changes to Tier 2 DCD Figure 9.2.2-2. Inclusion of the DCD markups in a future DCD revision is being tracked as **Confirmatory Item 23-04.**

23.C.4 Conclusion

Pending resolution of **Confirmatory Item 23-04** noted above, the staff's review concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related structures, systems, and components (SSCs) and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. These design changes were evaluated with respect to the adequacy of

ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable.

23.D Changes to Ancillary Diesel Generator System

23.D.1 Description of Proposed Changes

In letters dated May 10, 2010 (ML101380275), July 29, 2010 (ML102150199), and August 5, 2010 (ML102210127), Westinghouse proposed changes to the design of the ancillary diesel generator system. These design change apply to the nonsafety-related ancillary diesel generators and include the following changes:

- Increase the rating (from 35 KW to 80kW) and physical size of the ancillary diesel generators
- Increase the size of the diesel fuel oil storage tank and change principal construction code to ASME Section VIII
- Revise certain circuit breaker sizes
- Revise corresponding physical drawings and electrical one line diagrams
- Revise design of HVAC system for ancillary diesel generator and oil storage tank rooms

23.D.2 Regulatory Basis

Sections 8.3.1.3 and 9.5.4 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," address the function and acceptability of the ancillary diesel generators for the AP1000. No regulatory basis documents are identified as applicable to the ancillary diesel generators since they are nonsafety-related, commercial grade equipment. Acceptability of these design changes were based on conformance with the existing AP1000 licensing basis.

23.D.3 Technical Evaluation

The ancillary diesel generators are designed to provide the post-72 hour power requirements following an extended loss of onsite power sources. The ancillary diesel generators are not safety-related but provide electric power for Class 1E post-accident monitoring, main control room (MCR) lighting, MCR and division B and C instrumentation and controls room ventilation and for refilling the PCS water storage tank and the spent fuel pool when no other sources of power are available. The ancillary diesel generators are not required to perform these functions for the first 72 hours following a loss of all other ac power sources. As stated in the proposed design changes submittal, the design of the anchorage for these skid-mounted package units is consistent with the SSE design of equipment anchorages of seismic Category II equipment. The off-skid fuel oil piping and fuel oil storage tank are analyzed to show that they withstand an SSE.

During the staff's evaluation of the proposed design changes to the ancillary diesel generator design, it was determined that additional information was required from the applicant to complete the staff's evaluation. The staff generated RAI DCP-CN-59-SBP-01 requesting clarification of the design changes and justification for the changes to the ventilation system for cooling the diesel. In addition, the description of the design changes did not appear to be consistent with the changes made to the DCD description for this system.

The heat removal arrangement for the operating diesels was changed to include ducting from the discharge of the diesel radiators through a backdraft damper to a hurricane-proof louver on an exterior wall of the diesel enclosure building. There is also a damper and recirculation bypass in the cooling exhaust duct that can be closed in cold weather to recirculate engine heat to the diesel generator room for room heating.

The inlet cooling air is admitted to the diesel generator room by opening a personnel access door on an exterior wall of the enclosure building when the diesel generator is operating. The staff questioned the inconsistency between an exhaust protected against hurricanes and an intake that is not protected during operation of the diesel generator.

The applicant response to RAI DCP-CN-59-SBP-01 provides sufficient clarification of the design change to support the staff's evaluation and also provides proposed additional changes to the DCD to document the clarifications. With respect to the design for hurricane conditions, the applicant stated that a hurricane is only considered to be an initiating event and is postulated to occur when the ancillary diesel generators are not in operation. When the ancillary diesel generators are not operating, the exterior door used to provide intake of engine cooling air will be closed and this door is designed to withstand a hurricane, including windborne missiles. Since no more than a single initiating event hurricane is postulated for the design basis, hurricane protection is not needed when the ancillary diesel generators are operating and the door that provides inlet cooling air is open.

The change in design code for the ancillary diesel fuel oil storage tank from UL to ASME Section VIII provides a more rigorous and extensive set of design, fabrication and testing requirements for the tanks, and is therefore acceptable to the staff.

The proposed changes to the ancillary diesel generator do not change the ancillary diesel generators' physical or functional relationship to safety-related structures, systems and components (SSCs) and, therefore, do not increase the potential to adversely affect these SSCs. In addition, the proposed DCD changes include a statement that the ancillary diesel generators and associated SSCs are designed to preclude spatial interaction with any other non-seismic SSC that could adversely interact to prevent the functioning of the post-72 hour SSCs following a safe shutdown earthquake. The proposed DCD changes as revised in the response to RAI DCP-CN-59-SBP-01 are acceptable. Inclusion of the DCD markups in a future DCD revision is being tracked as **Confirmatory Item 23-05**.

The applicant proposed to change the size of the ancillary generator from 35 kW to 80 kW because the starting motor current of the passive containment cooling system (PCS) recirculation pump, which can draw up to 6.5 times the full load current, was not considered in the generator sizing calculations. In addition, as part of this modification, the applicant proposed to revise the input circuit breaker size for the regulating transformer from 20 amps to 125 amps, the load test breaker from 100 amps to 60 amps, the PCS pump motor from 20 amps to 50 amps, revise Figure 8.3.1-3 of DCD Tier 2, and update Class 1E 208/120 UPS one line diagram (Figure 8.3.2-2) to change "Transportable" AC Generator to "Ancillary" AC Generator. In addition, the applicant proposed other associated physical arrangement drawing changes to accommodate the increased size of the ancillary generator.

The NRC staff reviewed the information provided by the applicant and was concerned that the revised rating of the distribution panel and the size of the output breaker for load testing of ancillary generator were not compatible to accommodate the higher output current from the upgraded ancillary generator. In addition, the staff found that the proposed change to revise the input circuit breaker size for the regulating transformer from 20 amps to 125 amps did not seem to be adequate. In RAI SRP 8.3.1-EEB-2, the NRC staff requested the applicant to provide its basis for the following:

1) Utilizing a 125 amp breaker to protect 45 kVA regulating transformer.
2) Adequacy of the distribution panel rating of 100 amps for the 80 kW ancillary AC generators.
3) Adequacy of the 60 amps breaker for full load testing of the 80 kW ancillary AC generators.

In a letter dated August 5, 2010, the applicant stated that it had reviewed the adequacy and accuracy of both the rating of the distribution panel and size of each of the breakers on the ancillary diesel generator bus. Based on its review, the applicant revised the distribution panel rating and breaker sizes as follows:

1) The 125 amp breaker to protect the 45 kVA regulating transformer will be changed from 125 amp breaker to 20 amp breaker, as was originally designed, to a breaker size appropriate to the load on the transformer under ancillary diesel generator operating conditions of 20 amps.
2) The distribution panel rating will be increased from 100 amps to 225 amps so that it will be protected by the generator output breaker. This generator output breaker will be changed from 100 amps to 150 amps so as to be greater than 125% of the generator output.
3) The 60 amp breaker for full load testing of the 80 kW ancillary generators will be replaced with a 150 amp breaker to allow for full testing of the ancillary diesel generator.

4) The 50 amp breaker to protect PCS motor will be changed to 100 amps in order to avoid spurious tripping of the breaker on PCS motor start.

The staff has reviewed the above information and concludes that the proposed revised rating of the distribution panel and the size of each of the breakers on the ancillary diesel generator bus are compatible with the revised rating of the ancillary diesel generator. Therefore, the staff finds this concern resolved.

23.D.4 Conclusion

Pending resolution of **Confirmatory Item 23-05** noted above, the staff's review concludes that the proposed design changes above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the ancillary diesel generators to perform their post-72 hour function. These design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable. RAI-DCP-CN-59-SBP-01 and RAI SRP 8.3.1-EEB-2 are resolved.

The NRC staff has reviewed the proposed changes to the ancillary diesel generator design and concludes that the ancillary diesel generator distribution panel and the associated load feeder breakers are sized in accordance with the National Electric Code and, therefore, the proposed changes are acceptable. Also, the staff finds the changes made in Figure 8.3.2-2 to be minor and acceptable.

23.E Changes to Potable Water System

23.E.1 Description of Proposed Changes

In letters dated April 26, 2010 (ML101180111), and August 2, 2010 (ML102170032), Westinghouse proposed changes to modify the design of the potable water system (PWS) to add a safety-related loop seal in the PWS piping that penetrates the main control room (MCR) envelope boundary to prevent in-leakage into the main control room envelope during main control room emergency habitability system (VES) operation.

23.E.2 Regulatory Basis

The applicable regulatory requirement and regulatory guidance are as follows:

- General Design Criterion (GDC) 2, "Design Basis for Protection against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake.

- Regulatory Guide (RG) 1.29, "Seismic Design Classification."

- RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors.”

23.E.3 Technical Evaluation

The current PWS is a nonsafety system. Westinghouse proposed a design modification of the PWS for the consideration of control room envelope integrity under a seismic event. If a pipe break occurs resulting from a seismic event, the current design of the PWS would not maintain the MCR VES pressure boundary. In the proposed design modification, Westinghouse added an isolation valve, a loop seal, and a vacuum breaker to upgrade portions of the PWS to provide safety-related function for MCR isolation in meeting RG 1.197 for MCR envelope integrity.

To meet the requirements of GDC 2 as it relates to structures and systems being capable of withstanding the effects of natural phenomena including seismic event, acceptance depends on meeting the guidance of the portions of Regulatory Position C.1 of RG 1.29 for the safety-related portions of the system and Regulatory Position C.2 of RG 1.29 for the nonsafety-related portions of the system.

To determine the adequacy of the safety-related function of the system modification, the staff requested more detailed information in RAI-DCP-CN01-SBP-COLP-01. In the RAI, the staff requested the applicant to provide additional information pertaining to (1) a figure and system description for the illustration of the system modification in the DCD, (2) the design basis and maintenance requirements for the loop seal, (3) the DCD changes including both Tier 1 and Tier 2, (4) the single failure discussion for the vacuum breaker, (5) the minimum response time for operator manual actions, (6) flood protection, (7) testing requirements, (8) Technical Specifications requirements, and (9) MCR envelope integrity.

In a letter dated August 2, 2010 (ML102170032), responding to RAI-DCP-CN01-SBP-COLP-01, the applicant provided more detailed information. Based on the review of the additional information, the staff found the following discussion from the RAI responses to form the basis for its finding:

1. Figure 9.2.5-1, “Main Control Room Potable Water System Isolation,” and system description are added in the DCD Tier 1 and Tier 2 to illustrate the portions of the PWS that have safety-related function pertaining to preventing in-leakage into the main control room envelope.

2. The loop seal is designed to maintain MCR isolation providing the operator sufficient time to close the PWS isolation valves in the event that the non-seismic PWS piping broke. The potable water tank is located at a higher elevation that maintains water in the loop seal by design. Therefore, no maintenance requirements are needed for the loop seal.

3. DCD changes were made in Tier 1 Section 2.7.1, Table 2.7.1-1and Table 2.7.1-2, to include the safety-related portions of the PSW. These two tables are referenced by

ITAAC Table 2.7.1-4. In Tier 2, changes were made in Table 3.2-3, Table 3.9-12, Table 3.9-16, Table 3.11-1, Section 9.2.5.1.1, Section 9.2.5.4, Section 14.2.9.1.6, which address the safety design basis, safety-related portions of the PWS in AP1000 Class C, seismic Category I, ASME III-3, inservice test requirements, and environmental qualification.

4. The vacuum breaker is to help ensure that a break in the nonsafety-related potable water supply piping would not cause water to siphon from the loop seal. These pressure/vacuum relief devices are not required to consider single active failures. This is consistent with the implementation of a single vacuum relief device in the automatic depressurization system and VES.

5. The safety-related portions of the potable water seal assure main control room pressure boundary integrity after a design basis event. Since a seismic event is not assumed in the analysis to occur simultaneously with another design basis event (such as LOCA), there are no radiological conditions that would change MCR habitability. VES actuation occurs only when there is sustained loss of electrical power. For this situation, the design basis for VES actuation is to provide breathable air for the MCR occupants. The MCR occupants will have a supply of breathable air for 72 hours and the MCR would remain habitable. The operators would be alerted to a loss of air through the loop seal piping by the low differential pressure alarm for the MCR and would close the potable water manual isolation valves. The time for the operator actions is not a critical parameter for the safety-related function.

6. The non-seismic PWS lines in the MCR are limited to the kitchen and restroom areas with line sizes of one inch (2.54 cm) and smaller. There are no safety-related components that would be adversely affected by a rupture of the nonseismic PWS piping in the MCR kitchen and bathroom areas.

7. The initial test program and inservice testing included the PWS to ensure the integrity of the MCR pressure boundary; as shown in Tier 2 Section 14.2.9.1.6, Table 3.9.6, and Table 3.9-16.

8. Technical Specifications, Section 3.7.6, “Main Control Room, Habitability System,” has been updated to include the safety-related isolation valves of the PWS. The MCR pressure boundary includes the PWS water seal that prevents gas flow through the piping. These TS changes were addressed in the response to RAI-SRP-6.4-SPCV-03.

9. The sanitary drainage system (SDS) is one of the systems that penetrate the MCR boundary. Portions of the SDS including isolation valves and loop seal have been modified to safety-related and seismic Category I. The modification of the PWS, DCD Tier 1 and Tier 2 information pertaining to the SDS was revised accordingly. The integrity of the AP1000 MCR boundary was addressed in the response to RAI-SRP-6.4-SPCV-03.

The staff has determined that the proposed modification meets RG 1.29, Position C.2 for the nonsafety-related PSW based on the information in Items (1) and (6) above – that the change portions of the nonsafety-related PWS will be safety-related, and that a failure of the nonsafety-related portions of the system will not adversely affect any of the safety-related functions. Further, the DCD information discussed in Item (3) above demonstrates that RG 1.29 Position C.1 is met for the safety-related portions of the system because appropriate classification designations are specified for the PWS consistent with the approach described in Tier 2, DCD, Section 3.2. Based on meeting the guidance of RG 1.29 Positions C.1 and C.2, the staff has determined that the modified PWS meets GDC 2.

The staff reviewed the response to Items (2) and (4) and agrees with the applicant's justifications that (1) no maintenance is required for the loop seal because the elevation of the water tank is sufficient to provide the water seal without maintenance and that (2) the vacuum breaker is not required to consider single active failures because it is consistent with the VES design for the automatic pressure relieve devices.

The staff reviewed the response to Item (5) for the human factors concern of the minimum response time for operator manual actions. The applicant explained that operator action was not credited for any design basis event. The safety-related (seismic) design of the loop seal and surrounding pipe in coordination with other safety-related systems used to manage design basis events ensure that either the hazardous environment that would necessitate control room isolation is prevented and/or that the control room isolation function is maintained. The manual valves were installed as a defense-in-depth measure to address longer-term evaporation of the water in the loop seal, and for deterministic beyond-design-basis evaluations that simply considered the loop seal to be unavailable. There is no regulation or regulatory guidance applicable to operator manual actions used in this manner.

In reviewing Items (7) through (9), the staff finds that there is sufficient information to address the need for testing the isolation valves and establishing their Technical Specifications to ensure the integrity of the MCR pressure boundary. The review of the integrity of the MCR pressure boundary is under SRP Section 6.4. The proposed changes were found acceptable in SER Section 6.4.1.3 because the overall effectiveness of the control room envelope is demonstrated through the testing associated with the control room integrity program. Therefore, the proposed design modification along with the changes in the DCD for testing and TS is consistent with RG 1.197 as it relates to demonstrating control room envelope integrity.

Based on the above, the staff has determined that sufficient information is provided to address the staff's questions posed in RAI-DCP-CN01-SBP-COLP-01 and that sufficient details are provided in the DCD markups to address the safety-related function of the PWS. Inclusion of the DCD markups in a future DCD revision is being tracked as **Confirmatory Item 23-06.**

23.E.4 Conclusion

Pending resolution of **Confirmatory Item 23-06** noted above, the staff's review concludes that the design changes described above are acceptable because they meet the requirements and guidance of GDC 2, RG 1.197, and RG 1.29. RAI-DCP-CN01-SBP-COLP-01 is resolved.

23.F Changes to Reactor Coolant Pressure Boundary Leakage Detection

23.F.1 Description of Proposed Changes

In letters dated January 20, 2010 (ML100250888), and July 29, 2010 (ML102150199), Westinghouse proposed changes to revise the licensing basis for unidentified Reactor Coolant System (RCS) leak detection by removing the N13/F18 containment atmosphere radiation monitor and replacing it with the Flourine-18 (F-18) particulate monitor. The Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.4.9a, and the related discussions in the TS Bases B 3.4.7 and B 3.4.9 are revised to reflect the change from the "N13/F18 gaseous monitor" to an "F18 particulate monitor."

In the process of reviewing this proposed design change, the staff identified an operating issue pertaining to RCS leakage detection. Operating experiences at Davis Besse (NRC Bulletin 2002-01) indicated that prolonged low-level unidentified reactor coolant leakage inside containment could cause material degradation that could compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary.

23.F.2 Regulatory Basis

The applicable regulatory requirement and regulatory guidance are as follows:

- General Design Criterion (GDC) 30, as it relates to providing means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

- 10 CFR 52.79(a)37, as it relates to "information necessary to demonstrate how operating experience insights have been incorporated into the plant design."

- Regulatory Guide (RG) 1.45, Revision 1, as it relates to "Guidance on Monitoring and Responding to Reactor Coolant System Leakage."

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the containment atmosphere radioactivity monitoring system design information as described in DCD Sections 5.2.5 and 11.5. The proposed changes were evaluated using the guidance provided by Standard Review Plan

(SRP) Chapter 16, “Technical Specifications – Revision 3, March 2010.” Acceptability was based on conformance with the guidance specified by SRP Chapter 16.

23.F.3 Technical Evaluation

GDC 30 requires that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. RG 1.45 describes acceptable methods for implementing GDC 30.

F18 Particulate Monitor

The staff reviewed the proposed design changes, including changes in DCD Tier 2 Appendix1A, Sections 3.1.4, 3.6.3.3, 5.2.5.3.3, and 11.5.2.3.1, and Technical Specifications LCO 3.4.9, Sections B.3.4.7, and 3.4.9. The proposed design changes revise the licensing basis for unidentified RCS leak detection by removing the N13/F18 containment atmosphere radiation monitor and replacing it with an F-18 particulate monitor.

Detector Sensitivity

In the review, the staff determined that the applicant did not provide sufficient information to demonstrate that the newly proposed F-18 particulate radiation monitor (PSS-JE-RE027) sensitivity is capable of detecting the RCS leak rate of 0.5 gpm according to the Technical Specifications. Therefore, in RAI SRP-11.5-CHPB-05, the staff requested additional information on the analysis demonstrating the sensitivity of the proposed radiation monitor.

Although this design change specifies the radiation monitor sensitivity for particulate radioactivity, the change does not provide an analysis to demonstrate that the specified monitor sensitivity is capable of satisfying the technical basis of using realistic radioactive concentrations in the RCS, as described in RG 1.45, Revision 1.

In a May 14, 2010, response to RAI-SRP-11.5-CHPB-05, the applicant provided the analysis demonstrating that the monitor can detect a RCS leak rate [ ] lower than that specified in the AP1000 technical specifications. Based on its review, the staff found that the results of the applicant’s analysis were highly and directly dependent upon the F-18 concentration in the reactor coolant and the fraction of F-18 assumed to enter the containment atmosphere.

The applicant derived the coolant concentration from two references [ ] and to be conservative and to minimize the amount of radioactivity leaked from the RCS, the applicant used the lower value in its analysis. The staff’s own literature review of F-18 in PWR reactor coolant indicated that the concentration could be 0.014 uCi/ml, [ ] which is lower than that used by the applicant.

The applicant estimated the amount of F-18 entering the containment atmosphere to be [ ] a small fraction of the total leaked from the RCS. Westinghouse derived this fraction from an analysis that estimated the fraction flashed into the space between a RCS coolant pipe and the insulation surrounding the pipe, and the fraction escaping from the insulation into the containment atmosphere. The applicant estimated the fraction flashed into the space between the pipe and insulation using a method described in NUREG-1320,"Nuclear Fuel Cycle Facility Accident Analysis Handbook" dated May 1988. Using an aerosol transport code, Westinghouse then calculated the fraction escaping the insulation into the containment atmosphere. The staff independently verified the flash fraction calculation using the method described in NUREG-1320. The staff also concluded that the final escape fraction calculated by the aerosol transport model was conservative because almost all aerosols larger than 1 micron diameter fail to escape into the containment atmosphere, thus greatly reducing the amount of radioactivity reaching the monitor.

Based on its review and independent verification, the staff concludes that the proposed monitor is sufficiently sensitive to detect the technical specification leak rate. Hence, the staff closes RAI SRP-11.5-CHPB-05.

Response Time

In the letter dated March 12, 2010, the applicant proposed to change the radiation monitor response time for the leak detection from 0.5 gpm within one hour to 0.5 gpm within two hours. However, in a May 14, 2010 response to RAI-SRP-11.5-CHPB-05, the applicant stated that the radiation particulate monitor is capable of detecting a 0.5 gpm leak in one hour, and the applicant provided corresponding Tier 2 DCD changes. In a teleconference on July 29, 2010, the applicant clarified that the letter dated May 14, 2010, superseded the early letter regarding the response time. The staff has determined that the response time of detecting 0.5 gpm leakage in one hour as specified in the May 14, 2010, letter is consistent with the certified design of DCD Revision 15. Based on the clarification and the updated Tier 2 DCD changes, the staff has determined that no change in the response time for the radiation monitor is necessary, and therefore, detecting 0.5 gpm leakage in one hour is acceptable.

Based on the above, the staff has determined that the proposed F18 particulate radiation monitor sensitivity and response time pertaining to the RCS detection function are acceptable. Further, using a particulate radiation monitor as one of the leakage detection instruments is consistent with the guidance in RG 1.45, Revision 1. Therefore, the staff has determined that the proposed F-18 particulate radiation monitor is acceptable.

With respect to proposed changes to GTS 3.4.9 and their associated bases, the staff finds these changes, as modified as a result of the Westinghouse's response to RAI SRP11.05-CHPB-05, acceptable because they reflect the system design and operating information described in DCD Sections 5.2.5 and 11.5.

Davis Besse Operating Experience with RCS Leakage Detection

NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," described that the operating experience at Davis Besse in 2002 indicated that prolonged low-level unidentified reactor coolant leakage resulting from nozzle cracking of the control rod drive mechanism inside containment could cause material degradation that could compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary. The question was raised regarding licensees' practices for identifying and resolving degradation of the reactor coolant pressure boundary. Pursuant to 10 CFR 52.79(a) 37, relating to the requirement to provide "information necessary to demonstrate how operating experience insights have been incorporated into the plant design," in RAI-DCP-CN45-SBP-01, the applicant was requested to address this issue.

COL Information Item

In a letter dated July 29, 2010 (ML102150199), responding to RAI-DCP-CN45-SBP-01, the applicant revised DCD to add a new COL information item, COL 5.2-3, and new Section 5.2.6.3 to describe COL 5.2-3 as the following.

5.2.6.3 Response to Unidentified Reactor Coolant System Leakage Inside Containment

> The Combined License applicant will provide information to address prolonged low-level unidentified reactor coolant leakage inside containment which could cause material degradation such that it could potentially compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary. This issue could be addressed by operating procedures. The procedures should address operator actions in response to prolonged low level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the Technical Specification (TS) limits to provide operator sufficient time to take actions before the TS limit is reached. The procedures should address identifying, monitoring, trending, and repairing prolonged low-level leakage. The procedures should also define the alarm setpoints and demonstrate that the setpoints are sufficiently low to provide an early warning for operator actions prior to Technical Specification limits. In addition, the procedures should address converting the instrument output to a common leakage rate.

The staff reviewed the RAI-DCP-CN45-SBP-01 response and determined it to be acceptable because the description of the COL information item in DCD Section 5.2.6.3 markups is consistent with the guidance in RG 1.45, Revision 1, pertaining to managing the prolonged low-level reactor coolant system leakage. Therefore, GDC 30 is met based on the conformance to RG 1.45. Inclusion of the DCD markups in a future DCD revision is being tracked as **Confirmatory Item 23-07.**

23.F.4 Conclusion

Pending resolution of **Confirmatory Item 23-07** noted above, the staff's review concludes that the design changes described above are acceptable because they meet the requirements and guidance of GDC 30 and RG 1.45, Revision 1. In addition, this design change was evaluated with respect to the adequacy of technical specifications that have been established for the RCS leak detection instrumentation system and found acceptable. COL applications that incorporate by reference the DCD must address the new COL information item described in DCD Section 5.2.6.3. RAI SRP11.05-CHPB-05 and RAI-DCP-CN45-SBP- 01 are resolved.

23.G Changes to Spent Fuel Flood-up Valves Remote Position Indication

23.G.1 Description of Proposed Changes

In letters dated April 26, 2010 (ML101180111), and August 3, 2010 (ML102170031), Westinghouse proposed changes to the spent fuel flood-up valves remote position indication. AP1000 DCD Revision 17 requires several valves that connect to the refueling cavity to have their position status monitored during plant shutdowns to prevent draining of the refueling cavity and the spent fuel pool. Tier 1 Table 2.3.7-1 presents a list of these valves. These valves are also designed as Seismic Category 1 components and identified as such in Tier 1 of the DCD. Westinghouse has identified in these proposed design changes that the Spent Fuel Pool Cooling System (SFS) valves SFS-PL-V031 and SFS-PL-V033 are also required to have their position status monitored during plant shutdowns to prevent draining of the refueling cavity and the spent fuel pool, but were not identified as such in the DCD. This design change proposes to modify Tier 1 Table 2.3.7-1 to require that these valves have their position status displayed in the Main Control Room (MCR). In addition, a previous design change identified that Valve SFS-PL-V075 is required to be locked open during normal operation to provide a flow path during scenarios requiring containment flood-up. During refueling, V075 provides the same function of preventing the draining of the refueling cavity and the spent fuel pool as V031 and V033. This design change proposes to modify Tier 1 Table 2.3.7-1 to require valve V075 to have its position indicated in the MCR as well. This design change proposes to add two external Class 1E limit switches (open/closed) to the following isolation valves:

- Spent Fuel Pool Cooling System (SFS) refueling cavity drain to Steam Generator System (SGS) compartment isolation valve (SFS-PL-V031)
- SFS refueling cavity drain to compartment sump isolation valve (SFS-PL-V033)
- SFS containment floodup isolation valve (SFS-PL-V075)

23.G.2 Regulatory Basis

The regulatory basis for evaluating these proposed design changes are documented in Section 7.5 and Section 9.1.2 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." In particular, GDC 61 requires the fuel storage system to be designed for adequate safety under anticipated operating and accident conditions. The system must be designed with the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions. During refueling operations the reactor cavity is connected to the spent fuel pool (SFP), and failure of the valves and piping sections identified above could drain the water from the refueling cavity and from the SFP. Acceptability of the proposed design changes was based on conformance with the existing AP1000 licensing basis and the guidance specified by SRP Section 9.1.2 (as applicable).

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of General Design Criteria (GDC) 13 and 19 in 10 CFR Part 50, Appendix A, and should meet guidance in RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." The proposed changes were evaluated using the guidance provided in Standard Review Pln (SRP) Section 7.5, "Information Systems Important to Safety." Acceptability was also based on conformance with the existing AP1000 licensing basis and criteria specified in SRP Section 7.5 (as applicable).

23.G.3 Technical Evaluation

During the staff's evaluation of these proposed design changes, it was determined that additional information was required from the applicant. The staff identified that the markup of DCD pages affected by the proposed design changes contained changes that were not directly related to the justification provided in the design change submittal. In RAI-DCP-CN55-SBP-01.a and -01.b, the applicant was requested to provide justification for these additional changes. In RAI response letter dated August 3, 2010, the applicant stated that the changes identified by the staff in RAI-DCP-CN55-SBP-01.a were not part of this proposed design change, but were part of RAI-SRP6.4- SPCV-03, R2. The staff found the applicant's response acceptable because these changes were already evaluated by the staff in Section 6.4 of this SER. Therefore, RAI-DCP-CN55-SBP-01.a is considered closed.

In response to RAI-DCP-CN55-SBP-01.b, the applicant stated that the deletions in Tier 2 Table 3.9-16 identified by the staff in RAI-DCP-CN55-SBP-01.b were not deleted from the table, but were moved to the next page. Since this represents an editorial change, the staff finds the applicant's response acceptable and RAI-DCP-CN55-SBP-01.b is considered closed.

In RAI-DCP-CN55-SBP-01.c, the staff identified that the Tier 1 drawings depicting this system did not show all the valves mentioned in the design change proposal. Therefore, the applicant was requested to update the Tier 1 drawings impacted by this design change. In RAI response letter dated August 3, 2010, the applicant included a markup of Tier 1, Figure 2.3.7-1 which now included valve SFS-PL-V075. Therefore, the staff finds the applicant's response to be acceptable and RAI-DCP-CN55-SBP-01.c is considered closed. The incorporation of the updated Tier 1, Figure 2.3.7-1 (as presented in response to RAI-DCP-CN55-SBP-01.c) into DCD revision 18 is identified as **Confirmatory Item 23-08.**

DCD Tier 2 Section 9.1.3.3.5, “Spent Fuel Pool Cooling System Valves,” provides a description of the valve arrangement needed for refueling. It was not clear to the staff if this description was impacted by the proposed change. In RAI-DCP-CN55-SBP-01.d, the applicant was requested to confirm that the configuration description provided in Tier 2 Section 9.1.3.3.5 was still valid and had not been impacted by the proposed change. In RAI response letter dated August 3, 2010, the applicant stated that the valve configuration description provided in Tier 2 Section 9.1.3.3.5 is still valid and has not been impacted by the proposed change. The staff agrees with the applicant’s determination that Tier 2 Section 9.1.3.3.5 is still valid; therefore, RAI-DCP-CN55-SBP-01.d is considered closed.

The staff evaluated Tier 2 Figure 9.1-6 (Sheet 1 of 2) “Spent Fuel Pool Cooling System Piping and Instrumentation Diagram,” and identified additional possible refueling cavity drain paths. In RAI-DCP-CN55-SBP-01.e, the applicant was requested to confirm that all refueling cavity drain path isolation boundary piping and components were described in Tier 2 and included in the appropriate Tier 1 sections and tables. In the RAI response letter dated August 3, 2010, the applicant stated that all refueling cavity penetrations and associated isolation valves that are at elevations below the minimum safety level outlined in Tier 2, Chapter 16 Tech Spec 3.9.4 are designed as Seismic Category I, and have been identified in the appropriate DCD (Tier 1 and 2) Section. The applicant also clarified that all refueling cavity penetration lines and associated isolation valves that are not Seismic Category I are located at elevations that preclude the possibility of draining the refueling cavity below the minimum safe level for refueling operations. The staff finds that the applicant has properly identified all the possible refueling cavity drain paths in the appropriate DCD sections, and that these drain paths are designed as Seismic Category I components. Therefore, RAI-DCP-CN55-SBP-01.e is considered closed.

As documented in NUREG-1793, the staff reviewed and approved the SFS system in AP1000 DCD Tier 1 Section 2.3.7 and Tier 2 Section 9.1.3. The staff reviewed the applicant’s proposed design changes to the SFS system in the AP1000 standard design by using the review procedures described in SRP Section 7.5 and requirements of GDC 13 and 19 in 10 CFR Part 50, Appendix A. AP1000 DCD, Revision 17 Tier 1 Table 2.3.7-1 shows that the safety-related display in the MCR is required for the SFS refueling cavity drain to the SGS compartment isolation valve (SFS-PL-V031) and SFS refueling cavity drain to the compartment sump isolation valve (SFS-PL-V033). These two valves are also required to have their position status monitored during plant shutdowns to prevent draining of the spent fuel pool. However, in AP1000 DCD, Revision 17, these two valves were designed without limit switches. To achieve the above required safety-related display functions, two external Class 1E limit switches (open/closed) are provided for Valve (SFS-PL-V031) and Valve (SFS-PL-V033).

The SFS containment floodup isolation valve (SFS-PL-V075) was added to the spent fuel pool cooling system by a previous DCP. This valve is required to be locked open to provide a flow path during scenarios requiring containment flood-up. Valve SFS-PL-V075, when closed, also provides the function of preventing the refueling cavity from draining. This function requires the status indication displayed for this valve in the MCR. But, the current valve functional requirements do not provide for remote position indication. Therefore, two external Class 1E limit switches (open/closed) have to be provided for the SFS containment floodup isolation valve to achieve the required safety-related display function.

For the specific DCD changes, the applicant added valve SFS-PL-V075 to DCD Tier 1 ITAAC Table 2.3.7-1 and the new SFS containment flood-up line to DCD Tier 1 ITAAC Table 2.3.7-2. The applicant also added the status of the above three valves to DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements," Table 3.11-1, "Environmentally Qualified Electrical and Mechanical Equipment," Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment," Table 7.5-1, "Post Accident Monitoring System"; and Table 7.5-7, "Summary of Type D Variables." The staff finds that the proposed design changes described above are acceptable. The revisions to the above DCD Tier 1 and Tier 2 tables are being tracked as **Confirmatory Item 23-09**.

23.G.4 Conclusion

The staff evaluated these proposed design changes against conformance with the existing AP1000 licensing basis and the guidance specified by SRP Sections 7.5 and 9.1.2 (as applicable). The staff also evaluated these proposed design changes against the requirements of GDC 61, which requires the fuel storage system to be designed for adequate safety under anticipated operating and accident conditions.

Pending resolution of **Confirmatory Item 23-08** and **Confirmatory Item 23-09**, the staff concludes that, based on the description provided in these design changes and the RAI responses discussed above, these proposed design changes are in compliance with the requirements of GDC 61 and follow the guidance provided in SRP Sections 9.1.2 and 7.5; therefore the staff finds the proposed changes to be acceptable.

23.H Changes to the AP1000 Steam Generator Thermal-Hydraulic Data Report

23.H.1 Description of Proposed Changes

In letters dated May 10, 2010 (ML101380275); August 4, 2010 (ML102210126); and August 12, 2010 (ML102290041); Westinghouse proposed the following changes to the AP1000 steam generator (SG) design:

- In DCD subsection 5.4.4.3, the pressure drop through the SG flow restrictor at 100 percent steam flow is changed from approximately 15 to approximately 20 psig.
- In DCD Table 5.4-5, the SG design fouling factor is changed from 1.1x10-4 to 9.0x10-5 hr-F-ft2/BTU.
- In DCD Table 10.3.2-2 (and technical specifications (TS) Table 3.7.1-2), the Main Steam Safety Valve (MSSV) lifting settings and relieving capacities are changed as follows:

Valve Nos.		Lift Setting (Psig)		Relieving Capacity (lb/hr)	
		From	To	From	To
MSSV #1	V030A/B	1185	1185	1,310,000	1,320,000
MSSV #2	V031A/B	1196	1197	1,320,000	1,340,000
MSSV #3	V032A/B	1208	1209	1,346,000	1,350,000
MSSV #4	V033A/B	1219	1221	1,356,000	1,360,000
MSSV #5	V034A/B	1231	1232	1,368,000	1,370,000
MSSV #6	V035A/B	1242	1232	1,370,000	1,370,000

The proposed changes to the MSSV lift settings and relieving capacities are in response to an increased pressure drop calculation for the SG steam outlet nozzle, which impacts the MSSV inlet line losses.

23.H.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes to lift settings and relieving capacity is documented in Chapters 10 and 15 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." General Design Criteria (GDC) 10 and 15, respectively, in 10 CFR Part 50, Appendix A, require that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any condition of normal operation, including anticipated operational occurrences. The design basis events in DCD Chapter 15 were analyzed to ensure compliance with GDC 10 and 15.

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the MSSV design and operating information described in DCD Section 10.3. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in SRP Chapter 16.

23.H.3 Technical Evaluation

As discussed in Westinghouse's August 12, 2010, response to RAI-DCP-CN58-SRSB-02, Westinghouse revised the calculation of the pressure differential across the SG steam outlet nozzle flow restrictor in the thermal-hydraulic data report, which is a document typically provided by Westinghouse to the utility operating the steam generator. The revised calculation showed an increase of the pressure loss from approximately 15 psig to 20 psig. Therefore, Westinghouse proposed to revise DCD Section 5.4.4.3 by stating that the resultant pressure drop through the SG flow restrictor at 100 percent steam flow is approximately 20 psig, rather than approximately 15 psig in DCD Revision 17. Also, the SG design fouling factor is revised from $1.1x10^{-4}$ to $0.9x10^{-5}$ hr-F-ft^2/BTU. Westinghouse stated that this reduction is based on the operating experience of replacement steam generators with Alloy 690 tubing since 1989, and SG fouling factor is reduced to offset the increased pressure loss while still maintaining sufficient margin for the SG heat transfer performance. For the chapter 15 safety analysis, a higher fouling factor was used where reduced heat transfer to the SG is limiting. The proposed lower value of SG design fouling factor provides increased margin. Therefore, there is no effect on the safety analysis.

Because of the increased pressure loss through the SG flow restrictor, Westinghouse proposed to change the relief capacities and lift settings of the MSSVs in accordance with ASME Section III, subsection NC-7300. The setpoint of MSSV #6 (V035A/B), which is the highest MSSV setpoint, was lowered to account for the increased pressure losses and still maintain the required relieving capacities. In response to RAI-DCP-CN58-SRSB-01, Westinghouse stated that, for the purposes of the AP1000 safety analysis, the significant MSSV setpoints are MSSV#1 (V030A/B) and MSSV #6 (V035A/B). MSSV #1 represents the lowest safety valve setpoint and is used to determine if design transients will challenge the MSSVs. MSSV #6 is the highest valve setpoint and is used to determine the overall steam pressure relief capacity in the safety analysis. In this proposed change, the lift setpoint of MSSV #6 is reduced from 1242 psig to 1232 psig to account for the impact of the revised SG flow restrictor pressure loss, but the relieving capacity of MSSV remains unchanged. The reduced lift setting resulted in a marginal increase in valve capacity with respect to the safety analysis of the limiting overpressure turbine trip event. Also, the lift setting of MSSV #1 is not changed while its relieving capacity is increased slightly. The setpoints and relieving capacities of MSSV #2 through MSSV #5 are also slightly increased, but they do not affect the safety analyses. Therefore, the proposed slight changes to the MSSV setpoints and relieving capacities have minimal effect on the safety analyses of the overpressure events. For the limiting turbine trip event, the minimum departure from nucleate boiling (DNBR) remains well above the safety analysis DNBR limit, and the RCS pressure and the secondary side steam pressure remain within 110 percent of the respective design pressures.

With respect to proposed changes to TS 3.7.1, the staff finds these changes acceptable because they reflect the MSSV design and operating information described in DCD Section 10.3. However, verification that these changes are correctly incorporated in a future revision of the DCD is **Confirmatory Item 23-10.**

23.H.4 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes are acceptable because these changes have minimal effect on the safety analysis, and GDC 10 and 15 continue to be complied with. These design changes were also evaluated with respect to the adequacy of technical specifications that have been establishedfor the MSSVs. The staff finds these changes acceptable pending resolution of **Confirmatory Item 23-10**.

23.I Changes Related to the Implementation of P-17 for Rod Withdrawal Prohibit

23.I.1 Description of Proposed Changes

In a letter dated May 10, 2010 (ML101380275), Westinghouse proposed design changes related to the implementation of P-17 logic for Rod Withdrawal Prohibit. The current design requires the P-17 signal coincident with the Beacon Unavailable Signal to generate the automatic rod withdrawal prohibit. The proposed changes would remove the Beacon Unavailable Signal and the associated AND gate to enable an automatic rod withdrawal prohibit solely on the rate of change in nuclear power (P-17). The implementation of the P-17 logic to prohibit rod withdrawal is a conservative change from the current design. As a result of this proposed P-17 logic change, Westinghouse proposed to revise DCD Section 15.4.3.2 by revising the sequence of dropped rod event in DCD Table 15.4-1 and DCD Figures 15.4-1 through 4 to reflect this change.

23.I.2 Regulatory Basis

The regulatory basis for evaluating the changes related to the implementation of P-17 logic for Rod Withdrawal Prohibit is documented in Chapter 15 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The proposed design changes must also conform to the requirements of General Design Criteria (GDC) 10, GDC 13, GDC 20, and GDC 25 in 10 CFR Part 50, Appendix A.

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

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

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

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

23.I.2 Technical Evaluation

The proposed changes to enable the P-17 automatic Rod Withdrawal Prohibit solely on the rate of change in nuclear power affect the response to dropped rod control cluster assembly (RCCA) events, described in DCD Section 15.4.3.2.1. A drop of one or more RCCAs from the same group results in an initial reduction in the core power and a perturbation in the power distribution. In the automatic control mode, the plant control system detects the drop in the core power and initiates withdrawal of a control bank, which could result in power overshoot. The implementation of the proposed P-17 logic to prohibit automatic rod withdrawal prevents the potential power overshoot, and is therefore conservative relative to the current design. Therefore, the safety analysis of dropped RCCA events would be bounded by the existing analysis described in DCD 15.4.3.2.1, which demonstrated that the minimum Departure from Nucleate Boiling Ratio (DNBR) for one or multiple rod drops from the same group is greater than the DNBR limit.

Westinghouse has run a spectrum of transients, varying key input parameters to verify that DNBR limits continue to be met with the proposed P-17 logic implementation. The sequence of a rod drop event displayed in DCD Table 15.4-1 is considered to be representative of all of the cases run. The results of a representative dropped RCCA event are provided in the revised Figure 15.4.3-1 through 15.4.3-4. DCD Table 15.4-1 is also modified to reflect a representative case where the peak nuclear power occurs at time 21.7 seconds and peak core heat flux occurs at 24.2 seconds.

The staff has reviewed these proposed changes by Westinghouse to implement the P-17 logic for Rod Withdrawal Prohibit. Based on its review, the staff finds that this change does not affect the safety analysis of the dropped RCCA events described in DCD Section 15.4.3.2.1, and the analysis continues to satisfy the acceptance criteria of SRP Section 15.4.3 with respect to the minimum DNBR, peak pressure, and fuel cladding integrity. Therefore, the staff concludes that these proposed changes are acceptable.

23.I.4 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 proposed design changes are acceptable because they meet the requirements of GDC 10, GDC 13, GDC 20, and GDC 25.

23.J Changes Related to Post-Design Basis Accident Transmitters

23.J.1 Description of Proposed Changes

In letters dated May 25, 2010 (ML101470309) and July 29, 2010 (ML102150197), and October 20, 2010, Westinghouse proposed design changes to relocate seven containment pressure transmitters outside containment and connect to remote pressure sensors inside containment by sealed capillary tubing. These changes require the addition of four new containment penetrations, one for each safety division. Westinghouse also proposed to relocate 18 Category 1 Post Accident Monitoring System (PAMS) transmitters above the maximum Design Basis Accident (DBA) flood level. In addition, the applicant proposed to reduce post-accident operability time for 18 Category 2 PAMS transmitters from 4 months to 2 weeks.

23.J.2 Regulatory Basis

The regulatory basis for evaluating the changes to post-design basis accident (DBA) transmitter requirements is documented in Chapters 3, 5, 6, 7, and 9 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design."

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of General Design Criteria (GDC) 13 in 10 CFR Part 50, Appendix A, and should meet the guidance in RG 1.11, RG 1.97, and RG 1.151. The proposed changes were evaluated using the guidance provided in Standard Review Plan (SRP) Section 7.5, "Information Systems Important to Safety." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified by SRP Section 7.5 and RG 1.1, RG 1.97, and RG 1.151 (as applicable). The addition of four new containment penetrations was evaluated using the guidance provided in SRP Section 6.2.4, "Containment Isolation System," and SRP section 6.2.6, "Containment Leakage Testing," and the requirements of 10 CFR 50, Appendix J, "Containment Leak Rate Testing," and GDC 52, "Capability for Containment Leak Rate Testing."

23.J.3 Technical Evaluation

As documented in NUREG-1793, the staff reviewed and approved the post-DBA transmitter requirements in AP1000 DCD, Revision 15. The staff reviewed the applicant's proposed design changes using the review procedures described in SRP Sections 6.2.4, 6.2.6, and 7.5, requirements of GDC 13 and GDC 52 in 10 CFR Part 50, Appendix A, and 10 CFR Part 50, Appendix J, and the guidance in RG 1.11, RG 1.97, and RG 1.151.

Westinghouse proposed to relocate seven containment pressure transmitters outside containment. These changes will allow direct measurement of differential pressure across the containment shell. These changes also allow those seven transmitters to be located in a mild environment. Relocation of the seven transmitters outside of containment could address the

Limiting Conditions of Operation (LCO) in Technical Specifications that containment pressure shall be maintained between -0.2 psig to +1.0 psig, which are used in Safety Analysis.
[

] Relocation of the seven transmitters outside containment allows the plant to operate within Technical Specifications for containment pressure. Moving the transmitters outside containment also eliminates the need to include a DBA environmental allowance in the determination of channel accuracy and the setpoint for Passive Containment Cooling System (PCS) actuation.

Moving seven (7) containment pressure transmitters outside containment and connecting to remote pressure sensors inside containment by sealed capillary tubing requires the addition of four new containment penetrations, one for each safety division. In divisions A, B and C, the normal-range and wide-range transmitters share one capillary line and penetration. For division D there is no wide-range transmitter, so the normal-range transmitter is on its own capillary. These four new instrument Penetrations, P46, P47, P48, and P46 are identified in marked up DCD Tier 1 Table 2.2.1-1 and Figure 2.2.2-1, "Containment System." With **Confirmatory Item 23-11**, the staff will track this addition to the DCD Tier 1 table and figure.

GDC 55 or 56 usually require each line that penetrates the containment and is either a part of the reactor coolant pressure boundary or that connects directly to the containment atmosphere, to be provided with containment isolation valves. However, SRP Section 6.2.4 endorses the containment isolation provisions described in RG 1.11 for instrument sensing lines**.** RG 1.11 finds sensing lines with no isolation valves acceptable as long as the lines are sized to limit the potential offsite exposure to be below the guidelines of 10 CFR 100. The capillary tubing meets this criterion.

The four (4) new containment penetrations to accommodate the new instrument capillary tubing will be leak rate tested, as required by GDC 52, Capability for Containment Leakage Rate Testing, The four new penetrations will be Type A tested, which meets the requirements of 10 CFR 50, Appendix J and the guidance of SRP 6.2.6. This leak rate testing commitment will be added to DCD, Tier 2, Table 6.2.3-1, "Containment Penetrations and Isolation Valves." Also using **Confirmatory Item 23-11**, the staff will track this addition to DCD Table 6.2.3-1.

Westinghouse also proposed to move 18 Category 1 PAMS transmitters above the maximum DBA flood level within containment. This change is made to ensure that the 18 Category 1 PAMS transmitters are available following the DBA flood. [

]

Westinghouse also proposed to change the post-accident operation period for instruments that monitor Category 2 parameters. These changes would not affect Type A, B, or C primary post-accident parameters. These proposed design changes reduce the post-accident operation period from 4 months to 2 weeks. The Category 2 PAMS transmitter instruments are not required long term following a DBA, so their post-accident operation period can be changed from 4 months to 2 weeks. As Category 2 parameters, these parameters are not considered to be primary post-accident parameters for Type A, B, or C, and therefore are not required to be qualified long-term following a DBA.

In the markup for AP1000 DCD, Tier 1, Table 2.2.2-1, the applicant changed the qualification for harsh environment from "Yes" to "No" for the seven containment pressure sensors. This change is not justified because the seven containment pressure sensors are still inside containment, even though the seven transmitters are moved outside containment. The applicant also failed to identify the 18 Category 1 PAMS transmitters that are proposed to be relocated above the maximum DBA flood level. As a result, the staff issued RAI-DCP-CN64-ICE-01 requesting the applicant to justify the changes in the environmental qualification of the seven containment pressure sensors. Also, the applicant was requested to identify which Category 1 PAMS transmitters are relocated.

In the RAI response, the applicant restored a harsh environment qualification requirement for the seven containment pressure sensors that are still located inside containment. In the response, the applicant also identified all 18 Category 1 PAMS transmitters that are relocated above the maximum DBA flood level. After reviewing all proposed design changes and information provided in the RAI response, the staff finds that the design changes are acceptable. The markups to the above DCD Tier 1 and Tier 2 are being tracked as **Confirmatory Item 23-11**.

These proposed changes also affected the TS Bases for Engineered Safety Feature Actuation System Instrumentation, B 3.3.2. The staff found this change acceptable since it provides details of the system design described in the changes to Section 6.2.2 of the DCD.

23.J.4 Conclusion

After reviewing the proposed design changes, markups to DCD Tier 1 and Tier 2 and RAI responses, the staff finds that the proposed design changes meet the post-DBA monitoring requirements for the transmitters. Therefore, the staff concludes that the design changes proposed by Westinghouse are acceptable, pending the resolution of **Confirmatory Item 23-11** noted above.

23.K Changes to Startup Feedwater System and Chemical and Volume Control System Isolation Logic

23.K.1 Description of Proposed Changes

In letters dated May 10, 2010 (ML101380275), and July 29, 2010 (ML102150197), Westinghouse proposed design changes to add an AND logic to the Protection and Safety

Monitoring System (PMS) Functional Diagram Sheets 6 and 10 of Figure 7.2-1 in the AP1000 DCD to isolate the Startup Feedwater System (SFW) and close Chemical and Volume Control System (CVS) isolation valves earlier in the Steam Generator Tube Rupture (SGTR) transient sequence in order to maintain the margin to the Steam Generator (SG) overfill. The SG narrow range level high coincident with reactor trip limiting setpoint is proposed to be changed to 85% and the actuation signal added to Table 7.3-1, Table 15.0-6, and TS Table 3.3.2-1 and its Bases.The newly added AND logic is to combine Steam Generator System (SGS) narrow range level high with P-4 reactor trip.

23.K.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes of SFW and CVS isolation on SGS narrow range level high coincident with P-4 reactor trip is documented in Chapter 7 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design."

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of General Design Criteria (GDC) 13 and GDC 20 in 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in Standard Review Plan (SRP) Section 7.3, "Engineered Safety Features Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified by SRP Section 7.3 (as applicable).

23.K.3 Technical Evaluation

As documented in NUREG-1793, the staff reviewed and approved the Engineered Safety Features in Section 7.3 of AP1000 DCD Tier 2. The staff reviewed the applicant's proposed design changes to the SFW and CVS isolation logic by using the review procedures described in SRP Section 7.3 and requirements of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A.

The containment pressurization backpressure was not considered in the original transient analysis for the design basis accident of SGTR. However, the increase of the containment pressure resulting from heat transfer to containment via the Passive Residual Heat Removal (PRHR) Heat Exchanger (HX) impacts the boiling temperature of the In-containment Refueling Water Storage Tank (IRWST) water and results in a longer duration of SGTR breakflow. The analysis also showed that the SG overfill occurs at 12,739 seconds during a design basis SGTR accident with no operator actions modeled. These design changes have been proposed in order to maintain SG overfill margin without reliance on operator actions.

For the specific DCD changes, the applicant mainly added an AND logic to the PMS software logic (Sheets 6 and 10 of Figure 7.2-1), which combines the existing SG narrow range level high signal from either of the two SGs and P-4 reactor trip signal to isolate SFW and close CVS isolation valves. The narrow range level high signal for each SG results from a coincidence logic of two out of four divisions with bypass functionality. This change meets the reliability and testability requirements in GDC 21. Sections 7.3.1.2.13 and 7.3.1.2.15 in AP1000 DCD Tier 2 are also revised to include the new isolation signal for the SFW and CVS systems. The SG narrow range level high limiting setpoint at 85% and the actuation signal are added to Table 7.3-1, Table 15.0-6 and Tech Specs Table 3.3.2-1 and its bases. The staff finds that these design changes are acceptable. The markups to the above DCD Tier 2 are being tracked as **Confirmatory Item 23-12.**

The AP1000 design provides automatic protection actions to mitigate the consequences of a SGTR event. The automatic actions include reactor trip, actuation of the PRHR heat exchanger, initiation of core makeup tank flow, termination of pressurizer heater operation, and isolation of CVS and SFW on high SG narrow range water level. The proposed change to add the SG narrow level high coincident with P-4 reactor trip would isolate the CVC and SFW at a relatively lower SG level and is therefore conservative during the SGTR. This added trip isolation function would conservatively maintain the SG overfill margin without relying on operator actions.

The staff has reviewed the proposed change by Westinghouse to add an AND logic to the PMS software that combines the existing SG narrow range level high alarm and P-4 reactor trip to isolate SFW and CVS. Based on its review, the staff finds that the results of the analysis continue to show that the SG will not overfill with water, the maximum Reactor Coolant System (RCS) will not exceed 110 percent of design pressure, and the minimum Departure from Nucleate Boiling Ratio (DNBR) will remain greater than the safety DNBR limit. Therefore, the staff concludes that this proposed change is acceptable.

The proposed design changes also include the addition of isolation functions to the Engineered Safety Feature Actuation System (ESFAS) system. If a narrow range high SG level signal is received, coincident with a P-4 reactor trip, signals are sent to isolate the CVS and SFW. In the case of CVS, this prevents additional make-up to the RCS in case of a SGTR casualty. The SFW isolation prevents an overfill condition, and therefore an over cooling condition, if the reactor is tripped and the SFW system is not isolated. These changes affect the TS for ESFAS Instrumentation, Table 3.3.2-1, as well as the discussion in bases sections B 3.3.2 and B 3.4.17. The staff finds these changes acceptable since they conform to the guidance provided in the Standard Technical Specifications, and add appropriate isolation for protection in these events, as discussed in the changes to Section 15.6.3 of the DCD.

23.K.4 Conclusion

After reviewing the proposed design changes and markups to DCD Tier 2 sections and tables, the staff finds that the proposed changes meet the applicable requirements and guidance for the safety-related functions. Therefore, the staff concludes that these design changes are acceptable, pending the resolution of **Confirmatory Item 23-12**, as noted above.

Based on the above evaluation, the staff concludes that the proposed design changes are acceptable because the SGTR analysis continues to show margin to SG overfill, and meet the pressure and core safety DNBR limits, even though they are not required to be met for a SGTR event.

23.L Changes to Passive Core Cooling System Injection Lines

23.L.1 Introduction

In letters dated May 25, 2010 (ML101470309), August 2, 2010 (ML102160742), and August 20, 2010 (ML102350441), Westinghouse proposed design changes to the passive core cooling system (PXS) injection lines to address gas intrusion, which included the addition of manual vent valves, pipe stubs, manual drain valves, instrumentation, and re-routing accumulator discharge line connections to the direct vessel injection (DVI) lines. These manual vent valves are located in containment rooms that are constructed to permit entry during full power conditions. Analyses have shown that with enough noncondensible gas accumulation, in-containment refueling water storage tank (IRWST) injection through the affected flow path could be delayed. Therefore, excessive amounts of noncondensible gas accumulation in the high point vents of the IRWST injection lines may potentially impact the passive injection of IRWST borated water into the reactor vessel. However, the presence of a small amount of noncondensible gases does not imply that the IRWST injection capability is immediately inoperable but rather that gases that are accumulating need to be vented. The venting of these gases requires containment entry to manually operate the vent valves. Since gas accumulation is a slow process, plant operators have sufficient time to vent the noncondensible gases upon receiving an alarm. To incorporate the proposed change, the system modification would include revision to Tier 1 Figure 2.2.3-1, Tier 2 Figures 5.1-5 and 6.3-1, and Tier 2 Tables 3.2-3, 3.9-17, 3.11-1 and 3I.6-3 to identify the new components. In addition, the proposed change includes three new DCD Subsections 6.3.6.3, 6.3.6.3.1, and 6.3.6.3.2, which provide a description of the plan to mitigate gas intrusion and accumulation by means of periodic system surveillance and venting procedures, review of pipe layout and routing drawings to identify high-point vent and low-point drain locations, and assessment of system design features. Also, to ensure proper operational implementation of this design change since gas accumulation has the potential to impact safety-related systems, IRWST required actions and surveillance requirements are added to DCD Chapter 16.1, “Technical Specifications.” For controls of IRWST operations, the affected TS sections are TS 3.5.6, TS 3.5.7, and TS 3.5.8 for during Modes 1 through 4, Mode 5, and Mode 6, respectively.

23.L.2 Regulatory Basis

General Design Criterion (GDC) 27 in 10 CFR Part 50, Appendix A, requires that the emergency core cooling system be designed to provide the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

GDC 27 is applicable because upon actuation the emergency core cooling system provides rapid injection of borated water to ensure reactor shutdown. Injection of borated water provides negative reactivity to reduce reactor power to residual levels and ensures sufficient cooling flow to the core. Meeting the requirements of GDC 27 for the emergency core cooling system augments the protection for the primary fission product barrier by providing a means to ensure that the core, under postulated accident conditions, can be safely shut down and will be maintained in a coolable geometry. Noncondensible gas accumulation has the potential to delay injection of borated water. Such a delay would impact the moderating and heat removal capabilities thus providing an adverse challenge to the primary fission product barrier and maintenance of coolable core geometry.

GDC 35 requires, among other things, that the emergency core cooling system be designed to provide an abundance of core cooling to transfer heat from the core at a rate such that fuel and clad physical damage that could interfere with continued effective core cooling is prevented.

GDC 35 is applicable because following a breach in the reactor coolant pressure boundary, reactor coolant is lost at a rate determined by several factors, including break size and RCS pressure. The emergency core cooling systems are relied upon to inject adequate cooling water into the reactor coolant system (RCS) during a loss of coolant accident (LOCA) and to circulate the water through the core to provide for core cooling. The emergency core cooling system must inject cooling water at a rate sufficient to ensure that the calculated changes in core geometry will be such that the core remains amenable to cooling, and that the calculated cladding oxidation and hydrogen generation meet the specified performance criteria. Based on analysis, noncondensible gas accumulation has the potential to delay injection of borated water; such a delay may adversely affect fuel and cladding physical configuration with potential to challenge the coolability of the core geometry.

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the IRWST design and operating information described in DCD Section 6.3. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance specified by SRP Chapter 16.

23.L.3 Technical Evaluation

Noncondensible gas accumulation effects on the performance of safety related "active" emergency core cooling, residual heat removal, and containment spray systems have been documented based on operating plant experience and system analysis. Some of these events are identified in the NRC generic letter (GL) 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems" (ML072910759) which illustrates the need for license holders to review their safety systems to resolve this issue. The focus of this GL is centered on four principal concerns: (1) licensing basis, (2) design, (3) testing, and (4) corrective actions. Also, the GL provides an assortment of pertinent generic communications documents related to this subject which was helpful in the staff's evaluation. Although the GL refers to active safety systems focusing on gas accumulation effects on pump performance, Westinghouse expanded its existing design activity to include AP1000 passive safety systems to "integrate the draft Interim Staff Guidance (ISG) document ISG-019 regarding gas intrusion assessment guidance into the design process, helping to confirm that the potential issues identified in" the GL have been addressed within the AP1000 design. However, this safety evaluation is related to proposed design changes that primarily address gas intrusion and accumulation in the PXS.

The proposed changes include the addition of the following components:

- 8 manual maintenance vent valves in 6 PXS passive injection and recirculation line piping high point locations;
- 4 pipe stubs with maintenance vents and associated valves, line routing to tee into core makeup tank (CMT) vent line routing to the reactor coolant drain tank (RCDT);
- Remote pipe stub gas void indications at the outlets of each of the IRWST passive injection squib valves;
- Re-routing the 2 accumulator discharge line connections to both DVI line vertical cold trap (riser) pipes to tee in physically above (in elevation) and downstream of the associated IRWST connection into the DVI riser pipes (instead of locating the accumulator connection upstream of and below the IRWST tee);
- 20 manual maintenance drain valves in fourteen PXS passive injection and recirculation piping, 5 normal residual heat removal system (RNS) piping locations, and 1 RCS piping location. (Note: The RNS and RCS drains are unrelated to gas intrusion effects but part of design finalization to consolidate required PXS line changes and piping analyses.)

Issues identified during staff review were addressed in three RAIs, identified as RAI-DCP-CN66-SRSB-01 through RAI-DCP-CN66-SRSB-03, with the primary issues summarized below:

1. RAI-DCP-CN66-SRSB-01
    - Explain the exclusion of vent valves to Train B of the containment recirculation.

2. RAI-DCP-CN66-SRSB-02
    - Provide the bases for selecting 0.2 cubic feet as the noncondensible gas volume limit in the Technical Specifications (TS) surveillance requirements (SR).
    - Describe the volume measuring method of the noncondensible gas.
    - Describe how the 0.2 cubic feet volume of noncondensible gas is accounted for in the safety analyses for LOCA and post-LOCA long term cooling.

3. RAI-DCP-CN66-SRSB-03
    - Provide justification for excluding the new instrumentation and valve components in ITAAC.
    - Describe the calibration frequency of RTD switches used to measure the volume.

- Discuss whether the calibration frequency is controlled by TS.

In its August 20, 2010 response to RAI-DCP-CN66-SRSB-01, the applicant stated that containment recirculation Train A and Train B paths have different configurations to avoid other components and piping lines. Whereas Train A was routed with two high points to circumvent interference with other plant components, Train B layout constraints were less complicated; hence, there are no local high points that could accumulate gas. After further review, the staff finds the RAI response acceptable.

With regard to RAI-DCP-CN66-SRSB-02, the applicant responded that originally the 0.2 cubic feet noncondensible gas volume limit was selected for the CMT inlet high point pipe TS Surveillance Requirements (SR) 3.5.2.4 because it was slightly larger than the internal volume between the sensors location on the pipe stubs and the first normally closed manual vent valve that was previously evaluated and approved. Since the gas intrusion and accumulation is a slow process and the alarm occurs before voiding is extended into the line, the relocation allows sufficient time for the operators to vent the line before gas buildup could adversely affect the passive safety operability performance. Therefore, initially the PXS volume limit of 0.2 cubic feet was selected to be consistent with the CMT SR requirements since the IRWST injection line high points, pipe stubs, and sensor configuration is similar to that of the CMT. However, based on the applicant's response, the staff concludes that the sensor location not the volume limit is relevant to proper performance of venting the line because the sensor is configured as a level switch, as discussed below. Therefore, the applicant has proposed to remove the reference to the volume limit from TS SRs 3.5.2.4, 3.5.4.3, and 3.5.6.3 and replace it with "has not caused the high-point-water level to drop below the sensor." Also, the corresponding TS Bases would be revised to appropriately reflect the removal of the volume limit. The staff finds this proposed change acceptable because it refers to the correct function of the sensor.

In addition, the applicant stated that the volume is not directly measured but the change of state from liquid to gas provides the mechanism that controls the sensor output. The sensors are thermal dispersion sensors consisting of one heated RTD and one non-heated RTD configured such that the RTDs function as a thermal dispersion level switch where the temperature difference is based on the conductivity of the medium in contact with the two elements. When the RTDs are exposed to gas, the change in the differential temperature of the elements causes the output switch to actuate providing an alarm to the operator. This sensor configuration has been used in other plant applications with reliable results. Therefore, the staff finds this method to be acceptable.

With respect to safety analyses for LOCA and post-LOCA long term cooling, the applicant stated that the proposed volume is not considered in any safety analyses because the pipe stub and sensor configuration allows for sufficient time for venting before the actual injection path begins to void. The staff finds this acceptable because the alarms trip with less than half the volume of gas that could affect the injection flow performance, the rate of gas accumulation is expected to be sufficiently slow, and there are no credible postulated gas intrusion mechanisms for these locations.

In its August 20, 2010, response to RAI-DCP-CN66-SRSB-03, the applicant stated that the proposed instrumentation switches provide pre-event operability confirmation and are not used for reactor trip, safeguards actuation, or post-accident monitoring functions. As such, these switches are not required to be part of ITAAC or TS. Also, since the switches are nonsafety related process instrumentation, there are limited periodic calibration or functional checking requirements. The staff considers acceptable the response that the process switches do not meet the requirements to include them in ITAAC and TS.

The proposed changes also included the addition of two new Actions and one SR to TS 3.5.6, TS 3.5.7, and TS 3.5.8 related to one or two IRWST injection line(s) "inoperable due to presence of noncondensible gases" and surveillance of the new sensors indication. Also, the appropriate TS Bases would be revised to reflect this change. The staff reviewed the new Actions, SR, and Bases, which provide for conditions, required actions, completion times, and surveillance requirements. The staff found that these parameters conform to guidance provided in the STS and reflect the IRWST injection line gas intrusion and accumulation process analysis described in DCD Section 6.3. Therefore, the staff finds the TS change, as modified in the response to RAI-DCP-CN66-SRSB-03 noted above, to be acceptable.

In addition, the proposed change would provide for system surveillance and venting procedures, as described in DCD Chapter 13, to include inspection of the passive safety system location equipped with manual vent valves to eliminate any identified gas accumulation. Procedural inclusion of vent component inspection is acceptable to the staff because timely inspections could reduce intrusion and accumulation of gases in the injection lines.

In accordance with SRP 6.3, the staff's evaluation of these proposed design changes must ensure that compliance with GDC 27 and GDC 35 is satisfied. As stated in the SRP, meeting the requirements of:

> …GDC 27 for the ECCS augments the protection for the primary fission product barrier by providing a means to ensure that the core, under postulated accident conditions, can be safely shut down and will be maintained in a coolable geometry.
>
> …GDC 35 ensures that the ECCS, assuming a single failure, can provide core cooling under accident conditions sufficient to maintain the core in a coolable geometry and to minimize the reaction of water with the fuel cladding.

The staff evaluation of the proposed change concludes that the reliability and performance of PXS is improved. The requirements of GDC 27 and GDC 35 are satisfied based on the following:

- TS requirements that ensure that the PXS is operated in safe condition;

- Adequate procedural inspections to identify and eliminate gas intrusion and accumulation; and
- Sufficient monitoring and alarming features in the control room to ensure timely venting.

23.L.4 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes to provide venting of noncondensible gases in the injection lines enhance the reliability and performance of the PXS to ensure that the core, under postulated accident conditions, can be safely shut down, maintain a coolable geometry, and minimize the reaction of water with the fuel cladding, thus satisfying the requirements of GDC 27 and GDC 35. Therefore, the proposed design changes are acceptable.

23.M Changes to Squib Valve Actuation Time

23.M.1 Description of Proposed Changes

In letters dated April 28, 2010 (ML101180111), July 29, 2010 (ML102150197), and August 12, 2010 (ML102290041), Westinghouse proposed design changes to the Protection and Safety Monitoring System (PMS) and Diverse Actuation System (DAS) controls for Passive Core Cooling System (PXS) In-containment Refueling Water Storage Tank (IRWST) injection squib valves actuation by incorporating a 5-second time delay in PXS actuation control for the automatic (and manual) actuation circuitry between the firing of the first and second valve for each pair of squib valves in the same process line. These proposed design changes address the unacceptable design stress on Direct Vessel Injection (DVI) line piping during simultaneous actuation of squib valves in the parallel configuration. Westinghouse also made similar changes to PMS controls for PXS containment sump recirculation squib valves in parallel path of the same process line. These proposed design changes revise functional diagrams in DCD Figure 7.2-1 by adding Note 5 in Sheet 16, “In-containment Refueling Water Storage Tank Actuation,” and Note 6 in Sheet 20, “Diverse Actuation System Logic, Manual Actuation,” which state that, for redundant components in a parallel configuration, the components use different time delays to prevent simultaneous actuation.

23.M.2 Regulatory Basis

General Design Criterion (GDC) 35 in 10 CFR part 50, Appendix A, requires that an emergency core cooling system (ECCS) be provided to transfer heat from the reactor core following any loss of coolant accident (LOCA) at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. 10 CFR 50.46 specifies that the calculated ECCS cooling performance be shown to comply with the acceptance criteria specified in 10 CFR 50.46(b).

The regulatory basis for evaluating the changes in squib valve actuation time is documented in Chapters 6, 7, and 15 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.”

Reviews of the proposed changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes should also conform to the guidelines of IEEE Std. 7-4.3.2, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by RG 1.152, “Criteria for Use of Computer in Safety Systems of Nuclear Power Generating Stations,” and the SRM on SECY-93-087. The proposed changes were evaluated using the guidance provided in Standard Review Plan (SRP) Section 7.3, “Engineered Safety Features Systems,” and Section 7.8, “Diverse Instrumentation and Control Systems.” Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified in SRP Sections 7.3 and 7.8 (as applicable).

23.M.3 Technical Evaluation

The proposed changes add notes in the PMS and DAS functional diagrams in DCD Figure 7.2-1 to state that the redundant components in a parallel configuration use different time delays to prevent simultaneous actuation from adversely impacting accident timing in the safety analyses. For the automatic depressurization system (ADS) stage-4 squib valves, DCD Table 15.6.5-10 specifies the actuation time delay of 60 seconds between the ADS-4A and ADS-4B squib valves. The proposed change was only required for the IRWST injection and containment recirculation sump squib valves since the ADS stage-4 valves already have a 60-second time delay difference in their automatic actuation circuitry.

In its August 12, 2010, response to RAI-DCP-CN08-SRSB-01ML102290041, Westinghouse stated that its proposed 5-second time delay for the actuation of redundant squib valves in each IRWST injection path and containment recirculation flow path, respectively, was included not to correct the system safety performance, but rather to protect the integrity of components, piping and structural and mechanical modules, in relationship to the specific squib valves actuation characteristics. Westinghouse performed an evaluation of LOCA safety analyses with the 5-second delay and found that the impact is not significant. The AP1000 design has the squib valves actuate a relatively long time before they are needed, on the order of several hundreds of seconds, so that the very short squib valve actuation delay of 5 seconds or reasonably longer time for the second squib valve in each path is not significant to the IRWST injection and/or containment recirculation performance of the plant events. Therefore, the safety analyses have not been revised to implement this 5-second time delay. The staff agrees with this explanation. The existing safety analyses in DCD Section 15.6.5 for the large-break LOCA, small-break LOCA, and long-term cooling remain valid and in compliance with GDC 35 and 10 CFR 50.46. Therefore, this proposed change is acceptable.

As documented in NUREG-1793, the staff reviewed and approved Functional Diagram Sheet 16 of Figure 7.2-1 for PXS IRWST injection and containment recirculation isolation squib valves actuation and Functional Diagram Sheet 20 of Figure 7.2-1 for DAS manual actuations for

IRWST injection squib valves. The staff reviewed the applicant's proposed design changes to the two functional diagrams using the review procedures described in SRP Sections 7.3 and 7.8.

10 CFR 52.47 requires that an application contain a sufficient description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluation. The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Hence, the staff issued RAI-DCP-CN08-ICE-02 requesting the applicant to provide clarification for the time delay inconsistency between the design change description and marked-up diagrams and also justify the need for a 5 second time delay between the firing of the two IRWST injection squib valves. In the RAI response, the applicant explained that the 5-second time delay is required to prevent a negative structural impact on the supporting bracket from induced vibrations if both squib valves are fired simultaneously. The firing of one of the commonly mounted explosive valves followed by a five seconds time delay will have no negative impact on operation. The design change provides an improved protection of the safety function of the IRWST injection and containment recirculation by avoiding potential adverse consequences related to the simultaneous firing of the two squib valves physically housed on the same structural frame module. The staff finds that the design changes and response to the RAI are acceptable. Therefore, RAI-DCP-CN08-ICE-02 is considered resolved. The markups to Functional Diagram Sheets 16 and 20 of Figure 7.2-1 in the DCD is being tracked as **Confirmatory Item 23-13**.

23.M.4 Conclusion

Pending the resolution of **Confirmatory Item 23-13**, the staff concludes that the proposed changes of adding notes to the PMS and DAS function diagrams in DCD Figure 7.2-1 are acceptable because they have negligible effect on the LOCA safety analysis, and continue to comply with GDC 35 and 10 CFR 50.46.

In addition, the staff concludes that the applicant's proposed design changes are acceptable because they have no negative impact on operation, and provide improved protection of the integrity of components, piping and structural and mechanical modules.

23.N Changes Related to Anticipatory Reactor Trip in the Event of an Inadvertent Passive Residual Heat Removal Actuation

23.N.1 Description of Proposed Changes

In a letter dated May 10, 2010 (ML101380275), and in two letters dated July 29, 2010 (ML102150197, ML102150198), Westinghouse proposed design changes related to anticipatory reactor trip in the event of an inadvertent Passive Residual Heat Removal (PRHR) actuation. Westinghouse proposed design changes to the Protection and Safety Monitoring System (PMS) to include one additional reactor trip to mitigate one inadvertent PRHR event caused by the opening of either of the two PRHR Heat Exchanger (HX) discharge valves when the reactor is in power operation. In the PMS logic control system, a reactor trip that is to be added will be generated when either of the two PRHR HX discharge valves comes off its fully shut seat while the reactor is at power. Another proposed design change is to adjust the frequency of the Inservice Test (IST) for the two PRHR HX discharge valves from the current once every quarter to every cold shutdown.

In order to mitigate an inadvertent PRHR actuation event, Westinghouse proposed changes to add a PRHR actuation reactor trip function, which is based on the PRHR HX control valve indication, to DCD Chapter 7.2, "Reactor Trip," Related to this design change, a sentence is added to DCD Section 6.3.7.6.1, which states that for the PRHR HX discharge valves, valve position indication is used to initiate a reactor trip upon opening of these valves while the reactor is at power. Also, DCD Section 15.1.6, "Inadvertent Operation of the PRHR Heat Exchanger," is revised to state that to prevent the reactivity increase (as a result of inadvertent actuation of the PRHR HX event) from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve comes off of its fully shut seat. DCD Subsection 15.1.6.2 is revised to state that since a reactor trip is initiated as soon as the PRHR discharge valves are not fully closed, this event is essentially a reactor trip from the initial condition and requires no separate transient analysis. In DCD Table 15.0-4a, the reactor trip function on the PRHR discharge valve not closed with a time delay of 1.25 seconds is added, and in DCD Table 15.0-6, this reactor trip function is listed for the inadvertent operation of the PRHR event.

23.N.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes to the AP1000 PMS reactor trip system is documented in Chapters 7, 15, and 16 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design."

Reviews of the changes are also based on meeting the relevant requirements of regulations and guidelines in 10 CFR 50.55a(h), 10 CFR 52.47, and General Design Criteria (GDC) 20 and 21 of 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in Standard Review Plan (SRP) Section 7.2, "Reactor Trip Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified in SRP Section 7.2 (as applicable).

GDC 10 in 10 CFR Part 50, Appendix A, specifies that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDL) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOO). An inadvertent operation of the PRHR heat exchanger is an AOO, and therefore must comply with GDC 10.

23.N.3 Technical Evaluation

As documented in NUREG-1793, the staff reviewed and approved the reactor trip system as specified in Section 7.2 of AP1000 DCD, Revision 15. The staff reviewed the applicant's design changes to the reactor trip system using the review procedures described in SRP Section 7.2.

If either of the two PRHR HX discharge valves inadvertently comes off its normally fully closed seat while the reactor is at power, it allows a slug of cold water to be introduced into the reactor core through the PRHR HX to cause a marked increase in reactor power, which may exceed the fuel design limits. Therefore, a reactor trip is needed to mitigate this inadvertent PRHR event introduced by the opening of either of the two PRHR HX discharge valves, and prevent the fuel design limits from being exceeded. However, in the event of such an inadvertent PRHR event, the current reactor trip design included in the AP 1000 DCD, Revision 17 will not be able to function fast enough to mitigate this inadvertent PRHR event. Therefore, the proposed design changes result in an improved reactor protection system to mitigate the inadvertent PRHR event. In addition, the 2-out-of-4 control logic with bypass capability is also used for this reactor trip condition in this DCP. The staff finds that all those changes meet applicable requirements in 10 CFR 50.55a(h), GDC 20, and GDC 21. The applicant made necessary changes to DCD Tier 1 Section 2.5.2, Tier 2 Section 7.2, and other related tables and figures. The staff finds that the proposed changes to reactor trip logic for an inadvertent PRHR actuation are acceptable because they satisfy the requirements of 10 CFR 50.55a(h), GDC 20, and GDC 21. Markups to DCD Tier 1 Section 2.5.2, Tier 2 Section 7.2, and other related tables and figures are being tracked as **Confirmatory Item 23-14**.

An inadvertent operation of PRHR HX causes an injection of relatively cold water into the Reactor Coolant System (RCS), resulting in a reactivity insertion due to a negative moderator temperature coefficient. Currently, several reactor trip functions are available to mitigate the event, including the overpower ΔT and overtemperature ΔT trip functions, to prevent a power increase, which could lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis DNBR limit. The safety analysis of the limiting inadvertent operation of PRHR HX, presented in DCD Section 15.1.6, shows that, without taking credit for a reactor trip function, the nuclear power rises to about 120 percent temporarily, then drops and reaches a new equilibrium condition at about 108 percent of the nominal value. The RCS pressure and minimum DNBR are within the respective limits.

The proposed design change to add a PRHR actuation reactor trip function will enhance the reactor protection to trip the reactor upon event initiation. In its July 29, 2010, response to RAI-DCP-CN60-SRSB-01, Westinghouse stated that it has performed an evaluation of the operation of the PRHR HX transient assuming a conservative 1.25 second reactor trip response time, which covers the time to sense the opening of a PRHR valve through initial insertion of control rods. A confirmatory analysis using LOFTRAN shows that the minimum time between the opening of the PRHR valves and any colder water reaching the reactor core inlet is at least 2 seconds. Since the control rods are already inserting into the core before any colder water reaches the core inlet, there will not be an adverse power increase for this transient. Since an inadvertent operation of the PRHR HX event can only be caused by an inadvertent opening of the PRHR HX discharge valves (either by operator error, false actuation signal, or malfunction of a discharge valve), and a reactor trip will be initiated as soon as a PRHR discharge valve comes off the closed seat position, the consequence is mitigated as soon as the inadvertent PRHR HX operation event occurs. The proposed design changes continue to comply with SAFDL and no additional safety analysis is needed.

Each of the two PRHR HX discharge valves has one set of existing class 1E magnetic valve position indicators (VPIs), one for the open position indication and one for the close position indication. The design change will use the existing close position indicator and add three more class 1E magnetic VPIs to give a total of four closed signals per valve. This configuration is necessary for the "two out of four" logic required for a reactor trip signal. The proposed change in DCD Subsection 6.3.7.6.1 merely indicates that the PRHR HX discharge valves' position indication is used for the reactor trip function, and is, therefore, acceptable.

Westinghouse's proposed design changes add a reactor trip signal generated from the opening of valves in the PRHR system. This trip would minimize the effect of the anticipated reactivity excursion due to cold water addition by the PRHR system initiating flow. This change affected the TS for Reactor Trip System Instrumentation, Table 3.3.1-1, as well as discussions in bases sections B 3.1.1 and B 3.3.1. The additional information included in this change added appropriate descriptions of the trip function and purpose, and created a trip function in Table 3.3.1-1 that provides adequate protection against this event. The staff finds this change acceptable since it conforms to the format and trip functions described in the Standard Technical Specifications (STS).

Westinghouse's proposed design changes also include changing the inservice testing full-stroke exercising requirement for the PRHR heat exchanger valves PXS-V108a and PXS-V108b from once a quarter to every Cold Shutdown. Valves PXS-V108a and PXS-V108b are Class 1, Category B air-operated valves that function as discharge valves for the PRHR heat exchanger. The inadvertent actuation of the PRHR heat exchanger causes an injection of relatively cold water into the reactor coolant system. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. To prevent this reactivity increase from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve comes off its full shut seat. The staff considers changing the inservice testing full-stroke exercising requirement for valves PXS-V108a and PXS-V108b from once a quarter to every Cold

Shutdown to be acceptable based on ASME Code (2004E) ISTC-3521(c), which states that if exercising is not practicable during operation at power, it may be limited to full-stroke testing during cold shutdown.

23.N.4 Conclusion

After reviewing the proposed changes and markups to associated DCD Tier 1 and Tier 2 sections, tables, figures, and technical specifications, the staff finds that the proposed changes provide an improvement to the reactor protection system. The staff concludes that the proposed changes to add a PRHR actuation reactor trip function is acceptable because it will trip the reactor as soon as an inadvertent operation of PRHR HX event occurs, thereby ensuring that the SAFDL will not be exceeded. Therefore, the staff concludes that these proposed design changes are acceptable, pending resolution of **Confirmatory Item 23-14**.

23.O Changes to Reactor and Turbine Trips Functional Logic of Diverse Actuation System

23.O.1 Description of Proposed Changes

In letters dated May 20, 2010 (ML101380275) and July 29, 2010 (ML102150197), Westinghouse proposed design changes to add a reactor trip and turbine trip to the functional logic of the Diverse Actuation System (DAS) based on the 2-out-of-2 control logic of the Passive Residential Heat Removal (PRHR) high hot leg temperature sensor outputs.

23.O.2 Regulatory Basis

The regulatory basis for evaluating these proposed design changes of DAS PRHR high hot leg temperature logic is documented in Chapter 7 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design."

Reviews of the changes are also based on meeting the relevant requirements of regulations and guidelines in 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of General Design Criteria (GDC) 13 and 20 in 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in Standard Review Plan (SRP) Section 7.8, "Diverse Instrumentation and Control Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified by SRP Section 7.8 (as applicable).

23.O.3 Technical Evaluation

As documented in NUREG-1793, the staff reviewed and approved the DAS in AP1000 DCD Tier 1 Section 2.5.1 and Tier 2 Section 7.7.1.11. The DAS system in the certified AP1000 DCD, Revision 15, uses the 2-out-of-2 control logic, which is based on the Probabilistic Risk Analysis

(PRA) evaluation. The staff reviewed the proposed design changes using the review procedures described in SRP Section 7.8 and applicable regulations of GDC 13 and 20 in 10 CFR Part 50, Appendix A.

According to the modeling in the Probabilistic Risk Assessment, the original DAS design in the AP1000 DCD should initiate a reactor trip and turbine trip for Anticipated Transients without Scram (ATWS) sequences with the main feedwater available. Since the main feedwater is still available, the low steam generator water level signal will not be generated to initiate the reactor or turbine trip in the DAS. Therefore, the high hot leg temperature signal is needed in the DAS to trip the turbine or the reactor via the control rod motor-generator (MG) sets.

10 CFR 52.47 requires that an application contain a sufficient description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluation. The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Hence, the staff issued RAI-DCP-CN63-ICE-01 requesting the applicant to provide justification for adding another time delay for opening PRHR discharge valves, which is not explained in the DCP. The applicant's July 29, 2010, RAI response stated that the time delay has been provided in the functional design to support sequencing of the output field devices associated with a system-level control function. This delay time is added to be consistent with the use of timers in the existing functional design (e.g., low steam generator water level logic). The staff finds that the response to the RAI is acceptable.

After reviewing the proposed design changes and RAI responses, the staff finds that the proposed design changes are acceptable. The markups to DCD Tier 1 Section 2.5.1, Tables 2.5.1-1, 2.5.1-3, 2.5.1-4, Tier 2 Section 7.7.1.11, Functional Diagram Sheet 19 of Figure 7.2-1, Tables 14.3-3, 14.3-6, and 16.3-2 are being tracked as **Confirmatory Item 23-15**.

23.O.4 Conclusion

After reviewing the proposed design changes, including markups to DCD Tier 1 and Tier 2 and RAI responses, the staff finds that the proposed changes will meet the required DAS functions as modeled in the PRA and related regulatory criteria. Therefore, the staff concludes that these proposed design changes are acceptable, pending resolution of **Confirmatory Item 23-15** noted above.

23.P Changes to Steam Generator System Instrument Piping

23.P.1 Description of Proposed Changes

In a letter dated July 8, 2010 (ML101930143), Westinghouse proposed design changes in the material for the steam generator system instrument piping. These design changes would modify Figure 10.3.2-1 of the AP1000 DCD to specify stainless steel piping for all AP1000 Quality Classes B and C instrument piping for the steam generator system. AP1000 Quality Classes B and C are designed and fabricated to ASME Code, Section III, Class 2 and 3, respectively. The main steam supply system (MSSS) includes the AP1000 steam generator system.

23.P.2 Regulatory Basis

The staff reviewed and evaluated the proposed design changes in accordance with the guidance in SRP Section 10.3.6 to ensure that the ASME Code, Class 2 and 3 MSSS components use material specified in Sections II and III of the ASME Code, thereby meeting the requirements of GDC 1 and 10 CFR 50.55a.

23.P.3 Technical Evaluation

Figure 10.3.2-1 modified all of the ASME Code, Section III, Class 2 and 3 steam generator system instrument lines, which typically are 1 inch NPS and less, to be a corrosion resistant (stainless steel) material. Section 10.3.6.2 of the AP1000 DCD specifies that the material selection and fabrication for ASME Code, Section III, Class 2 and 3 are in accordance with the requirements of ASME Code, Section III, Class 2 and 3 components outlined in Sections 6.1.1.1 and 6.1.1.2 of the AP1000 DCD. Section 6.1.1.1 of the AP1000 DCD specifies that pressure-retaining materials meet the requirements of Articles NC-2000 and ND-2000 of the ASME Code, Section III for Class 2 and 3 components, respectively. The staff notes that Articles NC-2000 and ND-2000 for ASME Code Classes 2 and 3, respectively, specify that the material be in accordance with Section II of the ASME Code. Based on this information, the staff finds that the use of stainless steel material specified in ASME Code, Section II for ASME Code Classes 2 and 3 steam generator instrumentation piping, typically 1 NPS or less, is acceptable. The staff's acceptance is based on the use of stainless steel for the instrumentation piping, which is more corrosion resistant than carbon steel, and that the material will be procured in accordance with ASME Code Section II and fabricated in accordance with ASME Code Section III. The staff identifies **Confirmatory Item 23-16** to verify that the proposed changes are included in Revision 18 to the AP1000 DCD.

23.P.4 Conclusion

The staff concludes that the proposed use of stainless steel instrument piping for the steam generator system provides higher corrosion resistance than carbon steel, and that the material will be procured and fabricated in accordance with ASME Code Sections II and III as specified in the guidance of SRP 10.3.6, and therefore meets the requirements of GDC 1 and 10 CFR 50.55a. The staff concludes that the proposed design changes, upon resolution of **Confirmatory Item 23-16**, are acceptable.

23.Q Changes To The Steel Containment Vessel Girder And Polar Crane Rail Clip

23.Q.1 Description of Proposed Changes

In letters dated April 26, 2010 (ML101180111); August 26, 2010 (ML102430183); and September, 16, 2010 (ML102630015), Westinghouse proposed design changes to the steel containment vessel girder and polar crane rail clip. These design changes apply to the safety-related structure (containment vessel) and the heavy load handling system (heavy load lifting equipment and support) and include the following changes:

- Increase in the thickness of the girder top plate from 1.5 inches to 1.75 inches.
- Change in the rail support from Gantrex Pad to a bolted clip design.
- Extension of the steel containment vessel (SCV) girder inward by 2-3/4 inches.

23.Q.2 Regulatory Basis

Sections 3.8.2 and 9.1.5 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," address the function and acceptability of the steel containment vessel (SCV) associated support structures for the overhead heavy load handling system in the AP1000 standard design. Acceptability of the proposed design change is evaluated to determine whether the design meets the relevant requirements specified in the regulations of 10 CFR Part 50, Appendix A, Criterion 2; General Design Criterion (GDC) 2 in 10 CFR Part 50, Appendix A, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

23.Q.3 Technical Evaluation

The SCV associated support structures are designed to provide the necessary support for the overhead heavy load handling (polar crane) system in the AP1000 standard design. The polar crane systems are safety-related and are used to handle heavy equipment such as the integrated head package, steam generators, and heat exchangers. As stated in the DCD changes included in this design change, the design of the SCV associated support structures for the polar crane system support is consistent with the SSE design of equipment anchorages of seismic Category I equipment. The girder top plate and the new bolted clip design for the rail support are analyzed to show that the structure can withstand an SSE event.

During the staff's evaluation of the proposed design changes to the SCV associated support structures, it was determined that additional information was required from the applicant to complete the staff's evaluation. The staff generated RAI DCP-CN-5-SEB-01 requesting (1) An analysis including materials and design basis loading used to demonstrate that the increase of one quarter (1/4) inch from 1-1/2 inches to 1-3/4 inches thick in top plate of the crane girder is adequate to support the design-basis loadings including a seismic load; (2) An analysis to demonstrate the new rail clip design, including clip spacing is sufficient to meet the design-basis load requirements including the seismic load demands, and (3) the basis for determining the extent of the inward extension of the CV girder top plate.

The applicant's response to RAI DCP-CN-5-SEB-01 provided sufficient information on the design analysis for the proposed design changes to support the staff's evaluation. In addition to evaluating the applicant's proposed design changes and RAI responses, the staff performed two audits.

With respect to the SCV PC girder top plate design, the applicant presented a design analysis which is documented in the technical report APP-MV50-S2C-020, Khanh Do, "Polar Crane Girder Top Plate Analysis". The stress analysis considers a total of 192 load combinations. Using ANSYS 11.0, a commercially available general purpose code, a 3D finite element model was constructed with the boundary conditions fixed at EL 100 ft. (i.e., at the grade level) for the purpose of determining the maximum stress intensities at the top plate. It was found that the maximum stress intensity [

] is well within the allowable stress intensity of 90 ksi (620.5 MPa) according to ASME NE-3221 in ASME Sec. III, Division 1, Subsection NE, Class MC, 2001 edition. Accordingly, a 1.75 inch thick plate is shown to be adequate to support the design-basis loadings, because the maximum stress intensity generated in the top plate is less than the allowable stress intensity the steel plate can offer. The staff reviewed the methodology of the analysis including model construction, boundary and loading conditions, material properties and applicable code; and determined that the design analysis is acceptable, and the use of a 1.75 inch thick top plate is in compliance with the ASME code requirements.

With respect to the new rail clip design with the bolted clips, the applicant presented an analysis in PR-08-5020/70587483, Roger Johnson, "Polar Crane Mechanical Calculations," Rev.1, March 5, 2010, to show that the proposed new design is capable of resisting a horizontal wheel load [ ] due to seismic conditions. With the help of MathCad Version 13 and GTSTRUDL 29.1 seismic analysis, it was found that the rated load [ ] governs the main hoist system, whose design meets NUREG-0554 requirements. The new clip design is given in Westinghouse drawing APP-MH01-V2-021 and Westinghouse drawing APP-MH01-V6-041 and is based on the adopted 30-inch wheel diameter. [

]

With the use of the clip plate [ ] with three slotted holes, the maximum bending stress

[ ] is less than the 81 ksi allowable specified by the code. The clip plate is bolted [ ] to the filler plate welded to the girder top plate [

] and the induced shear [ ] will be within the allowable shear stress of 21.6 ksi based on the AWS code (the ASME NOG-1 code allowable is 50% higher, i.e., 32.4 ksi). The staff reviewed the detailed calculations and found the new design as given in drawings APP-MH01-V2-021 and APP-MH01-V6-041 to be in compliance with the AWS and ASME codes; thus, it is acceptable to the staff.

The third item of design changes is concerned with the inward extension of 2-3/4 inches of the CV girder top plate. The reasoning, according to the submittal, is that the tolerance requirement of the polar crane rail is much tighter than the SCV girder. The staff requested the applicant to provide the basis to justify that the increase of 2.75 inches inward of the SCV girder top plate is quantitatively adequate to meet the tolerance requirement. In the response, the applicant provided ASME code, ASME-NOG-01-1998 (NOG-1) as the basis in the PC design documented in Westinghouse calculation APP-GW-GEE-513, DCP3513, "Steel Containment Vessel Girder and Polar Crane Rail Clip Design". The code NOG-1 specifies that the maximum and minimum span between the rails must be no greater or less than the nominal span of the rails +/- 3/8 in. This tolerance is tighter than the girder, based on construction experts, who specify the tolerance for the radius of the SCV girder to be +/- 1.95 inches. Because of this difference in tolerance requirements, the girder, including top plate, web and bottom plate, must be extended radially inward toward the center of the CV. Figure 5 in Westinghouse calculation APP-GW-GEE-513, DCP3513, "Steel Containment Vessel Girder and Polar Crane Rail Clip Design" provides the new design showing the tolerance stack-up and the required extension of the top plate. Figure 1, Note B and Figure 2 show the associated design changes. The staff reviewed the drawings and the new design, and found the design changes are in compliance with the ASME Code NOG-1, and thus, are acceptable.

The proposed design changes, including DCD markups and RAI responses, are acceptable. RAI DCP-CN-5-SEB-01 is resolved.

23.Q.4 Conclusion

The staff concludes that the proposed design changes are acceptable because the proposed changes will not adversely affect safety-related SSCs and the capability of the polar crane systems to perform their intended functions of heavy load lifting and transportation. The proposed design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable.

23.R Changes to the Reactor Vessel Support System

23.R.1 Description of Proposed Changes

In letters dated May 10, 2010 (ML101380275), and September 9, 2010 (ML102590074), Westinghouse proposed design changes related to the structural support system for the reactor vessel, which is a safety-related structure. These design changes involve the following modifications of the reactor vessel support system:

- Eliminate reliance on the CA04 structural module as part of the reactor pressure vessel (RPV) support system.
- Provide support boxes as RPV supports
- Change the anchor supports from a “bolted into embedded plates” configuration to a “anchored directly to primary shield wall concrete base via steel embedment plates” configuration.
- Increase the length of the RPV support boxes or legs.
- Install wear plates for the RPV bearing and tribological performance (i.e., to support RV and reduce frictional wear due to thermal expansion).

23.R.2 Regulatory Basis

Sections 3.8.3 and 5.4.10 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” address the function and acceptability of the RPV support structural box in the AP1000 standard design. Acceptability of the design change was evaluated for its conformance with the existing AP1000 licensing basis.

23.R.3 Technical Evaluation

In the AP1000 standard design, the RPV structural support system is designed to provide the necessary support for the RPV. The original anchorage design was bolting into embedded plates of the CA04 structural module. Design finalization analyses determined that the resulting stresses exceed the allowable stresses. As a result, the applicant proposed design changes in which the overstressed CA04 module is not used to support the RPV. Instead, the applicant has proposed a revised design where four support boxes are used to support the RPV. There are four support “boxes” or “legs” located at the bottom of the RPV’s cold leg nozzles. The support boxes are anchored directly to the primary shield wall concrete base via steel embedment plates. As stated in the proposed design changes, the design of the RPV associated support structures is consistent with the SSE design of Seismic Category I equipment. The RPV support boxes, including the new anchorage design, are analyzed to show that they can withstand an SSE event.

During the staff’s evaluation of the proposed design changes to the RPV associated support structures, it was determined that additional information was required from the applicant. The staff requested additional information related to: (1) A stress analysis of the proposed design including materials and design basis loading used to demonstrate that the maximum stresses incurred at the critical sections of the RPV support box structure are lower than those of the previous design, and that the new design is in compliance with the adopted code requirements; (2) Specifications of the wear plates, including material, lubricant used, geometric dimensions in size and shape, and performance requirements; and (3) An assurance of satisfactory tribological performance that uneven settlement on the plate's top surface will not deter lateral movement of the RV due to thermal expansion, and that the wear plate can endure the frictional wearing due to cyclic motion of the heavy RPV over its entire design life without loss of its intended design function.

In a letter dated September 9, 2010, the applicant provided sufficient information on the design analysis to support the staff's evaluation. The supporting information also included documentation on the material specifications of the self-lubricating bearings used for the wear plates.

With respect to the RPV support box design, the applicant stated that the design is in compliance with the design specification depicted in APP-SS30-Z0-001, "Design Specification" Revision 1. Thus, the support box design meets the requirements of the ASME B&PV Code, Sec. III, Subsection NF, 1998 Edition with addenda up to, and including, 2000. The support structure is classified as NF Class 1. Allowable stress limits of ASME NF-3220 for the Level A, B, C and D load combinations were calculated for the material used [ ] and it was found that the Level D loading condition governs the design. For design analysis, a three-dimensional finite element analysis (FEA) model was constructed using the general purpose FEA computer code ANSYS. Details of the methodology including the modeling techniques, boundary conditions and loading input are documented in the report APP-PH01-Z0C-007, "Finite Element Analysis of the RV Support Structure," Revision 0, May 26, 2010.The resulting stresses, including linearized primary membrane (P_m) and primary membrane plus bending ($P_m + P_b$) stress intensities, were computed from the FEA model. [ ] The resultant stress intensities for the controlling Level D load case [ ] were compared with the allowable stress intensities. The comparisons demonstrate that the Level D loading condition [ ] is less than the allowable. The staff performed audits of a supporting document on design specifications (APP-SS30-Z0-001) and a supporting document on finite element analysis (APP-PH01-Z0C-007). These audits confirmed that the information in the supporting documents was consistent with the applicant's proposed design changes. Therefore, the staff finds the proposed design changes to be acceptable because they satisfy the ASME Code requirements.

In its September 9, 2010, letter, the applicant stated that the RPV support bottom and side wear plates are specified as Lubron wear plates fabricated by Lubron Bearing Systems. [

] Detailed specifications of the wear plates used were provided by the applicant. Specifications of the bottom and side wear plates are provided in AP1000 RPV support design drawings APP-PH01-V2-211 (General Assembly) and APP-PH01-V2-212 (Component Details). During RPV installation, the bottom wear plate and interfacing thermal plate are assembled with a bluing process to insure a high degree of uniform contact (>75%) between mating surfaces. By design, these wear plates are good for up to 8,000 psi (55 MPa) bearing pressure and up to 1,100 degrees F. The actual bearing pressure for the AP1000 RPV support for the Level A service condition is approximately 2,300 psi (15.9 MPa), much less than the bearing capacity of 8,000 psi. The RPV support, its connection to the foundation, the foundation, and the wear plate connection to the support have all been designed for friction loads during normal plant heat-up and cool down thermal cycles. The staff's review also considered the potential for uneven settlement of the RPV during installation. Tier 1 Table 2.1.3-2 specifies the inspections, tests, analysis, and associated acceptance criteria (ITAAC) for the reactor system, which includes the RPV. Several ITAAC exist to address this issue, and include acceptance criteria that a report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions. The staff finds that the proposed design changes are acceptable in terms of structural and tribological performance.

23.R.4 Conclusion

The staff concludes that these proposed design changes are acceptable because the proposed changes will adversely affect neither safety-related SSCs, nor the capability of the RPV support boxes to perform their intended function of supporting the RPV while allowing free lateral movement of the four legs due to thermal expansion. These design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable.

23.S Changes To The Passive Containment Cooling System

23.S.1 Description of Proposed Changes

The applicant has proposed changes to the shield building from the Revision 15 DCD to address additional external hazards and to simplify construction. Westinghouse assessed the impact of these changes to the passive containment cooling system (PCS) in APP-GW-GLR-096, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analysis," Revision 1, and submitted the report to the NRC in a letter dated August 6, 2010 (ML102220583).

This report includes a description of the enhanced shield building design changes, a discussion of how these changes impact design basis test results, and evaluations of the limiting design basis accidents (DBA) and beyond design basis accidents (BDBA). Appendix A of APP-GW-

GLR-096 contains proposed DCD changes. The changes that impact Section 6.2.1, "Containment Functional Design"; 6.2.2, "Containment Heat Removal Systems"; and 6.2.3 "Shield Building Functional Design"; of NUREG-1793 are as follows:

- Update containment response to reflect results of an analysis that incorporates the enhanced shield building design.
- Lower the required reactor decay heat limit for air only cooling.
- Specify the long-term makeup rates the PCS must simultaneously provide to containment and the spent fuel pool (SFP) when the plant is refueling.

These proposed design changes also include revisions to several technical specifications (TS). In TS Table 3.3.2-1, "Engineered Safeguards Actuation System Instrumentation"; TS Table 3.3.5-1, "DAS Manual Controls"; and TS 3.6.7, "Passive Containment Cooling System (PCS) – Shutdown"; the Modes 5 and 6 applicability is revised to reflect the lower reactor decay heat limit for air only cooling mentioned above. In TS 3.7.9, "Fuel Storage Pool Makeup Water Sources," the notes associated with LCO 3.7.9, which list special plant conditions for each available makeup water source, are revised to account for changes to TS 3.6.7 regarding the availability of the passive containment cooling water storage tank (PCCWST) as a makeup water source. The cask loading pit (CLP) is added as a third makeup water source in addition to the PCCWST and the cask washdown pit. Two new surveillance requirements are also added to ensure readiness of water inventory from the PCCWST and the CLP, when needed.

Incorporation of changes proposed in Appendix A of APP-GW-GLR-096 in a future revision of the DCD will be tracked as part of **Confirmatory Item 23-17** except for proposed changes to Chapter 9. Proposed changes to Chapter 9 are evaluated in Section 9.1.3 of this SER.

23.S.2 Regulatory Basis

The following Commission regulations are related to the evaluation of the enhanced shield building:

- GDC 16, "Containment Design," as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require
- GDC 38, "Containment Heat Removal," as it relates to the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a loss-of-coolant accident (LOCA) and to maintain them at acceptably low levels
- GDC 50, "Containment Design Basis," as it relates to demonstrating sufficient margin in accident analysis
- 10 CFR 52.47(c)(2), as it relates to design certification testing in support of a passive plant design

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the PCS and the SFP design and operating information described in DCD Sections 6.2 and 9.1.3 respectively. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in by SRP Chapter 16.

23.S.3 Technical Evaluation

One of the safety-related functions of the passive containment cooling system (PCS) is to provide air flow over the outside of the containment vessel by means of a natural circulation air flow path. Air enters through inlets located at the top of the shield building and then travels downward into the annulus between the inside of the shield building and the air baffle. At the bottom of the baffle wall, the air turns 180 degrees into the riser annulus formed by the baffle and the outside of the containment vessel. Air continues past the top of the containment vessel and exits through the shield building chimney.

In the Revision 15 design, air is admitted into the top of the shield building through 15 large, uniformly distributed openings, with fixed louvers and screens to prevent weather and wildlife from entering the shield building. In the modified design, a steel structure containing 29 uniformly distributed openings with fixed louvers and screens is added around the top outside of the shield building. The flow area in the steel structure is slightly higher than the flow area through the louvered and screened inlet openings in the previous shield building design. Air enters through these openings, collects in the plenum created by the structure and outside of the shield building, and then enters the shield building through 236 inlet ducts. These shield building inlets have a much lower flow area than the louvered and screened inlet openings in the previous shield building design. The enhanced shield building chimney is shorter than the original design; therefore, the buoyant driving head for the PCS flow is expected to decrease. In the enhanced design, two heavy steel grates are added within the chimney region to protect the containment shell from external hazards, which will increase the discharge flow resistance and reduce the discharge flow area.

The NRC approved the functional design of the containment systems for the AP1000 as described in NUREG-1793. The agency based its approval on data from a number of test facilities that were specifically designed to model processes and phenomena of the AP1000 containment design following a major piping rupture within the containment building. These data were used for the qualification of the WGOTHIC computer code, which was then used for the safety analysis of the AP1000. The test series included heat and mass transfer tests, wind tunnel tests, water distribution tests, integral system tests of the PCS, and air flow path characterization tests. All of these tests were to be within the range of the physical conditions predicted for the enhanced shield building design except for the air flow path characterization test which modeled the previous shield building. The flow path pressure drop is expected to increase in the new design; therefore, the applicant repeated the air flow path characterization test on apparatus specific to the revised shield building. The results were used to confirm the local annulus loss coefficients used in the revised WGOTHIC model.

From the phenomena identification and ranking table that was developed for the AP1000 in WCAP-15613, "AP1000 PIRT and Scaling Assessment," issued February 2001, Westinghouse identified the high-ranked (most important) phenomena and processes for evaluation of the containment response following a LOCA or main steamline break (MSLB). These include condensation on the inside surface of the containment shell, conduction through the shell, and evaporation from the containment shell liquid film that flows down from the PCS water storage tank to the top outside of the containment building. Air flow through the containment shield building annulus is not listed with the high-ranked phenomena.

The staff agrees that air flow through the shield building need not be highly ranked following a LOCA or MSLB, because in these instances, the PCS will release water over the containment shell to provide evaporative cooling, and evaporative cooling is a much more significant heat removal mechanism than air flow convection. Analyses by Westinghouse and the staff illustrate this conclusion. APP-GW-GLR-096 describes the Westinghouse analysis based on the WGOTHIC evaluation model that was approved by the staff for the Revision 15 DCD analysis and subsequently modified to incorporate the proposed wet bulb temperature increase to 30 degrees C (86.1 degrees F) discussed in Section 6.2.1.1.1 of this amendment along with the pressurizer room changes discussed in Section 6.2.1.2 of this amendment.

To address the enhanced shield building design, Westinghouse further revised the model to include the addition of the PCS air inlet structure, reduction of the flow areas in the shield building inlet and exit, reduction of the shield building chimney height, and increase to the air flow path resistance. Westinghouse ran the design basis LOCA and MSLB events with the revised model and the results demonstrated that even though the natural circulation air flow decreased, the effect on containment pressure and temperature was insignificant. Additionally, the pressure at 24 hours following a large cold leg break LOCA was unchanged. The staff reviewed the detailed modeling changes during audits on April 21, 2010 and September 3, 2010. The subject of the audits was APP-SSAR-GSC-746, "Containment Response Analysis for the AP1000 Shield Building Design Change." The staff found the estimates on loss coefficients, which were made prior to completion of the air flow path characterization test, to be reasonable because they bounded predictions [ ] for a thick orifice. Furthermore, when pressure drops from the air flow path characterization test became available, these estimated loss coefficients were demonstrated to be conservatively high.

The staff performed a confirmatory analysis using the CONTAIN computer code with an AP1000 model developed by the staff during the DCD review. The shield building changes were incorporated and used to evaluate a large cold-leg break LOCA, which is the peak pressure DBA. The results were consistent with the Westinghouse evaluation; the PCS air flow decrease had a negligible impact on peak pressure, peak temperature, and containment pressure after 24 hours.

Westinghouse proposed changes to incorporate the results of its analysis into the DCD tables summarizing postulated accident values, DCD Tier 2, Tables 6.2.1.1-1 and 6.2.1.1-3. The changes to the LOCA and MSLB results were found acceptable because they are consistent with the enhanced shield building design. Changes made to the external pressure results are evaluated in section 23.W of this SER. **Confirmatory Item 23-17** does not include incorporation of any changes to external pressure parameters in DCD Tier 2, Table 6.2.1.1-3.

The natural circulation air flow is reduced in the enhanced shield building design; therefore, the amount of heat that can be removed during air only PCS operation will also be reduced. As a result, Westinghouse proposed lowering the reactor decay heat limit for air only PCS operation from 9 MWt to 6 MWt. APP-GE-GLR-096 describes the supporting WGOTHIC analysis demonstrating the containment pressure could be maintained below the design value of 59 psig for seven days with no PCS water release (and subsequent evaporative cooling) assuming an initial decay heat rate of 6 MWt.

The staff reviewed the analysis basis during its July 27, 2010, audit of APP-SSAR-GSC-749, "AP1000 Dry PCS Heat Removal Capability." The evaluation model was the previously described LOCA model modified to turn off the PCS water, replace the accident mass and energy forcing functions with decay heat input to the IRWST, and adjust the air flow path loss coefficients to reflect a 30% increase over the air flow path characterization test results. The staff finds the loss coefficient values reasonable because they are based on physical measurements with added margin to bound uncertainties in the test results.

The staff ran a confirmatory CONTAIN analysis using the previously described LOCA model modified for no PCS water flow and an initial decay heat rate of 6MWt.
The results were consistent with the Westinghouse evaluation; the air flow across the containment shell decreased and the containment pressure increased compared to the original shield building, but containment pressure remained below the design value for the run duration of seven days.

In an additional study, Westinghouse re-evaluated the containment pressure resulting from the BDBA of a prolonged loss of offsite power concurrent with a complete loss of the passive containment cooling water. For this unlikely event, the passive residual heat removal heat exchanger (PRHR HX) transfers reactor decay heat and the system sensible heat into the IRWST pool. When the water in the IRWST is heated to boiling, the steam that is released condenses on the containment internal structures. As the internal structures are heated, the containment pressure rises and heat is transferred through the containment shell to the air traveling up through the shield building annulus. Westinghouse performed the analysis with a "best estimate" WGOTHIC model. The results showed that even with the reduced natural circulation air flow associated with the enhanced containment building, it will take more than 24 hours for the containment pressure to reach the maximum pressure capability limit of 889 kPa (129 psig) defined in DCD Section 3.8.2. Therefore, the enhanced shield building meets the DCD Section 19.34.2.6 statement that, with air-only cooling, containment failure is predicted to occur more than 24 hours after accident initiation. The staff reviewed the analysis basis during its July 27, 2010, audit of APP-SSAR-GSC-749. The assumptions for the "best estimate" BDBA evaluation model, which included changes to initial temperatures, heat transfer coefficients, and loss coefficients, were found to be reasonable and consistent with evaluations of beyond design basis events.

The PCS has a design commitment to provide containment cooling and SFP makeup simultaneously from post-72 hours to seven days after DBA initiation at the minimum flow rates specified in DCD Tier 2, Table 6.2.2-1. Westinghouse recognized that the proposed reduction to the maximum reactor decay heat limit for air only containment cooling created a scenario whereby the currently specified minimum SFP supply of 35 gallons per minute (gpm) would not be adequate to maintain coverage of the fuel in the SFP. During refueling, when the full core is split such that the reactor decay heat is greater than 6 MWt and the spent fuel pool (SFP) decay heat is less than 7.2 MWt, the PCS water is reserved for containment cooling for the first 72 hours and cannot be used for SFP makeup until after this time. Westinghouse determined a DBA at this plant condition would require a minimum of 50 gpm SFP supply. Because the total PCS flow is limited, an increase in the SFP supply requires a reduction to the currently specified containment make-up rate of 100 gpm. The 100 gpm is based on containment cooling requirements post-72 hours after a limiting DBA occurring at full power. The necessary flow following a DBA during refueling will always be less than this because the reactor must be shut down for 100 hours prior to the start of refueling to provide time for the RCS to cool down and depressurize. Westinghouse reduced the required flow rate for containment cooling during refueling to 80 gpm. The supporting analysis, APP-SSAR-GSC-750, "WGOTHIC Validation of Post-72 Hour Containment Cooling Flow Rates for Accident Scenarios after Refueling" was audited by the staff on July 30, 2010.

The evaluation considered a loss of power event coincident with the start of refueling, modeled by adding decay heat representative of 100 hours after shutdown to the IRWST to represent the PRHR HX system. The model assumed full PCS flow for the first 72 hours and 80 gpm flow thereafter. The results demonstrated that the containment pressure remained well below the design limit of 59 psig for seven days. The staff found the modeling assumptions to be consistent with design basis analysis and the significant margin in the results provided further assurance that this is an acceptable change.

With respect to proposed changes to TS Sections 3.3.2, 3.3.5, 3.6.7 and 3.7.9 and their associated Bases, the staff finds these changes acceptable because they reflect the PCS and the SFP cooling system design and operating information described in DCD Sections 6.2 and 9.1.3, respectively. Incorporation of these TS changes in a future revision of the DCD will be tracked as part of **Confirmatory Item 23-18**.

23.S.4 Conclusion

Pending resolution of **Confirmatory Item 23-17 and Confirmatory Item 23-18**, the staff's review concludes that the design changes described above are acceptable and compliant with GDC 16, GDC 38, GDC 50 and 10 CFR 52.47(c)(2).

23.T Changes To The Main Control Room Emergency Habitability System

23.T.1 Description of Proposed Changes

By letters dated July 29, 2010 (ML102140343), and September 22, 2010 (ML102670162) Westinghouse Electric Company has proposed changes to the minimum amount of stored compressed air in the main control room emergency habitability system emergency air storage tanks. To increase the margin to the control room operator dose limits and to expand the site dispersion factors that are permitted for the AP1000 design, a passive filtration sub-system design was added to the MCR Emergency Habitability System (VES) to filter potential contaminated in-leakage. The sub-system incorporates an eductor which uses the VES compressed air flow to induce recirculation of MCR air through a filtration unit. The performance of the added sub-system allows for 15 cfm of unfiltered in-leakage while maintaining operator dose below 5 rem TEDE required by GDC19. With the addition of the sub-system, the VES provides a filtration unit to capture potential contaminated air that may leak into the MCR envelope. Westinghouse conducted testing of the passive filtration sub-system at the Westinghouse Waltz Mill facility. The testing was confirmatory testing to show the performance characteristics of the added passive filtration design and to collect data on the performance of the eductor itself. The passive filter train, utilizing the eductor as well as the technical specifications and ITAAC were reviewed and approved in Chapter 6.4 and 9.4.1 of this Safety Evaluation. However, after the Chapter 6 and Chapter 9 Safety Evaluations were issued, the applicant identified the need to adjust the technical specification on quantity of compressed air needed for 72 hours of continuous operation. The quantity of air needed to support 72 hours of operation increased because the pressure regulating valve minimum required operating inlet pressure needed to increase to ensure that the set outlet pressure could be maintained during the duration of system operation.

The minimum amount of required stored compressed air is changed from 314,132 scf to 327,574 scf. Filling the tanks with 327,574 scf of air will ensure that the tanks are capable of providing 70 scfm of air for 72 hours. Operability is determined based on the amount of compressed air stored in the tanks as determined from tank pressure and storage room temperature. The new volume of air is provided in both Tier 1 and Tier 2, including the Technical Specifications. The relationship between tank pressure, room temperature and volume is provided in the Technical Specification Bases.

These proposed design changes also modified the instrumentation representation in DCD Figure 6.4-2 (sheet 2 of 2) to better represent the instrumentation used in the design. The figure changed a flow instrument from an orifice plate with a differential pressure sensor to a Thermal Dispersion Mass Flow transmitter.

23.T.2 Regulatory Basis

The applicable regulations for the control room habitability system aspects of these design changes are detailed in the SRP 6.4, “Control Room Habitability System.” For the changes proposed by the applicant, the following are relevant.

- 10 CFR Part 50, Appendix A, GDC 19, “Control Room.”
- 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria.

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.” The staff’s evaluation of the proposed changes focused primarily on confirming that the changes to the GTS and Bases reflect the VES design and operating information described in DCD Section 6.4. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Chapter 16, “Technical Specifications – Revision 3, March 2010.” Acceptability was based on conformance with the guidance specified in SRP Chapter 16.

23.T.3 Technical Evaluation

The proposed design changes were reviewed for compliance with the applicable regulations. This DCP raises the minimum amount of stored compressed air in the VES emergency air storage tanks to ensure the supply of 72 hours of air at the maximum flow rate of 70 scfm. The mission time for the passive safety-related system is 72 hours based on the Commission’s policy on passive systems. The design basis requires that 65±5 scfm flow be provided for all Design Basis Accidents. As a result, there needs to be enough air to provide 72 hours of flow at the maximum flow of 70 scfm.

The change is necessary to accommodate a 200 psig pressure regulator minimum inlet pressure instead of 100 psig prior to the passive filter train. This is done to ensure that downstream pressure and flow are maintained within the required tolerance. The applicant has demonstrated that the new volume of air is adequate to accomplish the design basis functions.

The staff audited Westinghouse “VES Minimum Pressure Calculations, APP-VES-M3C-005, Rev1.” The calculation determines the amount of breathable air required during 72 hours of VES operation. Basic heat transfer equations and air properties were used to determine heat transfer coefficients and transient heat loss from concrete to outside air and the tank package. Cooldown analysis for station blackout during extreme winter weather were used to determine VES tanks pressure decline due to temperature cooldown and air delivery to MCR. Assuming cold conditions is conservative because it minimizes the volumetric flow. The staff has found that the calculations demonstrate there is an adequate amount of gas to maintain pressure above 200 psig at the inlet of the pressure regulator. The calculations showed that for the limiting case, there was not much margin. However, because the calculations were done for the limiting cases, small margin is acceptable.

In Chapter 6 of this SER, the staff accepted the passive filter train. The staff came to this conclusion, in part, based on there being adequate ITAAC and Technical Specifications to demonstrate that 65±5 scfm flow is provided from the canisters to the eductor and that 600 scfm of control room air would be drawn through the filter train. The COL holder would need to demonstrate prior to plant operation and periodically thereafter that the system would accomplish the design basis functions. Because the passive eductor-driven filter train was new to the nuclear industry, the staff requested that testing be done to prove this design concept was capable of being operated successfully. Prior to the issuance of the Chapter 6 Safety Evaluation, the applicant completed testing to demonstrate that the system was capable of meeting the performance requirements.

When these proposed design changes were submitted by Westinghouse, the staff wanted to make sure the testing still demonstrated the system was capable of being operated successfully. As such, the staff also audited Westinghouse "AP1000 VES Air Filtration Test Specification, TS-SEE-111-09-03." The testing performed by the applicant had three objectives. The first was to demonstrate 60 scfm is capable of inducing 600 scfm, the second was to determine whether a feed flow rate higher than the design duct flow rate would damage system components, and the third was to demonstrate the system can be operated below maximum allowable noise levels defined in "NUREG-0700." The maximum allowable noise is 65 dB(A).

The staff observed that the testing did demonstrate the system could be built with the necessary performance characteristics. The staff also observed that the test results showed that the combined flow is sensitive to back pressure, feed pressure and feed flow. For example, for the same feed pressure, an increase in backpressure by 0.5 inch of water will reduce combined flow by approximately 100 scfm. Additionally, at the same feed flow an increase in the feed pressure by 5 psi will reduce combined flow by approximately 100 scfm. As a result, the staff notes that the parameters influencing the induced flow, for example line losses between the regulator and the eductor, will need to be carefully controlled by the COL holder and that the required 600 scfm induced flow may not be satisfied at all conditions. The COL holder has the responsibility to ensure that at least 600 scfm will be induced by a feed flow rate of at least 60 scfm.

The two safety-related flow rates will be demonstrated by ITAAC. Additionally, Tier 1 requires the following be demonstrated:

- The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.
- The airflow rate from VES is at least 60 scfm and not more than 70 scfm.

- The system provides a passive recirculation flow of MCR air to maintain main control room dose rates below an acceptable level during VES operation.
- The air flow rate at the outlet of the MCR passive filtration system is at least 600 cfm greater than the flow measured by VES-FT003A/B.
- The noise at the operator station is limited to 65 dB(A).

The staff has concluded that with the ITAAC to demonstrate the capacity of the system design the system will meet the requirements of GDC 19. Additionally, the ITAAC are sufficient to show the as-built plant will function and, as a result, 10 CFR 52.47(b)(1) is satisfied.

With regard to the change in instrumentation, the staff finds that either type of instrument can acceptably measure flow. The instrument is safety-related and subject to the quality assurance requirements. As a result, the staff finds the instrumentation change acceptable as well.

With respect to proposed changes to TS 3.7.6 and its associated bases, the applicant stated that both tank pressure and room temperature are used to determine the acceptable minimum storage capacity of the air tanks in term of standard cubic feet (scf). The acceptance criteria are presented in new Figures B 3.7.6-1 and B 3.7.6-2 for use in the verification of the minimum storage capacity of the air tanks specified in Required Action D.1 and SR 3.7.6.2 respectively. The staff finds these changes acceptable because they reflect the VES design and operating information described in DCD Section 6.4,

23.T.4 Conclusion

The changes proposed by the applicant are expected to be included in the next revision of the DCD. Incorporation of these changes in a future revision of the DCD will be tracked as part of **Confirmatory Item 23-19**. The staff finds that there will be an adequate amount of air for the extreme winter conditions for 72 hours of VES operation and to maintain pressure above 200 psig at the inlet of the pressure regulator. The staff has concluded that the proposed design change complies with GDC 19 and the acceptance criteria specified in Section 6.4 of the SRP. Lastly, the staff finds that the ITAAC are sufficient to demonstrate that the system when built will accomplish the safety function.

23.U Changes to Main Steam Isolation Valve Subcompartment

23.U.1 Description of Proposed Changes

In letters dated, August 12, 2010 (ML102290205), and September 30, 2010 (ML102780270), Westinghouse proposed design changes that make the vent paths associated with the main steam isolation valve subcompartments larger. The applicant also proposed to change the content of the pipe hazards analysis report described in Tier 2 DCD Section 3.6.2.5 and to remove tables in Chapter 6.2 of the FSAR that report mass and energy releases and compartment differential pressures outside of containment.

SRP 6.2.1.2 describes subcompartment analyses inside of containment. Additionally, SRP 6.2.1.4 describes mass and energy release from secondary side breaks inside of containment. In Chapter 3 of the SRP, high energy pipe hazards outside of containment are described. In the certified design, subcompartment analyses and mass and energy data for pipe breaks outside of containment are included in Chapter 6. This is not typical and not consistent with either the SRP or RG 1.206. Additionally, there is a COL holder item in FSAR Chapter 3 that requires the COL to perform a pipe hazards analysis. As a result, there is some confusion in the certified design. In FSAR section 6.2.1.2 entitled, “Containment Subcompartment Analysis,” there are analyses for subcompartments outside of containment. Additionally, in FSAR Chapter 3 there is the requirement for another pipe hazards analysis to be performed by the COL holder.

The applicant identified that the rupture of a feedwater pipe may produce more limiting results than the main steam line break that was described in the FSAR; therefore, larger vent paths were needed in the main steam valve rooms.

The applicant has proposed to increase the size of the vent paths in the roof of the Auxiliary Building to provide larger vent paths for high-energy hazards. The proposed design changes also modify the structural attachments at the Auxiliary Building roof. The applicant identified a pipe hazard that releases more energy than is currently considered in the hazards analysis. The vent paths are described in Chapter 3. Specifically the applicant has made changes in the doghouse structures on the roof of the Auxiliary Building. Each doghouse structure has a blowout panel and a louvered vent. The blowout panel is 10feet by 10 feet and the louvered vent is 6 feet by 6 feet.

The applicant has clarified FSAR Chapter 3.6.2.5 to explicitly include subcompartment pressurization of these compartments outside of containment in the pipe hazards analysis report. To eliminate ambiguity in the FSAR, the applicant has removed the subcompartment analyses and mass and energy tables for pipe hazards outside of containment from Chapter 6 of the FSAR.

23.U.2 Regulatory Basis

For the Chapter 6 changes, the applicable regulations for the containment systems aspects of this design change are detailed in the SRP 6.2.1.2, “Subcompartment Analysis,” and include the following:

- 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases.”
- 10 CFR Part 50, Appendix A, GDC 50, "Containment Design Basis."

However, because the subject subcompartments are not inside the containment, GDC 50 is not applicable.

For the Chapter 3 changes, the applicable regulations for the protection against pipe rupture are detailed in SRP 3.6.1, “Plant Design For Protection Against Postulated Piping Failures In Fluid Systems Outside Containment;” 3.6.2, “Determination Of Rupture Locations And Dynamic Effects Associated With The Postulated Rupture Of Piping;” and in Branch Technical Position (BTP) 3-3, “Protection Against Postulated Piping Failures In Fluid Systems Outside Containment.”

23.U.3 Technical Evaluation

The proposed design changes were reviewed for compliance with the General Design Criterion (GDC) 4, which requires that structures withstand the effects of high energy hazards. The design of internal compartments must accommodate the effects of, and be compatible with, the environmental conditions associated with postulated accidents or high energy hazards. The internal compartments shall be appropriately protected against dynamic effects.

The proposed design change of increasing the size of the vents improves the room’s or compartment’s ability to relieve pressure if a high energy pipe fails in the compartment. Confirmation that the final as-built design is in compliance with the requirements is the responsibility of the COL holder. This responsibility is clearly described in the COL holder item in FSAR Chapter 3.6.4.1. FSAR Chapter 3.6.2.5 outlines the content of the pipe hazards analysis report. The applicant proposed to change FSAR Chapter 3.6.2.5 to include the following statement:

> Evaluate compartment pressurization in the break exclusion zones in the vicinity of containment penetrations due to 1.0 square foot breaks in the main steam and feedwater lines.

The proposed change clarifies that the compartment pressurization in the break exclusion zones needs to be evaluated for 1.0 square foot breaks in the main steam and feedwater lines. The proposed design change is acceptable because it improves the ability of the facility to withstand high-energy pipe breaks. The change to FSAR Chapter 3.6.2.5 is acceptable because it clarifies the content of the high-energy line break analysis necessary. A COL holder item for high-energy line break analysis is an appropriate approach to demonstrate compliance with GDC 4 because much of the pipe hazards analysis is site-specific in nature.

The staff also finds the removal of the peak differential pressures and mass and energy tables from FSAR Chapter 6.2 acceptable. FSAR Chapter 6.2 is dedicated solely to containment issues. Neither the SRP nor RG 1.206 recommends these items be included in Chapter 6. Outside containment subcompartments are much more appropriate in Chapter 3 under the COL holder item. As a result, the staff finds this change acceptable as well.

The DCD Tier 2 Section 3.8.4.3.1.4 specifies that a differential pressure of 6 psi (41 kPa) be the design limit for subcompartment pressurization in the MSIV rooms. The design finalization shows that a main feedwater line pipe rupture without adequate venting will cause the subcompartment pressurization to exceed the 6 psi design pressure limit. Accordingly, Westinghouse decided to enlarge the venting area of the roof of the Auxiliary Building in order to meet the requirements of the DCD. Westinghouse provided the modified steam vent design with the proposed design changes. In addition, Westinghouse committed to finalize the design of the two doghouse structures on the roof of the Auxiliary Building in accordance with the design procedure of the critical sections as defined in DCD Section 3.8, its Appendices, and in TR 57.

During the staff's evaluation of these proposed design changes, the following additional information was requested from the applicant:

- Engineering drawings of the doghouse structure at the roof, detailing the venting assemblies, blow-out panels, louvered vents, and connections to the Auxiliary Building roof.
- Stress evaluation on the structures, including applied design-basis loading and the resulting maximum stress.
- The basis, including acceptance criteria and performance requirements, upon which the proposed new design is determined to be acceptable.

The applicant's September 30, 2010, response stated that the design will be finalized using the same methodology, specifications for load combinations, and safety factors that are specified in DCD Section 3.8. Because the final design will be consistent with the design procedure of the critical sections as defined in DCD Section 3.8, its Appendices, and in TR 57, the staff finds the change to the vent area to be acceptable.

23.U.4 Conclusion

On the basis of its review of the Chapters 3.6.2 and 6.2, the staff finds the proposed design changes acceptable. The approach chosen by the applicant will ensure compliance with GDC-4. The changes proposed by the applicant are expected to be included in the next revision of the DCD.

Incorporation of the updated Tier 2 Chapter 6.2 information into DCD revision 18 is identified as **Confirmatory Item 23-20.** Incorporation of the updated Tier 2 Chapter 3.6.2.5 information into DCD revision 18 is identified as **Confirmatory Item 23-21.**

The staff's review concludes that the proposed design changes are acceptable because they will not adversely affect safety-related SSCs or the capability of the MSIV depressurization subcompartments to perform their intended functions of pressure relief for the postulated high-energy line break.

23.V Changes to the Component Cooling Water System

23.V.1 Description of Proposed Changes

A leak or tube rupture in the Reactor Coolant Pump (RCP) external heat exchanger (EHX) would not result in over-pressurization of piping outside containment, since the pressure in the Component Cooling Water System (CCS) is controlled by the system's atmospheric surge tank. Such an event could result in a reactor trip and a nonisolable flow path from the RCS through the CCS piping and the surge tank vent to the turbine building (Interfacing System LOCA). The applicant has proposed design changes intended to add a safety related means to isolate this potential flow path. The proposed design changes would provide automatic, safety-related isolation of a LOCA caused by the rupture of one of the RCP EHX tubes. This automatic isolation would prevent discharge of reactor coolant to the turbine building through the CCS surge tank vent and would limit off-site doses to values below those already found acceptable in the event of a small break LOCA, as stipulated in 10 CFR 50.34a.

In letters dated July 28, 2010 (ML102110187); September 3, 2010 (ML102500462); September 29, 2010 (ML102780269); and October, 18, 2010 (ML102930085), Westinghouse proposed design changes to the component cooling water system (CCS). The proposed design changes relate to the following:

- Instrumentation and Controls – Containment isolation:
  The applicant proposed to modify the closure logic for CCS motor-operated containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 to add a requirement to close on generation of the reactor coolant pump (RCP) bearing water high temperature pump trip signal. This modification would add a new isolation signal to DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetration and Isolation Valves."

  A closure signal to the component cooling system containment isolation valves is derived from a coincidence of two of the four divisions of high reactor coolant pump (RCP) bearing water temperature for any reactor coolant pump. The high temperature setpoint and dynamic compensation are the same as used in the high reactor coolant pump bearing water temperature reactor coolant pump trip (Subsection 7.3.1.2.5, Condition 6), but with the inclusion of preset time delay.

- Instrumentation and Controls – CCS:
  The applicant proposed to remove the automatic isolation of the CCS RCP Heat Exchanger (HX) outlet isolation valves (CCS-PL-V256A/B/C/D) to close on high deviation between inlet and outlet flows. Simultaneous flow deviations in both the inlet and outlet lines would generate a flow deviation alarm and not isolate these valves. This alarm would be indicative of reactor coolant system (RCS) leak conditions and would alert plant operators to close the valve on the cooling water outlet line on each RCP to prevent reactor coolant flow throughout the CCS. Both the flow signals and the isolation valves are nonsafety-related.

- CCS:
  The applicant proposed to install a 4 inch x 6 inch safety-class relief valve, designated CCS-PL-V270 and CCS-PL-V271,respectively, on each of the 10-inch CCS supply and return lines (total of two relief valves), just inside the innermost containment isolation valves (CCS-PL-V201 and CCS-PL-V207). In addition, the applicant proposed to change the safety class of the section of line between the innermost containment isolation valves and the Appendix J test valves (CCS-PL-V214 and CCS-PL-V216) from Class ‘0’ to Class ‘C’ to ensure that the relief valves are installed as ASME safety-class piping.

- Technical Specifications:
  The applicant proposed to add an RCP bearing water temperature high trip function to Technical SpecificationsTable 3.3.2-1, “Engineered Safeguards Actuation System Instrumentation,” for closure of the CCS containment isolation valves.

These proposed design changes include revisions to the Tier 2 DCD Sections 3.2, 3.9, 3.11, 5.2, 6.2, 7.2, 7.3, 9.2.2, and 16.

23.V.2 Regulatory Basis

- Instrumentation and Controls – Containment isolation
  The regulatory basis for evaluating the changes to the AP1000 closure logic for CCS motor-operated containment isolation valves is documented in Chapter 7 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.” Review of these proposed design changes is also based on the requirements in 10 CFR 50.55a(h), 10 CFR 52.47, and General Design Criteria (GDC) 20 and 21 of 10 CFR Part 50, Appendix A.

  The applicable regulations for the containment systems aspects of this design change are detailed in the SRP 6.2.4, “Containment Isolation System,” and include the following:

  - GDC 16 requires that the containment isolation system allow the normal or emergency passage of fluids through the containment boundary while preserving the capability of the boundary to prevent or limit the escape of fission products from postulated accidents.

  - GDC 54, "Piping Systems Penetrating Containment," requires that the containment isolation system valves in piping systems that penetrate the containment be designed to close reliably under accident conditions and prevent the uncontrolled release of radioactive materials.

- Instrumentation and Controls – CCS:
  The regulatory basis for evaluating the proposed instrumentation and controls changes is documented in Chapter 7 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." Both the flow signals and the isolation valves for the CCS RCP HX outlet isolation valves are nonsafety-related and are not relied upon to perform any safety functions. However, they are part of the CCS, which is considered to be important to safety because it supports the normal (defense-in-depth) capability of transferring heat from various plant components and also removing reactor system and spent fuel decay heat. Reviews of the changes are based on meeting the relevant requirements of 10 CFR 50.55a(h), 10 CFR 52.47, and GDC 13 and 19 of 10 CFR Part 50, Appendix A.

- CCS:
  The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to regulatory treatment of nonsafety systems (RTNSS) in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes could not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of inspections, tests, analyses and acceptance criteria (ITAAC), test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Section 9.2.2, "Reactor Auxiliary Cooling Water System – Revision 4, March 2007," as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified by SRP Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

  The following regulatory guidance is also applicable to the proposed design changes:

  - RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"

  - RG 1.29, "Seismic Design Classification"

  - SECY 90-016, Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. This could result in a core melt accident with an energetic release outside the containment building, potentially causing a significant offsite radiation release.

- Technical Specifications:
  The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the CCS isolation function design and operating information described in DCD Sections 5.2, 7.3, and 9.2 respectively. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in SRP Chapter 16.

23. V.3 Technical Evaluation

During the staff's evaluation of these proposed design changes, additional information was requested from the applicant. In a letter dated September 3, 2010, the applicant responded to the request for additional information. The staff reviewed the applicant's responses and came to the following conclusions.

- The staff determined that the CCS piping system is adequately protected from over-pressurization due to a postulated RCP external heat exchanger tube rupture with the addition of two new relief valves, one near each CCS containment penetration. Both relief valves are designed to ASME Section III, Code Class 3 and have been added to a markup of Table 3.9-16, "Valve In-service Test Requirements," with a test frequency of at least once per 10 years. This will be tracked under **Confirmatory Item 23-22.** The failure of a spring-operated safety valve to open on a high pressure condition is excluded as a single active failure; however, two relief valves will see the over-pressurization event since there is no check valve between the cooling water line serving the reactor coolant drain tank. Since each valve has sufficient capacity to prevent system overpressure for the largest expected discharge from the heat exchanger tube rupture event, the staff finds the CCS piping system is adequately protected from over-pressurization. Therefore, the staff finds the proposed design changes to be acceptable.

- The proposed design changes were reviewed for compliance with GDC 16 and GDC 54 pertaining to containment isolation and containment integrity. The proposed design changes modify the closure logic for the motor operated CCS containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 by adding a requirement to close on the generation of a reactor coolant pump bearing water high temperature signal. This modification adds an additional isolation signal to these valves, and will be shown in FSAR Tier 2, Table 6.2.3-1, “Containment Mechanical Penetration and Isolation Valves.” This will be tracked as **Confirmatory Item 23-23**.

Containment isolation valves are required to close reliably under accident conditions and maintain their integrity to prevent the uncontrolled release of radioactive materials from containment. Reactor coolant pump bearing high water temperature would result from a rupture of one of the U-tubes in the RCP-EHX which is cooled on the shell side by the CCS. The tube rupture would result in both increased temperature and increased pressure in the CCS. Both GDC 16 and 54 require the containment isolation valves to close to prevent the uncontrolled release of radioactive materials from containment. Therefore the containment isolation valves must operate at the pressure and temperature conditions in the CCS generated by the RCS tube rupture. Additionally, the containment isolation valves must maintain their integrity and remain leak tight when closed.

Prior to a rupture, the tube side temperature and pressure in the RCP-EHX are approximately 160 °F and 2250 psia, whereas the temperature and pressure on the shell side of the heat exchanger are approximately 98 °F and 115 psia. The CCS is a 200 psig system, including the containment isolation valves and their included piping. Following a tube rupture the CCS pressure and temperature in the containment would increase significantly. The following conditions must be satisfied:

- The 200 psig design containment isolation valves CCS PL-V207, -V208, -V200, and –V201 must be able to close against the increased pressure due to the RCP-EHX tube rupture.

- The 200 psig design containment isolation check valve CCS-PL-V201 must maintain its integrity at the increased pressure and temperature due to the RCS-EHX tube rupture and remain closed.

- The new safety-class relief valves, CCS-PL-V270 and CCS-PL-V271, must operate under the conditions of flashing fluid at the maximum fluid temperature produced by the RCP-EHX tube rupture event.

The applicant proposed to add a single safety-class relief valve on each of the 10 inch CCS supply and return lines respectively just inside the innermost containment isolation valves, CCS-PL-V201 and CCS-PL-V207. The safety-class relief valves, with a set pressure of 200 psig, are intended to limit the CCS pressure at the containment penetrations to the design conditions for a 200 psig system.

The applicant performed an analysis, AP1000/ANSALDO RELAP calculation APP-CCS-M3C-164, rev 0, September 24, 2010, to simulate the occurrence of a postulated double ended tube break in the RCP-EHX in order to evaluate CCS transient and steady state conditions of temperature and pressure at the relief and containment isolation valves.

The results of the analysis for case 1, which postulates both safety class relief valves providing overpressure protection and opening at the set pressure of 200 psig, demonstrate the following:

- CCS-PL-V201 reaches a maximum pressure of 215 psia, a maximum temperature of approximately 380 °F, and reaches a steady state condition of approximately 180 psia and 372 °F at 1000 seconds.

- CCS-PL-V207 reaches a maximum pressure of 216 psia, a maximum temperature of approximately 380 °F, and reaches a steady state condition of approximately 180 psia and 373 °F at 1000 seconds.

The results of the analysis for case 2, which postulates a single relief valve, V270, opening upon reaching the set pressure of 200 psig, and a single failure of relief valve V271 to open, demonstrate similar results of temperature and pressure, but reach steady state conditions at 400 seconds.

The AP1000 piping class sheets and standard details, APP-PLO2-Z0-001, rev 5, p. 120, provide the design pressure and temperature envelope for class JCB and JCC piping and the CCS containment isolation and safety-related relief valves, summarized as follows;

32-100 °F	150 °F	300 °F
285 psig	270 psig	230 psig

Except for the CCS temperature of ~380 °F, the transient and steady state conditions are well within the pressure and temperature design envelope of the containment isolation valves and piping. The operating conditions during this transient will be added to the valve and piping data sheets, AP1000 Piping Class Sheets and Standard Details, APP-PLO2-Z0-001, rev. 5, and will be tracked as **Confirmatory Item 23-24**.

A RELAP calculation by staff confirmed the above transient results from the applicant's analysis, which was submitted in a letter dated October 18, 2010. The staff concludes that the containment isolation valves would be able to close at these pressure and temperature conditions, and maintain their integrity following a tube rupture in the RCP-EHX.

Each of the four CCS containment isolation valves will be explicitly identified in FSAR Tier 2, Table 3.9-12 (Sheet 1 of 7) as having both containment isolation and accident mitigation functions. The new safety-related relief valves, V270 and V271, will be explicitly identified as having an accident mitigation function. The conditions applicable for accident mitigation will be added to the valve data sheets as notes pertaining to the specific fluid temperature and pressure conditions for which the valves must remain operable and intact. This accident mitigation function and its associated operating conditions will be added to the valve and piping data sheets, AP1000 Piping Class Sheets and Standard Details, APP-PLO2-Z0-001, Rev. 5. These additions will be tracked as **Confirmatory Item 23-24**.

RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” recommends that portions of cooling water systems important to safety and which are designed for the functioning of components important to safety be designed to Quality Group C standards. For piping and valves the applicable standard is ASME Section III, class 3.

The applicant proposed to install two additional relief valves, CCS-PL-V270 and CCS-PL-V271, one on each of the 10-inch CCS supply and return lines, just inside the innermost containment isolation valves, CCS-PL-V201 and CCS-PL-V207, respectively. These relief valves are intended to limit the CCS pressure at the containment penetrations to approximately 200 psig. Since these relief valves are provided to protect the safety-related CCS containment isolation valves and piping, which are designed to Quality Group B standards, they will be designed to Quality Group C criteria, or ASME Section III, class 3. The staff has evaluated this designation, which will be added to the FSAR Tier 2, Table 3.2-3, “AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment,” and concludes that it meets the recommendations of RG 1.26.

Systems, structures and components of a nuclear power plant designated as Seismic Category I and designed to withstand the effects of an SSE and remain functional include components affecting the safety-related function of the primary containment, per RG 1.29. The relief valves, CCS-PL-V270 and -V271, are provided to protect the operability and long-term integrity of the CCS 10-inch containment penetrations. Therefore they will be designed to Seismic Category I criteria. The staff has evaluated this designation and concurs that it meets the guidance of RG 1.29. Adding the Quality Group C and Seismic Category I criteria to CCS-PL-V270 and CCS-PL-V271 to the AP1000 DCD, Table 3.2-3 and Table 3.9-12, “List of ASME Class 1, 2, and 3 Active Valves” will be tracked as **Confirmatory Item 23-25**.

For advanced reactor design, the staff stated its position regarding intersystem LOCA (ISLOCA) protection in SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements.” The staff stated that ALWR designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to full RCS pressure.

The CCS is a low-pressure system design with a URS below RCS operating pressure. Overpressurization of the CCS could occur following a tube rupture in the RCP-EHX. A relief valve has been added to each of the CCS supply and return headers inside the containment isolation valves in containment to prevent overpressurizing the CCS. The adequacy of pressure relief in protecting the operability and integrity of the containment isolation of CCS has been addressed above and found acceptable. Therefore, the staff finds CCS overpressure protection by the two safety-related relief valves meets the intention of the guidance for ISLOCA.

- Since DCD Tier 1 testing of the CCS-PL-V200 and V201 valves is adequately described in Table 2.2.1-1, the staff determined no other Tier 1 changes or ITAAC are required. DCD Tier 2, Section 14.2.9.2.5 adequately addresses CCS testing, which includes proper operations of controls, instrumentation, actuation signals and interlocks. Furthermore, DCD Tier 2, Section 14.2.9.1.10, “Containment Isolation and Leak Rate Testing,” addresses testing for proper operation of safety-related containment isolation valves listed in Table 6.2.3-1. Table 6.2.3-1 was modified to include the high bearing temperature valve closures as part of DCP 72, and will be tracked as **Confirmatory Item 23-26**.

  The proposed revision to DCD Section 9.2.2.4.5.2 provides an adequate description of the automatic isolation function and will be tracked as **Confirmatory Item 23-27**.

  The applicant proposed to modify the closure logic for CCS motor-operated containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 to add a requirement to close on generation of the reactor coolant pump (RCP) bearing water high temperature pump trip signal. If a tube break occurs at normal (higher) RCS temperatures, the RTDs used for the automatic isolation function sense the temperature of the reactor coolant flowing through the collection header almost immediately after the break initiates. However, for an idled RCP with the RCS at low temperatures, such as near 93 °C (200 °F), RCS flow is much less (as low as 5%) than the flow rates seen at 100% shaft speed. Even with reactor coolant temperature near 93 °C (200 ^{o}F), the RTDs will still sense the reactor coolant temperature within minutes and produce the automatic RCP bearing water high temperature trip, which closes the CCS containment isolation valves. Therefore, the staff’s review finds the applicant’s design change to using reactor coolant temperature instruments to initiate CCS containment isolation acceptable.

- The change to add Modes 3 and 4 to Table 3.3.2-1 of the Technical Specifications is acceptable to the staff since it envelops the possible Modes within which this event is postulated to occur. The staff had additional questions concerning how the applicant would effect this trip in Modes 3 and 4 since the temperature differential between the RCPs and CCS may be so small that a trip signal would be delayed longer than necessary to effectively isolate the system. In its response, the applicant provided further details of the sensor arrangement and flow path of coolant for this event, should it occur in Modes 3 or 4. The staff found this discussion acceptable because it describes how the trip signal will be generated in sufficient time to isolate the system, without the need for permissive or interlocked trip signals. The addition of this information to the bases of TS 3.3.2 and inclusion of this trip function in Table 3.3.2-1 is **Confirmatory Item 23-28**.

- The applicant proposed to add a signal to Functional Diagram 7.2-1 (Sheet 5 of 20) in AP1000 DCD Tier 2 to close the component cooling system containment isolation valves. This closing signal is derived from the existing 2-out-of-4 control logic with bypass capacity of the high RCP bearing water temperature for any reactor coolant pump. The high temperature setpoint and dynamic compensation for this new closing signal are the same as those used for RCP trips based on the high RCP bearing water temperature, but it will be implemented with a preset time delay. The newly added logic provides safety-related protection system functions and also meets the reliability and test requirements for the safety protection system as required by GDC 20 and GDC 21. Hence, the staff concludes that this change is acceptable. Incorporation of changes to Functional Diagram 7.2-1 in a future revision of the DCD is being tracked as **Confirmatory Item 23-29**.

  The applicant has also proposed to remove the automatic control function to isolate the CCS RCP HX outlet isolation valves on a high delta-flow between the HX inlet and outlet flow. The high delta-flow between the HX inlet and outlet cooling water lines now will only generate a flow deviation alarm in the MCR. The original automatic isolation function is provided by the plant control system (PLS) and non-Class 1E sensors and instrumentation. This nonsafety-related isolation control function is redundant to and is replaced by the automatic closure of the CCS containment isolation valves based on the high RCP bearing water temperature produced through the safety-related PMS using safety-related RCP RTD sensors. In addition, a new alarm for the high delta-flow between the CCS RCP HX inlet and outlet cooling water lines is now provided in the MCR, and also the remote manual operation of CCS RCP HX outlet isolation valves is retained. If annunciated in the MCR, the new alarm would alert plant operators to close the RCP HX outlet isolation valve remotely in the MCR on the cooling water outlet line to prevent reactor coolant flow throughout the CCS. The staff finds that the above changes meet relevant criteria as required in 10 CFR 50.55a(h) and 10 CFR Part 50, Appendix A GDC 13 and 19. The staff therefore concludes that these changes are acceptable.

23.V.4 Conclusion

The staff's review concludes that these proposed design changes are acceptable because they will not adversely affect safety-related structures, systems, and components (SSCs); and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. Adequate overpressure protection of the CCS is provided with ASME Code relief valves during a postulated RCP external heat exchanger tube rupture. An automatic signal will be generated to the associated containment isolation valves in the event of a postulated RCP external heat exchanger tube rupture, as shown in Technical Specifications Table 3.3.2-1.

On the basis of its review of the containment isolation design aspect of the proposed CCS overpressure protection design change, the staff concludes that the design complies with the acceptance criteria in Section 6.2.4 of the SRP, including: GDC 16, "Containment Design"; GDC 54, "Piping Systems Penetrating Containment"; RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants”; and RG 1.29, “Seismic Design Classification.”

The staff also finds the design change to provide CCS overpressure protection by the two safety related relief valves meets the intention of the guidance for ISLOCA found in SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements”.

As described in the technical evaluation section above, the NRC staff is tracking nine confirmatory items related to these proposed design changes. The confirmatory items will be resolved upon NRC staff confirmation that Westinghouse has properly included these proposed design changes in the DCD.

Two new relief valves, one near each CCS containment penetration, have been added to a markup of Table 3.9-16, “Valve In-service Test Requirements,” will be tracked under **Confirmatory Item 23-22**.

The additional containment isolation signal on high RCP high bearing water temperature to valves V200, V207, and V208, will be shown in DCD Tier 2, Table 6.2.3-1, “Containment Mechanical Penetration and Isolation Valves,” and will be tracked as **Confirmatory Item 23-23.**

The operating conditions during the RCP-EHX tube rupture transient will be added to the valve and piping data sheets, AP1000 Piping Class Sheets and Standard Details, APP-PLO2-Z0-001, rev. 5, and will be tracked as **Confirmatory Item 23-24**.

The staff confirmed the transient results from Westinghouse calculation APP-CCS-M3C-164, rev 0, and concludes that the containment isolation valves will be able to close at these pressure and temperature conditions and maintain their integrity following a tube rupture in the RCP-EHX.

The designation of Quality Group C and Seismic Category I criteria to CCS-PL-V270 and CCS-PL-V271 in the AP1000 DCD, Table 3.2-3 and Table 3.9-12, “List of ASME Class 1, 2, and 3 Active Valves,” will be tracked as **Confirmatory Item 23-25**.

DCD Tier 2, Section 14.2.9.1.10, “Containment Isolation and Leak Rate Testing,” addresses testing for proper operation of safety-related containment isolation valves listed in Table 6.2.3-1. Table 6.2.3-1 was modified to include the high bearing temperature valve closures as part of DCP 72, and will be tracked as **Confirmatory Item 23-26**.

The proposed revision to DCD Section 9.2.2.4.5.2 provides an adequate description of the automatic isolation function and will be tracked as **Confirmatory Item 23-27**.

The addition of Modes 3 and 4 applicability to the bases of TS 3.3.2 and inclusion of this trip function in Table 3.3.2-1 is **Confirmatory Item 23-28**.

Incorporation of changes to Functional Diagram 7.2-1 in a future revision of the DCD is being tracked as **Confirmatory Item 23-29**.

23.W Changes to Add a Vacuum Relief System to the Containment

23.W.1 Description of Proposed Changes

In letters dated August 16, 2010 (ML102310235), September 29, 2010 (ML102770447), and October 15, 2010 (ML102920143), Westinghouse submitted proposed design changes and supporting documentation, which add a containment vacuum relief system to the existing Containment Air Filtration System (VFS) vent line penetration. The NRC staff reviewed the system design, containment external pressure analyses, containment isolation functions, leak rate testing, descriptions of the valve design, qualification and inservice testing (IST) programs, instrumentation and controls, and associated Technical Specifications (TS).

The applicant's proposed design changes add a vacuum relief system to the existing VFS 16-inch vent line penetration as seen in the DCD, Tier 2, Figure 9.4.7-1, "Containment Air Filtration System P&ID", sheet 1 of 2. The proposed vacuum relief system consists of redundant vacuum relief devices sized to prevent differential pressure between containment and the shield building from exceeding the design value. Each of the two vacuum relief device flow paths consists of a a check valve (VFS-PL-V803A/B) inside containment, a motor operated butterfly valve (VFS-PL-V800A/B) outside containment, and associated piping. Each of these four valves also has a containment isolation function. The redundant check valves inside containment share a common inlet line with the vent line penetration and have independent discharge lines into containment. The redundant motor operated butterfly valves outside containment share a common inlet line. Each relief device, consisting of a check valve, connecting piping, and a motor operated valve (MOV), is designed to provide 100% of the required capacity to prevent a differential pressure across the containment vessel from exceeding the design value. Each relief flow path provides the required capacity, such that a single failure of any of the relief devices would not limit the flow below what is required to mitigate a containment vacuum relief event.

The normally closed butterfly valves are designed with motor operators that are powered from separate Class 1E DC battery sources. They are designed to close within 30 seconds of receipt of either an automatic containment isolation signal, or a manual isolation signal. They are designed to open automatically within 30 seconds when the containment pressure signal reaches Low-2 level and remain open to preclude exceeding the containment external design pressure. While the vacuum relief system MOVs are open, the containment will be at a vacuum and flow will be into containment. Once the vacuum condition inside containment is reduced to near ambient pressure conditions, the open signal would automatically clear. This would allow the vacuum relief system MOVs, VFS-PL-V800A/B, to close automatically in the event that a containment isolation signal or high radiation signal is present. The check valves are balanced to remain closed during normal operations, including containment vessel venting.

The proposed vacuum relief system is being added to the existing VFS. A short description of the vacuum relief system is being added to DCD section 9.4.7. Additionally, the proposed vacuum relief system design is shown on DCD, Tier 2, Figure 9.4.7-1, sheet 1 of 2, "Containment Air Filtration System P&ID". Since the proposed vacuum relief system valves are also part of the containment isolation design, valves VFS-PL-V800A/B and VFS-PL-V803A/B have been added to DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves", with two notes. The first, Note 8, indicates that the valves VFS-PL-V800A/B close on either a containment isolation signal or a high radiation in containment signal, and open on a Low-2 containment pressure signal. The second, Note 9, indicates that the Type C leak rate testing of the valves VFS-PL-V800A/B would be in the reverse direction, and that closure would occur in 30 seconds.

The applicant has changed the containment external design pressure specified in DCD Section 3.8.2.1.1 from 2.9 pounds per square inch differential (psid) to 1.7 psid based on the actuation point of the vacuum relief system. The applicant has also changed the containment external design analysis in DCD Section 6.2.1.1.4 to demonstrate the vacuum relief system is sufficient to mitigate the maximum expected external pressure.

The applicant has also modified AP1000 DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements," to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. The Inservice Testing (IST) Table will specify VFS-PL-V800A/B as motor-operated butterfly valves with safety-related missions to maintain close, transfer close, maintain open, and transfer open; safety functions as active, containment isolation, safety seat leakage, and remote position; ASME Code Class 2 and IST Category A valves; and IST type and frequency as remote position indication and exercise every 2 years, containment isolation leak test, exercise full stroke quarterly and operability test. The IST Table will specify check valves VFS-PL-V803A/B as relief valves with safety-related missions as maintain close, transfer close, and transfer open; safety functions as active, containment isolation, and safety seat leakage; ASME Code Class 2 and IST Category AC valves; and IST type and frequency as containment isolation leak test, exercise full stroke every refueling shutdown, and vacuum relief test every 2 years.

The applicant's proposed design changes include the addition of TS 3.6.10, "Vacuum Relief Valves," which address the proposed containment vacuum relief valve operation. The proposed design changes add a function to Table 3.3.2-1, "Engineered Safeguards Actuation System Instrumentation," for the signal which would open the containment vacuum relief valves on a Low-2 containment pressure level. In addition, the proposed changes extend the Mode

Applicability for TS 3.6.4, “Containment Pressure,” and TS 3.6.5, “Containment Air Temperature,” to include Modes 5 and 6, which support TS 3.6.10 requirements.

The applicant has also proposed corresponding changes to Tier 1 DCD sections.

23.W.2 Regulatory Basis

The regulatory basis for evaluating the applicant’s proposed design changes and supporting documentation is documented in Chapters 3, 6, 7, 9, and 16 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.”

The applicable regulations for the design, analyses, containment isolation, and containment leak rate testing aspects of the applicant’s proposed design changes are detailed in SRP 6.2.1.1.A, “PWR Dry Containments,” SRP 6.2.1, “Containment Functional Design,” SRP 6.2.4, “Containment Isolation System,” SRP 6.2.6, “Containment Leakage Testing,” and SRP 9.4.3, “Auxiliary and Radwaste Area Ventilation System.” Specifically, the following regulatory requirements and guidance apply:

- 10 CFR Part 50, Appendix A, GDC 2, “Design Bases for Protection Against Natural Phenomena.”
- 10 CFR Part 50, Appendix A, GDC 16, “Containment Design.”
- 10 CFR Part 50, Appendix A, GDC 38, “Containment Heat Removal.”
- 10 CFR Part 50, Appendix A, GDC 50, “Containment Design Basis.”
- 10 CFR Part 50, Appendix A, GDC 52, “Capability for containment leakage rate testing.”
- 10 CFR Part 50, Appendix A, GDC 53, “Provisions for containment testing and inspection.”
- 10 CFR Part 50, Appendix A, GDC 54, “Piping Systems Penetrating Containment.”
- 10 CFR Part 50, Appendix A, GDC 56, “Primary Containment Isolation.”
- 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.”
- 10 CFR 50.34(f), “Additional TMI-Related Requirements,” subparagraph (2)(xiv), regarding TMI Action Plan Item II.E4.2, “Containment Isolation Dependability.”
- 10 CFR 52.47(b)(1), which requires that a design certification application contain the appropriate ITAAC.
- RG 1.141, “Containment Isolation Provisions for Fluid Systems.”
- RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.”
- CSB BTP 6-4, “Containment Purging During Normal Plant Operations.”

The applicable regulations for the staff’s review of the functional design, qualification, and IST programs for the valves described in these proposed design changes are detailed in SRP 3.9.6 (Revision 3), “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints.” The staff also considered guidance provided in applicable

Commission SECY papers, Commission Staff Requirements Memoranda, generic letters, regulatory guides, and regulatory issue summaries. Specifically, the following regulatory requirements apply:

- 10 CFR Part 50, Appendix A, GDC 1, “Quality Standards and Records.”
- 10 CFR Part 50, Appendix A, GDC 2, “Design Bases for Protection against Natural Phenomena.”
- 10 CFR Part 50, Appendix A, GDC 4, “Environmental and Dynamic Effects Design Bases.”
- 10 CFR Part 50, Appendix A, GDC 54, “Systems Penetrating Containment.”
- 10 CFR 50.55a(f), which requires that valves whose function is required for safety be assessed for operational readiness in accordance with the applicable revision to the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code).
- 10 CFR 50.55a(b)(3), which takes exception to, or supplements, the ASME OM Code provisions for components within the scope of the IST Program.

The applicable regulations for the instrumentation and controls aspects of the applicant’s proposed design changes are detailed in SRP Section 7.3, “Engineered Safety Features,” and SRP Section 7.5, “Information Systems Important to Safety.” Specifically, the following regulatory requirements and guidance apply:

- 10 CFR Part 50, Appendix A, GDC 13, “Instrumentation and Control.”
- 10 CFR Part 50, Appendix A, GDC 19, “Control Room.”
- 10 CFR Part 50, Appendix A, GDC 20, “Protection System Functions.”
- 10 CFR Part 50, Appendix A, GDC 21, “Protection System Reliability and Testability.”
- 10 CFR 50.55a(h), which requires that protection systems must meet the requirements stated in either IEEE Std. 279, “Criteria for Protection Systems for Nuclear Power Generating Stations,” or in IEEE Std. 603-1991, “Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995.
- RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.”

The applicable regulations for evaluating the applicant’s proposed TS are detailed in SRP Chapter 16, “Technical Specifications – Revision 3, March 2010.” The staff’s evaluation focused primarily on confirming that changes to the generic technical specifications (GTS) reflect the Containment Systems design and operating information in DCD Sections 7.3 and 9.4.7, and the containment analyses described in DCD Section 6.2, and that changes to the GTS meet the requirements of 10 CFR 50.36.

23.W.3 Technical Evaluation

23.W.3.1 System Design and Analyses

The NRC staff reviewed the proposed design changes for compliance with GDC 16, Containment Design, which requires that the reactor containment and associated systems be designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The proposed vacuum relief system includes an interlock with the inboard containment isolation valve for the containment purge line. The interlock prevents the vacuum relief system from opening if the inboard isolation valve is open. The staff requested the applicant to explain how the safety function would be accomplished if the inboard valve was open and the interlock prevented the vacuum relief system from functioning. The applicant responded that the vacuum relief system actuation had priority over the interlock and that if a containment vacuum condition existed, the system would actuate regardless of the interlock. The applicant revised the statements in the DCD to clarify this design feature. Specifically, the new DCD Section 7.6.2.4 states that if a vacuum relief actuation signal is present, the vacuum relief signal takes precedence over the valve closure interlock. This design feature ensures the safety function is accomplished and removes the possibility of a single failure associated with the interlock disabling the system.

The staff also reviewed the revised external pressure analysis in DCD Section 6.2.1 for compliance with GDC 16. Conformance with acceptance criteria from SRP Section 6.2.1.1.A formed the basis for concluding whether GDC 16 was satisfied.

In DCD Revision 15, Westinghouse determined that the worst case event for maximum external pressure was the loss of all ac power sources during extreme cold weather. The analysis was conducted using the WGOTHIC code with the following conservatisms:

- External temperature boundary condition set at the minimum site parameter from DCD Tier 1, Table 5.0-1, -40 °F.
- Initial containment relative humidity set to 100% to maximize the vapor content and allow for the greatest reduction in pressure due to steam condensation.
- Loss of ac power event immediately reduces the containment heat load to zero.
- Initial containment temperature set at the maximum temperature Limiting Condition For Operation (LCO) from Technical Specification 3.6.5, 120 °F.
- Initial containment pressure set at the minimum pressure LCO from Technical Specification 3.6.4, -0.2 psid.
- The velocity of air flowing over the containment shell is set at 24.8 ft/s, which correlates to an external wind speed of 48 mph.
- No air leaks into containment.

The analysis demonstrated the differential pressure across the containment vessel remained below the design value for one hour, which was found to be sufficient time for plant operators to take action to mitigate the event.

Westinghouse revised the external pressure evaluation to incorporate the vacuum relief system and eliminate the need for operator action, to remove unnecessary conservatisms associated with the first three bulleted items above (no changes were made to the final four assumptions), and to incorporate design changes associated with the DCD Amendment. The analysis was based on the WGOTHIC evaluation model approved by the staff in Section 23.S of this report, modified for use in an external pressure transient. The worst case event remains the loss of all ac power during cold weather. Westinghouse recognized that an operating reactor does not produce enough heat to raise the internal temperature to the maximum value on a cold day; thus, the combination of the maximum internal temperature and minimum external temperature is non-mechanistic. In order to remove conservatism, the first step in the revised analysis is determining the minimum external temperature that can sustain the internal temperature at 120 °F. This pre-transient stage of the analysis assumed a heat rate equal to the value used to size the active containment cooling system. In order to minimize the external shell heat transfer coefficient, the annulus air flow was set to natural convection. The applicant used this model to determine equilibrium containment temperatures associated with various external temperatures and found that 25 °F is the minimum external temperature capable of maintaining the maximum internal temperature (120 °F).

The next step in the analysis is the loss of power transient. In this phase, the initial temperatures of the internal containment volume, containment shell, baffle and shield building were set to the equilibrium values found during the pre-transient run. For the design basis run, the internal temperature is 120 °F and the external temperature is 25 °F, which bounds the proposed LCO for TS 3.6.10 requiring the internal/external temperature differential be less than or equal to 90 °F. Because the initial conditions exceed the TS LCO, the equilibrium temperatures derived for the shell, shield building and baffle were also found to be bounding. The original analysis assumed zero heat loss during the transient. This is non-mechanistic because it neglects the contribution of reactor system sensible and decay heat. In order to remove some of this conservatism, the heat load to containment is assumed equal to the sensible heat from a reactor in Mode 3 at normal operating temperature and normal operating pressure. Decay heat is not considered because it is assumed the reactor has never been critical. The staff agrees that these assumptions produce conservatively low heat loads. Westinghouse stated that the previous assumption of 100% relative humidity was also non-mechanistic because the temperature of the containment vessel is below the dew point. As a result, the revised analysis uses a relative humidity of 82%, representing a 25% margin over the equilibrium value. The staff finds this approach realistically bounds the relative humidity.

During the transient, the external temperature is assumed to decrease 30 °F per hour from an initial value of 25 °F until it reaches the minimum site value of -40 °F at 7800 seconds. To demonstrate this assumption is bounding, Westinghouse evaluated hourly meteorological data gathered at Charlotte, NC from 1/1/1975 to 6/24/1996 and at Duluth, MN between 1/1/1975 and

1/1/2010. Charlotte, NC was chosen by the applicant as a location having typical meteorological behavior for the Southeast Regional Climate Zone. The applicant also chose the Duluth, MN location because it represents the basis for the AP1000 DCD minimum allowable operation temperature of -40 °F. Westinghouse found that the maximum observed hourly temperature decrease in Charlotte, NC was 20 °F (from 73 °F to 53 °F). Westinghouse also found that the maximum observed hourly temperature decrease in Duluth, MN, during below-freezing conditions was 17 °F (from 19 °F to 2 °F). The staff finds the applicant's use of a 30 °F per hour temperature drop to be reasonable because Westinghouse's analysis included several recent years worth of data at a typical southeast regional site (Charlotte, NC) and a typical cold weather site (Duluth, MN), and because there is significant margin between the observed values (20 °F for Charlotte, NC and 17 °F for Duluth, MN) and the assumed hourly decrease (30 °F).

The WGOTHIC model incorporated one 6-inch valve with a conservatively calculated system resistance, designed to open 20 seconds after the internal containment pressure reached the setpoint of -1.2 psid. The safety analysis limit (-1.2 psid) is clearly identified in the DCD and will be used to develop the Low-2 containment pressure setpoint under the TS setpoint methodology program. The 20 seconds required to develop full flow is consistent with the mechanical design requirements identified in Section 9.1.1 of Enclosure #4 of Westinghouse's October 15, 2010 submittal. Because it is reasonable to expect a butterfly valve to allow significant flow at 60% of the stroke, this is also consistent with the proposed acceptance criteria for ITAAC item 2 from Table 2.7.6-2 requiring the valves to open within 30 seconds.

The transient response demonstrated that the vacuum relief system limits the containment pressure to a minimum value of -1.63 psid, which is bounded by the design value of -1.7 psid. As described in Enclosure #6 of its October 15, 2010, submittal, Westinghouse ran additional vacuum relief scenarios at external temperatures of -40 °F and 50 °F and used the results to confirm that the design basis case was limiting. The -40 °F case, initiated with an internal temperature of 88 °F, was also used to demonstrate that the internal to external temperature differential LCO is not required if the internal containment temperature is less than 88 °F. This bounds TS 3.6.10 Required Action B.2 to reduce containment average temperature to ≤ 80 °F when the inside to outside differential air temperature does not meet the LCO.

The staff reviewed the analysis basis during August 27, 2010, and October 3, 2010, audits of APP-SSAR-GSC-112, "AP1000 External Pressure Analysis to Confirm Sizing of the Vacuum Relief System," Revision 0 and Revision 1. This Westinghouse report states that while various transients were considered for the external pressure evaluation, the loss of ac power event remains the most limiting. This is the same event that was found to be bounding in the original analysis, for the reasons discussed in Section 6.2.1 of NUREG-1793. The staff finds this evaluation remains applicable to the revised design because the proposed design changes will not impact determination of the limiting event.

The analysis applied a 24.8 ft/sec annulus velocity during the transient to represent an external wind of 48 mph. The staff finds this will produce conservatively high heat transfer coefficients because the basis for the correlation was testing on a prior version of the shield building. As discussed in Section 23.S, there is an increased pressure drop associated with the revised shield building, so the annulus velocity associated with a 48 mph wind speed will be less than 24.8 ft/sec.

The staff conducted an additional audit on August 30, 2010, on three supporting calculation notes. The staff found the baseline WGOTHIC external pressure model, described in APP-SSAR-GSC-746, “Containment Response Analysis for the AP1000 Shield Building Design Change,” Revision 0, acceptable because it used the same methodology as the Rev. 15 DCD analysis to transform the LOCA model into an external pressure model. The staff reviewed the calculations for the total system resistance in APP-VFS-M3C-224, “Containment Vacuum Relief System Resistance Calculation,” Revision 0. The analysis was found to be conservative because the resulting value represents the resistance associated with the more limiting single flow path configuration (which assumes two failures) rather than the worst case single failure. It also incorporates a 65% margin over the calculated value to account for potential design changes. The staff reviewed the design basis for the minimum heat load into containment as described in APP-SSAR-GSC-003, “Calculation of the Total Loss of RCS Heat from Mode 3 with Station Blackout,” Revision 0. The staff finds this analysis to be conservative because the code used to generate the heat load does not consider contributions from the major component support structures, which are typically large contributors to containment sensible heat.

The staff performed a confirmatory analysis using the CONTAIN computer code and the AP1000 model described in Section 23.S.3 of this report. The model was altered to incorporate one vacuum relief line and to remove the shield building in order to apply the annulus air velocity directly to the external containment shell, producing conservatively high heat transfer coefficients on this surface. The model incorporated the same valve parameters as the applicant except the valve opening delay time was increased to 30 seconds. The staff evaluated the design basis case and the two sensitivity studies. The containment internal pressure transients demonstrated the design basis case, with a minimum negative pressure of -1.56 psid, was limiting. These results are consistent with the Westinghouse evaluation.

SRP Section 6.2.1.1.A recommends that the margin between the external design pressure and the conservatively calculated minimum pressure be at least 10% at the construction permit (CP) or design certification (DC) stage of review. The requirement at the operating license stage of review is that the maximum calculated value be less than the design value (zero-margin criterion). The more restrictive margin is applied at the CP and DC stage to account for revised or upgraded analytical models or minor changes that may result in a decrease in the margin in the as-built design. The margin in the revised design basis analysis is 4%. Westinghouse proposed this is acceptable because the calculation of the vacuum relief system resistance included a 65% margin to accommodate system variances associated with plant construction. Westinghouse claims this addresses the as built considerations that form the basis for the SRP recommended 10% margin at the DC stage. Furthermore, when the margin was removed from the loss coefficient, the resultant minimum external pressure was 1.46 psid, which provides a 14% margin to the design value. The staff agrees this is an acceptable approach because it meets the intent of the SRP recommendation. This finding is consistent with the staff evaluation in NUREG-1793 Section 6.2.1 where a zero-margin criterion was deemed acceptable for the AP1000 peak accident pressures at the DC stage.

The staff reviewed the proposed containment vacuum relief design for compliance with the requirements of GDC 2, Design Bases for Protection Against Natural Phenomena, which requires that the system be capable of withstanding the effects of earthquakes. The new portions of the system are safety-related and Seismic Category 1 and, therefore, meet the requirements of GDC 2. However, in Chapter 9.4.7 of the DCD, the applicant removed statements that demonstrate the nonsafety related portions of the VFS system meet GDC 2. The staff requested that the applicant explain how the VFS system meets GDC 2. The applicant revised the discussion in DCD Section 9.4.7 to state that system equipment and ductwork whose failure could affect the operability of the safety-related systems or components are designed to seismic Category II requirements. This conforms to the certified DCD Revision 15 design, and is adequate to demonstrate compliance with GDC 2.

As a result, the staff finds that the design and design features described in the DCD ensure the requirements of GDC 2 and GDC 16 will be met. The proposed changes to Tier 1, Section 2.7.6, and Tier 2, Sections 6.2.1, 6.2.3, and 9.4.7 of the AP1000 DCD will be tracked as **Confirmatory Item 23-30**. The staff's evaluation of the AP1000 steel containment, including load combinations and design and analysis procedures, is described in Section 3.8.2 of this SER.

23.W.3.2 Containment Isolation and Leak Rate Testing

The staff reviewed the proposed vacuum relief system design for compliance with GDC 56, Primary containment isolation, which requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment be provided with containment isolation valves. One acceptable design includes: One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. The design of the vacuum relief system complies with the requirements of GDC 56.

10 CFR 50.34(f)(2)(xiv)(B) requires that each non-essential penetration (except instrument lines) shall have two isolation barriers in series. The vacuum relief design consists of two lines which connect directly with the containment atmosphere and penetrate the primary containment. This design complies with the requirement of GDC 56 by providing each vacuum relief device with a check valve inside containment and a motor operated butterfly valve outside containment. This design complies with the 10 CFR 50.34(f)(2)(xiv)(B) redundancy requirement. If a check valve failed to close during an accident, the MOVs in series with it would close on a "T" signal (described in DCD Section 6.2.3), thereby providing containment isolation.

To meet the requirements of GDC 56, upon loss of actuating power, automatic isolation valves should take the position of greatest safety. All power-operated isolation valves should have position indications in the main control room. The safe post accident position for the valves in the vacuum relief system is closed. The motor operated butterfly valves would close on a "T" containment isolation signal. To improve the reliability of the isolation function, these valves also close on a high radiation in containment signal. These valves are each powered by Class 1E batteries to ensure that they would close on these signals. Their position indication in the main control room is shown on DCD Table 6.2.3-1. Valves VFS-PL-V803A and VFS-PL-V803B are self actuated check valves which would close if a vacuum does not exist inside containment. These check valves would either close or remain closed post accident. Since the proposed vacuum relief system valves are also part of the containment isolation design, VFS-PL-V800A/B and VFS-PL-V803A/B have been added to DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves," with two notes. The first, Note 8, indicates that the valves VFS-PL-V800A/B close on either a containment isolation signal or a high radiation in containment signal, and open on a Low-2 containment pressure signal. The second, Note 9, indicates that the Type C leak rate testing of the valves VFS-PL-V800A/B would be in the reverse direction, and that closure would occur in 30 seconds. The staff will track these changes to Table 6.2.3-1 with **Confirmatory Item 23-31**.

The staff also reviewed these proposed design changes for compliance with the requirements of GDC 54, as it relates to providing piping systems penetrating the containment with containment isolation capabilities having redundancy and reliability which reflect their importance to safety. To meet the reliability requirements, the components performing a containment isolation function are acceptable if Group B quality standards, as defined in RG 1.26, apply, and the components are designated seismic Category I in accordance with RG 1.29.

The containment isolation section of the vacuum relief design, consisting of the containment isolation valves and the included piping, are designed to ASME Section III, Class 2 criteria. The containment penetrations are classified as Quality Group B, as defined in RG 1.26, and seismic Category 1. These designations are shown in DCD Tier 2, Figure 9.4.7-1 and Table 6.2.3-1, and Tier 1, Table 2.2.1-1, Table 2.2.1-2, and Figure 2.2.1-1, "Containment Isolation System." Westinghouse has selected the appropriate mechanical design classification.

10 CFR 52.47(b)(1) requires that a Design Certification application include the appropriate inspections, tests, analyses and acceptance criteria (ITAAC). The ITAAC for the containment isolation MOV closing time are shown in DCD Tier 1, Table 2.7.6-2 and Table 2.2.1-3. With **Confirmatory Item 23-32,** the staff will track these additions in the tables and figures listed above.

In meeting the requirements of GDC 54, relating to lines which provide open paths from the containment to the environs, such as containment purge and vent, the closure times of the isolation valves should minimize the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and to prevent degradation of emergency core cooling system effectiveness by reduced containment back-pressure. CSB Branch Technical Position (BTP) 6-4 provides additional guidance on the design and use of the containment purge systems which may be used during the normal plant operating modes.

The VFS is used to purge the containment atmosphere of airborne radioactivity during normal plant operation. The proposed containment vacuum relief system is a safety grade system used to mitigate a containment external pressure scenario, and is part of the VFS, sharing the same containment penetration. The purge system is designed in accordance with CSB BTP 6-4. The purge component of the VFS uses 16-inch supply and exhaust lines and containment isolation valves designed to close in 10 seconds. The vacuum relief component of the VFS uses 6-inch supply lines and containment isolation valves designed to close in 30 seconds.

In the event of a LOCA, a maximum time of 30 seconds for closure of the 16-inch valves was assumed for the analysis for the radiological consequences, as seen in DCD Tier 2, Table 15.6.5-2. This closure time is conservative, as the valve design closure time is 10 seconds. This closure time is consistent with the guidance in CSB BTP 6-4, and was found acceptable as an assumed closure time in the radiological analysis in NUREG-1793, Section 6.2.4.13. The 30 second closure times of the two new 6-inch vacuum relief valves would be bounded by the current design, as the radiological consequences following a LOCA have already been found acceptable for two open 16-inch valves, closing in 30 seconds.

To analyze the LOCA containment minimum backpressure, containment purge was assumed to be in operation at time zero and air was vented through both the 16-inch diameter containment purge supply and exhaust lines until the isolation valves fully closed. These valves were modeled to close 12 seconds after the 8 psig closure setpoint was reached, as described in DCD Tier 2, Section 6.2.1.5.3. This closure time was found acceptable as an assumed closure time in the ECCS analysis (reflood backpressure) in NUREG-1793, Section 6.2.4.13. In DCD, Tier 2, Section 9.4.7.2.1, the maximum time for closure of the vacuum relief valves, 30 seconds, was evaluated for its impact on the calculation of the LOCA backpressure. The minimum containment backpressure following two, 6-inch valves closing in 30 seconds, is expected to be bounded by the containment backpressure resulting from two, 16-inch valves closing in 12 seconds.

In NUREG-1793, Section 6.2.6, the staff reviewed the applicant's proposed containment leakage rate testing program for AP1000 facilities described in DCD Tier 2, Section 6.2.5 and in the proposed Technical Specifications of DCD Tier 2, Chapter 16. The staff reviewed the information in the DCD for conformance to 10 CFR Part 50, Appendix J, and to GDC 52, "Capability for Containment Leakage Rate Testing," and GDC 53, "Provisions for Containment Testing and Inspection." GDC 52 requires the containment and associated equipment to be designed such that the periodic containment integrated leakage rate tests can be conducted at containment design pressure. GDC 53 requires that the containment allow periodic inspection, surveillance, and testing of certain systems, structures, and components.

The staff used the guidance, staff positions, and acceptance criteria of SRP Section 6.2.6 and RG 1.163, "Performance-Based Containment Leak-Test Program," in conducting its review. The staff concluded that the AP1000 containment leakage rate testing program complied with the acceptance criteria of Section 6.2.6 of the SRP by satisfying the containment leakage rate testing requirements of GDC 52 and GDC 53, and Appendix J to 10 CFR Part 50.

The proposed vacuum relief system valves, VFS-PL-V800A/B and VFS-PL-V803A/B are also part of the containment isolation design and have been added to AP1000 DCD Tier 2, Table 6.2.3-1, “Containment Mechanical Penetrations and Isolation Valves.” These new containment isolation valves will be included in the AP1000 containment leak rate test program, and will be Type C tested, as indicated in DCD Tier 2, Table 6.2.3-1. Adding the new containment isolation valves to the already approved containment leak rate test program, meets the Section 6.2.6 of the SRP acceptance criteria for containment leak rate testing of the new valves.

With regard to 10 CFR 52.47(b)(1), the applicant has added appropriate ITAAC to Tier 1. The applicant has included ITAAC to verify the valves function in both the open and closed directions. The staff finds the additional ITAAC acceptable.

23.W.3.3 Valve Design, Qualification, and Testing

The NRC staff reviewed the functional design, qualification, and IST program descriptions for the valves to be used in the applicant’s proposed containment vacuum relief system. The NRC staff provided comments to Westinghouse related to the functional design, qualification, and IST program descriptions for the valves to be used in the proposed containment vacuum relief system. Westinghouse provided responses to all NRC requests for information. The NRC staff also performed audits of the valve design specifications and data sheets referenced in the proposed design changes.

The proposed AP1000 containment vacuum relief system design includes parallel motor-operated butterfly valves VFS-PL-V800A/B outside containment connected by a single pipe line to parallel check valves VFS-PL-V803A/B inside containment. In the event of a vacuum condition within the AP1000 containment, butterfly valves VFS-PL-V800A/B will receive an automatic electric signal to their battery-powered motor actuators to open the valves. As a result, outside air will flow through the connected piping up to check valves VFS-PL-V803A/B that will open upon a preset differential air pressure across the valves and allow air to enter the AP1000 containment to relieve the vacuum condition. During its review of the applicant’s submittal, the NRC staff requested that Westinghouse clarify the design, qualification, and testing requirements for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. Westinghouse provided this additional information in letters dated September 29, 2010, and October 15, 2010.

In Section 3.9.6 of this SER, the NRC staff concluded that the AP1000 DCD continues to provide sufficient information to satisfy 10 CFR Parts 50 and 52 for the design aspects of the functional design, qualification, and IST programs for safety-related valves to be used in the AP1000 reactor, pending the resolution of confirmatory items identified in that SER section. The staff noted in SER Section 3.9.6 that the operational program aspects regarding the functional design, qualification, and IST programs for safety-related valves will be reviewed as part of the evaluation of a COL application referencing the AP1000 certified design. The NRC staff is tracking the modifications to the AP1000 DCD related to the functional design, qualification, and IST programs for valves as part of confirmatory items specified in SER Section 3.9.6.

In response to NRC staff comments, Westinghouse revised Subsection 9.1.1, “Mechanical Design Requirements,” in Section 9.1, “Outboard Motor Operated Valves VFS-PL-V800A/B,” of its submittal to specify that the butterfly valves in the AP1000 containment vacuum relief system will be designed in accordance with AP1000 DCD Tier 2, Section 3.9, “Mechanical Systems and Components.” The NRC staff describes its review of the design requirements for safety-related MOVs in Section 3.9.6 of this SER. For example, as indicated in the Westinghouse letter dated March 5, 2010, Westinghouse plans to revise Section 3.9 of the AP1000 DCD Tier 2 to specify that qualification of safety-related valves will be in accordance with ASME Standard QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.” The NRC staff accepted the application of ASME Standard QME-1-2007 in Revision 3 to Regulatory Guide 1.100, “Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” with certain staff positions. Also in response to NRC staff comments, Westinghouse revised Subsection 9.1.1 of its submittal to specify the design requirements for capacity coefficient and stroke time for full flow capacity of these butterfly valves. The NRC staff finds the reference to the provisions in AP1000 DCD Tier 2, Section 3.9, with the valve-specific design requirements in Subsection 9.1.1 of the applicant’s submittal, to be acceptable for the functional design and qualification of the butterfly valves to be used in the proposed containment vacuum relief system with respect to the design aspects of the AP1000 Design Certification amendment as discussed in Section 3.9.6 of this SER.

During its review, the NRC staff requested that Westinghouse describe the availability of adequate power supplies for motor-operated butterfly valves VFS-PL-V800A/B to perform their design-basis functions. Subsection 9.1.2, “Valve Electrical Requirements,” in Section 9.1 of the applicant’s submittal specifies the design requirements for butterfly valves VFS-PL-V800A/B to be powered from separate Class 1E DC battery sources. Subsection 9.1.2 also specifies that butterfly valves VFS-PL-V800A/B will be designed to stroke twice for their design-basis operation. Subsection 9.1.2 indicates that the electrical calculations using the methodology described in IEEE Standard 485, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications,” will take into consideration the starting current and stroke time for the MOV operations. Section 8.2, “DC Power Systems,” in AP1000 DCD Tier 2 describes the application of IEEE-485 for DC power systems in the AP1000 certified design. In addition, the functional qualification of butterfly valves VFS-PL-V800A/B in accordance with ASME Standard QME-1-2007 as accepted in Revision 3 to RG 1.100 will demonstrate the power capability of these MOVs to perform their design-basis functions. The NRC staff finds Westinghouse to have described the methodology for providing adequate power availability to butterfly valves VFS-PL-V800A/B in an acceptable manner.

In response to NRC staff comments, Westinghouse revised Subsection 9.1.3, “Testing Requirements,” in Section 9.1 of its submittal to specify that butterfly valves VFS-PL-V800A/B will be tested in accordance with AP1000 DCD Tier 2, Section 3.9.6, “Inservice Testing of

Pumps and Valves." As discussed above, the NRC staff describes its review of the description of the IST program for safety-related valves in Section 3.9.6 of this SER. Therefore, the NRC staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9.6 for the description of the IST program for the butterfly valves to be used in the AP1000 containment vacuum relief system to be acceptable with respect to the design aspects of the AP1000 Design Certification amendment as discussed in Section 3.9.6 of this SER. The NRC staff will review the operational aspects for the IST program for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

Subsection 9.2.1, "Mechanical Design Requirements," in Section 9.2, "Inboard Self Actuated Valves VFS-PL-V803A/B," of the applicant's submittal specifies the design requirements for the AP1000 vacuum relief containment isolation check valves. ASME OM Code, Table ISTC-3500-1 states that if a check valve used for pressure relief is capacity certified, then it shall be classified as a pressure or vacuum relief valve. Therefore, check valves VFS-PL-V803A/B are within the scope of ASME *Boiler & Pressure Vessel Code* (BPV Code), Section III, Subsection NC-7000, "Overpressure Protection," as defined in Paragraph NC-7110, and require capacity certification as defined in Paragraph NC-7750. In response to NRC staff comments, Westinghouse revised Subsection 9.2.2, "Testing Requirements," of its submittal to specify that VFS-PL-V803A/B are vacuum relief valves that will be designed and qualified in accordance with ASME BPV Code, Section III, Subsection NC-7000. During its review, the NRC staff informed Westinghouse that the design requirements for check valves VFS-PL-V803A/B should be included in Subsection 9.2.1 (rather than Subsection 9.2.2). As a result, Westinghouse modified Subsection 9.2.1 to specify (1) VFS-PL-V803A/B are vacuum relief valves that will be designed and qualified in accordance with ASME BPV Code, Section III, Subsection NC-7000; (2) the valves will be qualified using the provisions in AP1000 DCD Tier 2, Section 3.9; (3) the valves will be designed with an allowable tolerance for the 0.2 psi differential opening pressure; and (4) the valve flow capacity to relieve vacuum conditions to avoid the containment external design pressure from being exceeded. The NRC staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9 and the ASME BPV Code design requirements, with the valve-specific design requirements specified in Subsection 9.2.1 of the applicant's submittal, to be acceptable for the functional design and qualification of the check valves to be used in the AP1000 containment vacuum relief system.

Subsection 9.2.2 in Section 9.2 of the applicant's submittal specifies the testing requirements for AP1000 vacuum relief containment isolation check valves VFS-PL-V803A/B. Because these check valves provide a vacuum relief function with a design opening pressure, the NRC informed Westinghouse that its submittal should specify that these valves will be tested in accordance with ASME OM Code, Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants." In response, Westinghouse revised Subsection 9.2.2 to specify that these check valves are tested in accordance with AP1000 DCD Tier 2, Section 3.9.6 and ASME OM Code, Appendix I. Subsection 9.2.2 also states that these check valves will be tested in both directions in light of their safety functions in both the open and close directions. Subsection 9.2.2 specifies that 0.2 psid will be used as an acceptance criterion for the check valve opening test. As discussed above, the NRC staff describes its review of the description of the IST program for safety-related valves in Section 3.9.6 of this SER. Therefore, the NRC staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9.6 and ASME OM Code, Appendix I, for the description of the IST program for the check valves to be used in the AP1000 containment vacuum relief system to be acceptable with respect to the design aspects of the AP1000 Design Certification amendment as discussed in Section 3.9.6 of this SER. The NRC staff will review the operational aspects regarding the IST program for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

In response to NRC staff comments, Westinghouse modified Section 6.0, "Containment Isolation Consideration," of its submittal in the "Position" paragraph for 10 CFR Part 50, Appendix A, GDC 54, to clarify that butterfly valves VFS-PL-V800A/B cannot be tested in the direction of containment leakage. Therefore, these butterfly valves will be tested in the reverse direction of containment leakage. This testing is more conservative because the valves will be installed such that the containment pressure will assist in sealing the valve closed to minimize containment leakage. The applicant's submittal indicates the application of ANSI/ANS 56.8, "Containment System Leakage Testing Requirements," for containment valve testing as referenced in the AP1000 DCD. The NRC staff requested that Westinghouse clarify that the butterfly valves will be installed in an orientation that provides for containment pressure to help seal the valve closed. In response, Westinghouse revised Section 6.0 of its submittal to specify that the butterfly valves in the containment vacuum relief system will be installed such that containment pressure will assist in sealing the valve closed. The NRC staff finds that Westinghouse has clarified the orientation of the butterfly valves in the containment vacuum relief system to support the direction of leakage testing for these valves.

As part of its submittal, Westinghouse plans to modify AP1000 DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements," to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. The IST Table will specify VFS-PL-V800A/B as motor-operated butterfly valves with safety-related missions to maintain close, transfer close, maintain open, and transfer open; safety functions as active, containment isolation, safety seat leakage, and remote position; ASME Code Class 2 and IST Category A valves; and IST type and frequency as remote position indication and exercise every 2 years, containment isolation leak test, exercise full stroke quarterly and operability test. The IST Table will specify check valves VFS-PL-V803A/B as relief valves with safety-related missions as maintain close, transfer close, and transfer open; safety functions as active, containment isolation, and safety seat leakage; ASME Code Class 2 and IST Category AC valves; and IST type and frequency as containment isolation leak test, exercise full stroke every refueling shutdown, and vacuum relief test every 2 years. During its review, the NRC staff requested that Westinghouse resolve an apparent inconsistency between the IST Table and its Note 39. In a letter dated October 15, 2010, Westinghouse provided a planned revision to the IST Table to clarify that the containment vacuum relief butterfly valves will be exercised quarterly and that the containment vacuum relief check valves will be full stroke exercised every refueling shutdown. The Westinghouse submittal indicates that Note 39 to the IST Table will justify the extended test interval for the check valves because of their location inside containment. The NRC staff finds the planned provisions for IST activities for the butterfly valves and check valves in the containment vacuum relief system to comply with the requirements in 10 CFR 50.55a and the ASME OM Code and, therefore, to be acceptable. The proposed changes to the AP1000 DCD will be tracked as a **Confirmatory Item 23-33**.

The NRC staff requested that Westinghouse clarify the provisions for inspection and maintenance of the valves in the AP1000 containment vacuum relief system. In its response, Westinghouse clarified that the inspection and maintenance of butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B are the same as those for other safety-related valves. These valves are required to satisfy the design, inspection, and testing requirements described in the applicable section of the AP1000 DCD. These valves will be designed for the full lifetime of their service, and will not require inspection following individual actuations. Corrective action will be taken for any identified malfunction. Periodic testing and inspection results will be documented and trended. The plant predictive maintenance program and its associated condition monitoring will include these valves. The NRC staff finds that Westinghouse has clarified the implementation of plant programs for inspection and maintenance for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B that complies with the NRC regulations and, therefore, is acceptable.

The applicant's submittal indicates that the scope of AP1000 DCD Tier 1, Section 2.2.1, "Containment System," will be expanded to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. Table 2.2.1-3, "Inspections, Tests, Analyses, and Acceptance Criteria [ITAAC]," for the Containment System specifies ITAAC for piping and component design in accordance with ASME BPV Code Section III; piping, components, and welds satisfying ASME BPV Code, Section III requirements for integrity; seismic design-basis capability; Class 1E environmental qualification; valve operating times; and valve functional qualification. In its letter dated October 15, 2010, Westinghouse provided a planned change to Table 2.2.1-3 to specify the closing time for the butterfly valves in the AP1000 containment vacuum relief system. Westinghouse also indicated that Table 2.7.6-2 in AP1000 DCD Tier 1, Section 2.7.6, "Containment Air Filtration System," will include new ITAAC for the opening time for the butterfly valves in the AP1000 containment vacuum relief system. The NRC staff finds these ITAAC to be acceptable for the butterfly valves and check valves in the AP1000 containment vacuum relief system. The proposed changes to the AP1000 DCD will be tracked as **Confirmatory Item 23-34**.

Section 9.3, "Valve Design Specifications and Datasheets," of the applicant's submittal indicates that the design specifications and data sheets for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B were available for review at the Westinghouse office in Rockville, Maryland. On September 16, 2010, the NRC staff reviewed the referenced documents at the Westinghouse office. During this audit, the staff found that the referenced valve specifications and data sheets had not been updated to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. As a result, Westinghouse revised Section 9.3 in its submittal to describe the data sheets to be prepared for the butterfly valves and check valves to be used in the AP1000 containment vacuum relief system to support the applicable valve design specifications. On October 7, 2010, the staff audited the preliminary valve specifications and data sheets at the Westinghouse office in Rockville, Maryland. The NRC staff finds that Westinghouse has provided an acceptable description of the valve design specifications and data sheets to support the design and qualification of the butterfly valves and check valves to be used in the AP1000 containment vacuum relief system.

ASME BPV Code, Section III, Article NE-7000, “Overpressure Protection,” Subsection NE-7150, “Acceptable Pressure Relief Devices,” specifies in NE-7152 that vacuum relief devices shall meet the construction requirements applicable to Class 2 valves. The Code also states that valve devices intended to provide vacuum relief, and which are operated by indirect means depending upon an external energy source, are not acceptable unless the following conditions are met: (1) at least two independent external power-operated valve and control systems are employed so that the required relieving capacity is obtained if any one of the valve systems should fail to operate when called upon to do so; and (2) at least one self-actuating vacuum relief device of equivalent relieving capacity is provided in series with each of the external power-operated valves. The Code indicates that acceptable self-actuating vacuum relief devices include balanced self-actuating, horizontally installed, swing disk valves, with provisions for adjustment for the differential pressure under which the valves will operate.

For the proposed AP1000 containment vacuum relief system, Subsections 9.1.1 and 9.2.1 in Section 9.2 of the Westinghouse submittal specify that butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B will be designed as ASME BPV Code, Class 2 valves. Motor-operated butterfly valves VFS-PL-V800A/B will receive an automatic electric signal to their battery-powered motor actuators to open the valves from separate Class 1E DC battery sources. Check valves VFS-PL-V803A/B will be designed as swing check valves installed in the horizontal direction with a 0.2 psi differential opening pressure. The design of the proposed AP1000 containment vacuum relief system includes parallel check valves VFS-PL-V803A/B in series with the parallel butterfly valves VFS-PL-V800A/B. The NRC staff finds that the design of the vacuum relief devices for the proposed AP1000 containment vacuum relief system satisfies the provisions in Subsection NE-7150 of ASME BPV Code, Section III.

23.W.3.4 Instrumentation and Control

As documented in Chapter 7 of NUREG-1793, the staff reviewed and approved the Engineered Safety Features and Safety-Related Display Information as specified in Sections 7.3 and 7.5, respectively, of AP1000 DCD, Revision 15. The staff reviewed these proposed design changes using the review procedures described in SRP Sections 7.3 and 7.5.

The applicant created a new functional diagram Figure 7.2-1 (Sheet 19 of 21) for controlling the proposed containment vacuum relief MOVs from the PMS. The proposed vacuum relief system MOVs are designed to close within 30 seconds by using the existing VFS isolation signal from the PMS, which includes closure from an automatic containment isolation signal, a High-1 containment radiation signal, a manual containment isolation, or a manual containment cooling signal. In accordance with these proposed design changes, the normally closed vacuum relief system MOVs would open automatically during either of the following conditions:

- 2-out-of-4 coincidence logic for a Low-2 containment pressure condition.
- manual actuation of either of the two momentary controls.

The 2-out-of-4 logic also receives signals that indicate whether divisions have been bypassed. If one division is bypassed for test or maintenance, the logic will automatically be modified to 2-out-of-3 coincidence logic. Actuating either of the two momentary manual controls in the main control room (MCR) would actuate all applicable divisions. Separate momentary controls are also provided in the MCR for manually resetting the containment vacuum relief system. In these proposed design changes, the open control function has priority over the closing control function for the containment vacuum relief MOVs. In addition, in order to prevent the purge line isolation inside valve from being opened simultaneously with the vacuum relief system MOVs, an interlock logic is implemented to make sure that the vacuum relief system MOVs could not be opened unless the purge line isolation inside valve is closed; if open, vacuum relief system MOVs would close.

While the vacuum relief system MOVs are open, the air flow would be into the containment, which is at a vacuum. Once the vacuum condition inside containment is reduced to near ambient pressure conditions, the open signal would automatically clear. This would allow the vacuum relief system MOVs to close automatically in the event that a containment isolation signal or high radiation signal is present.

The applicant included the necessary design change information in the revised Tier 2, Section 7.3.1, Tables 7.3-1 and 7.3-3, new functional diagram Figure 7.2-1 (Sheet 19 of 21), and Tier 1, Table 2.5.2-3 in the DCD. The applicant also added a new Section 7.6.2.4 in Tier 2 to address the interlock for the containment vacuum relief isolation system, which was added to Tier 1, Table 2.5.2-7 as well for the PMS Interlocks. Since the above logic included for the proposed containment vacuum relief system is designed with division redundancy and bypass capability, the staff concludes that the proposed containment vacuum relief system meets the applicable criteria of protection, reliability, and testability as required in 10 CFR 50.55a(h), and GDC 20 and 21. The applicant added the status of the containment vacuum relief MOVs to Table 7.5-1 for post-accident monitoring, and revised Tier 1, Tables 2.5.2-3 and 2.5.2-4 to add the automatic and manual control functions in the PMS for the containment vacuum relief MOVs. The applicant revised Tier 1, Table 2.5.2-5 and Tier 2, Table 18.12.2-1 to include the display and control of the containment vacuum relief MOVs as part of the minimum inventory for the AP1000 DCD. On the basis of its review of the above changes, the staff finds that the design of the new containment vacuum relief system proposed by the applicant meets the applicable instrumentation and control requirements as mandated in GDC 13 and 19, and also complies with the regulatory guidance in RG 1.97.

The staff finds that the proposed instrumentation and controls design changes meet the relevant requirements in 10 CFR 50.55a(h), 10 CFR 52.47, and 10 CFR Part 50, Appendix A

GDC 13, 19, 20, and 21. The proposed design changes also conform to the related guidance in RG 1.97. Therefore, the staff concludes that the instrumentation and controls aspects of the proposed containment vacuum relief system are acceptable. Markups to DCD Tier 2, Sections 7.3.1, 7.6, and 18.12.2-1, Tables 7.3-1, 7.3-3, and 7.5-1, Figures 7.2-1 (Sheet 13 of 21) and 7.2-1 (Sheet 19 of 21), and Tier 1, Tables 2.5.2-3, 2.5.2-4, 2.5.2-5, and 2.5.2-7 are being tracked as **Confirmatory Item 23-35**.

23.W.3.5 Technical Specifications

With respect to the proposed TS requirements in TS 3.3.2 and TS 3.6.10, and the proposed changes to TS 3.6.4 and 3.6.5, the staff finds these additions and changes acceptable because they conform to guidance in the Westinghouse Standard Technical Specifications (NUREG-1431), and reflect the Containment Systems design information in DCD Sections 7.3 and 9.4.7, and their associated analyses described in DCD Section 6.2. Verification that these changes are correctly incorporated in a future revision of the DCD is being tracked as **Confirmatory Item 23-36**.

23.W.4 Conclusion

On the basis of its review and confirmatory calculations, the staff finds that the external pressure analysis meets the acceptance criteria from SRP Section 6.2.1.1.A, and thus satisfies GDC 16. The proposed changes to Tier 1 Section 2.7.6 and Tier 2 Sections 6.2.1, 6.2.3, and 9.4.7 of the AP1000 DCD will be tracked as **Confirmatory Item 23-30**.

On the basis of its review of the containment isolation design aspect of the proposed vacuum relief design, the staff concludes that the design complies with the acceptance criteria in Section 6.2.4 of the SRP, including 10 CFR 50.34(f)(2)(xiv), "Additional TMI-Related Requirements," and the CSB BTP 6-4, "Containment Purging During Normal Plant Operations." The staff will track the addition of the vacuum relief design to DCD, Tier 1, Section 2.2, and Tier 2, Section 6.2.3 as **Confirmatory Item 23-31** and **Confirmatory Item 23-32**.

On the basis of its review, the staff concludes that the proposed addition of the vacuum relief valves to the already certified AP1000 containment leakage rate testing program complies with the acceptance criteria of Section 6.2.6 of the SRP. Compliance with the SRP acceptance criteria provides adequate assurance that containment leaktight integrity can be verified before initial operation and periodically throughout its service life. Compliance with the criteria in Section 6.2.6 of the SRP constitutes an acceptable basis for satisfying the containment leakage rate testing requirements of GDC 52, GDC 53, and Appendix J to 10 CFR Part 50.

The NRC staff reviewed the functional design, qualification, and IST program description for the valves to be used in the proposed AP1000 containment vacuum relief system in accordance with the regulatory basis described in this safety evaluation. Based on its review, the NRC staff concludes that Westinghouse has provided an acceptable description of the design aspects of the functional design, qualification, and IST programs for the valves to be used in the AP1000 containment vacuum relief system, pending resolution of **Confirmatory Item 23-33 and Confirmatory Item 23-34**. The NRC staff will review the operational aspects regarding the functional design, qualification, and IST programs for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

The staff finds that the proposed instrumentation and controls design changes meet the relevant requirements in 10 CFR 50.55a(h), and 10 CFR Part 50, Appendix A GDC 13, 19, 20, and 21. Markups to DCD Tier 2 Sections 7.3.1, 7.6, 18.12.2-1, Tables 7.3-1, 7.3-3, 7.5-1, Figure 7.2-1 (Sheet 13 of 21), new Figure 7.2-1 (Sheet 19 of 21), and Tier 1 Tables 2.5.2-3, 2.5.2-4, 2.5.2-5, and 2.5.2-7 are being tracked as **Confirmatory Item 23-35**.

On the basis of its review, and pending resolution of **Confirmatory Item 23-36**, the staff finds that the proposed changes to the technical specifications that have been established for the Containment Systems meet the requirements of 10 CFR 50.36**.**

In addition to the specific DCD changes discussed above in this SER section, Westinghouse proposed additional changes to several sections, tables, and figures in AP1000 DCD Tier 2 Sections 3.2, 3.7, 3.8, 3.9, 3.11, 7.2, 7.7, 18.12, 19.55, 19.59, and Appendices 1A, 3I and 9A to incorporate the design, testing, and operation of the new containment vacuum relief system and its individual components. The NRC staff has reviewed those proposed DCD changes and finds them to reflect the design, testing, and operation of the new containment vacuum relief system in an acceptable manner. The staff will confirm the incorporation of these additional DCD changes in the next revision of the AP1000 DCD. The proposed changes to the AP1000 DCD will be tracked as **Confirmatory Item 23-37.**