6 Engineered Safety Features

Chapter 6, "Engineered Safety Features," of this safety evaluation report (SER) describes the results of the review by the staff of the U.S. Nuclear Regulatory Commission (NRC or Commission), hereinafter referred to as the staff, of Chapter 6 of Korea Electric Power Corporation (KEPCO) and Korea Hydro & Nuclear Power Co., Ltd (KHNP), hereinafter referred to as the applicant, Design Control Document (DCD), for the design certification (DC) of the Advanced Power Reactor 1400 (APR1400).

6.1 Engineered Safety Features Materials

6.1.1 Engineered Safety Features Materials

6.1.1.1 Introduction

Nuclear plants include engineered safety features (ESF) to mitigate the consequences of design-basis accidents (DBAs). ESF systems for the Advanced Power Reactor (APR) 1400 include the containment systems, the safety injection system (SIS), in-containment refueling water storage tank (IRWST), habitability systems, and the fission product removal and control systems. ESF systems shall be compatible with the fluids to which they may be exposed during normal operation, maintenance, testing, and postulated accident conditions. In order to maintain the integrity of the reactor coolant pressure boundary (RCPB), ESF components that are part of, or interface with, the RCPB shall be fabricated of materials that provide a low probability of significant degradation or rapidly propagating fracture. In addition to using appropriate materials, processes for welding, non-destructive examination techniques, and processes to control cleaning, ESF systems shall be controlled to assure initial quality and prevent deterioration.

6.1.1.2 Summary of Application

Design Control Document (DCD) Tier 1: The DCD Tier 1 information associated with this section is found in DCD Tier 1 for various ESF systems including Section 2.4.2, "In-containment Water Storage System," for the IRWST, Section 2.4.3, "Safety Injection System," for the SIS, and Section 2.11.2, "Containment Spray System," for the containment spray system (CSS).

DCD Tier 2: The applicant has provided a description of materials for the ESF systems in DCD Tier 2, Section 6.1.1, "Metallic Materials," which is summarized here in part as follows:

> The applicant stated that materials used for ESF components are compatible with the core coolant and containment spray solutions. The core coolant chemistry is specified in DCD Tier 2, Table 5.2-5, "Reactor Coolant Design Specification," and is maintained by the chemical and volume control system (CVCS). The CVCS is described in DCD Tier 2, Section 9.3.4 "Chemical and Volume Control System," and evaluated by the staff in Section 9.3.4, of this safety evaluation (SE). The APR1400 does not have a SIS that can inject water into the reactor coolant system (RCS) at the design pressure of 2250 pounds per square inch gauge (psig) (15.610 MPa), as shown in DCD Tier 2, Table 6.3.2-4, "Safety Injection System Flow Delivery to RCS." The introduction of core coolant into the ESF system would occur if the pilot operated safety relief valves (POSRV) released steam into the IRWST or in the event of a loss-of-coolant accident (LOCA).

The water source for safety injection (SI) and containment spray is the IRWST. In addition to providing a heat sink for the POSRV, the IRWST also provides water for filling the refueling pool and the source of water for the core flooding system. The chemistry of the IRWST is maintained by the spent fuel pool cooling and cleanup system and, if necessary, the water chemistry can be adjusted by the CVCS. The in-containment water storage system, which includes the IRWST, is described in DCD Tier 2, Section 6.8, "In-containment Water Storage System." Following a LOCA, the pH of recirculated water is maintained by tri-sodium phosphate (TSP), which is stored in the holdup volume tanks. Control of pH is described in DCD Tier 2, Section 6.5.2, "Containment Spray Systems," and is evaluated by the staff in Section 6.5.2 of this SE.

Material specifications for ESF components are provided in DCD Tier 2, Tables 6.1-1, "Principal Engineered Safety Feature Pressure Retaining Material Specifications," and 6.1-2, "Principal Engineered Safety Features Materials Exposed to Core Coolant and Containment Spray." The applicant committed to meeting the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, (henceforth referred to as the ASME Code), Section III, Articles NB/NC/ND/NE-2000.

Austenitic stainless steel components are of the general types 304, 316, 304L, 316L. Cast austenitic stainless steels are of the grades CF3, CF8, CF3M, and CF8M. Welding materials for these components are specified as E/ER308, E/ER309, E/ER308L, and E/ER309L.

Ferritic components are fabricated to the following specifications and grades:

Component Materials	Bolt Materials	Nut Materials
SA-105	SA-193 Gr. B16	SA-194 Gr. 2
SA-106 Gr. B	SA-193 Gr. B6	SA-194 Gr. 2H
SA-106 Gr. C	SA-193 Gr. B7	SA-194 Gr. 4
SA-182 F22 Class 3	SA-193 Gr. B8	SA-194 Gr. 6
SA-226 Gr 2	SA-638 Gr. 660	SA-194 Gr. 7
SA-333 Gr. 6	SA-564 Type 630 Cond. H1100	SA-194 Gr. 8
SA-350 Gr. LF1		SA-194 Gr. 8M
SA-350 Gr. LF2		SA-194 Gr. 16
SA-508 Gr. 3 Cl. 1		SA-638 Gr. 660
SA-516 Gr. 60		SA-564 Type 630 Cond. H1100
SA-516 Gr. 70		

Welding materials for ferritic components are specified as SFA-5.1 E7016, SFA-5.1 E7018, SFA-5.5 E9018-B3, SFA-5.18 ER70S-2, SFA-5.18 ER70S-6, and SFA-5.28 ER90S-B3.

Austenitic stainless steels used in ESF systems are protected from sensitization by limiting welding heat input, limiting interpass temperatures during welding, using austenitic stainless steels with a maximum carbon content of 0.065 percent, and limiting the use of cold-worked stainless steel.

The applicant prevented failure due to stress corrosion cracking (SCC) by:

(1) Utilizing augmented in-service inspection for stainless steel components that are cold worked,

(2) Imposing chemical restrictions on cleaning agents for austenitic stainless and ferritic steels,

(3) Specifying purity controls for water used in cleaning austenitic stainless and ferritic steel components,

(4) Conforming to the requirements of ASME NQA-1, 2008 edition and 2009 addendum, “Quality Assurance Requirements for Nuclear Facility Applications,” with the conditions set forth in Regulatory Guide (RG) 1.28, “Quality Assurance Program Criteria (Design and Construction),”

(5) Selecting of non-metallic insulation in accordance with RG 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,”

(6) Specifying restrictions for grinding and buffing wheels used in stainless steel component fabrication.

Austenitic stainless steels that are subjected to sensitizing conditions will be tested in accordance with ASTM A262, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” Practices A and E. Weld material for austenitic stainless steels will meet the delta ferrite content requirements of RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal.”

The applicant stated that cold working of austenitic stainless steel is avoided. If cold work is necessary, the process will be documented and the component will be subject to augmented in-service inspection.

Cleanness of ESF components will be controlled in accordance with ASME NQA-1, Subpart 2.1, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants.” The applicant will use acetone, isopropyl alcohol, and trisodium phosphate as degreasing solvents. The water used for cleaning will meet the requirements of ASME NQA-1, Subpart 2.1, Table 3.4.1, “Water Requirements,” and will be inhibited with 30-100 ppm hydrazine to prevent halide-induced intergranular corrosion. These provisions are applied to both austenitic stainless and ferritic steels.

Welding of ferritic steel ESF components will meet the requirements of RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel.” Conformance to this RG includes minimum preheat temperatures prior to welding, maximum interpass temperatures during welding, and moisture control for low hydrogen welding materials.

Welder qualification for areas of limited accessibility will conform to RG 1.71, “Welder Qualification for Areas of Limited Accessibility.”

6.1.1.3 Regulatory Basis

The staff reviewed APR1400 DCD Tier 2, Section 6.1.1, in accordance with U.S. Nuclear Regulatory Commission (NRC), NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," (hereafter referred to as the SRP) Section 6.1.1, "Engineered Safety Features Materials," Revision 2. In the APR1400 DCD, Tier 2, Section 6.1.1, the applicant described the selection, fabrication, and compatibility of materials with core cooling water and containment sprays for ESF systems. The staff based its review of DCD Tier 2, Section 6.1.1 and its acceptance criteria on the relevant requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, "Codes and Standards," 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 1, "Quality standards and records," GDC 4, "Environmental and dynamic effects design bases," GDC 14, "Reactor coolant pressure boundary," GDC 31, "Fracture prevention of reactor coolant pressure boundary," GDC 35, "Emergency core cooling," and GDC 41, "Containment atmosphere cleanup."

- GDC 1 and 10 CFR 50.55a(a)(1), require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions they perform.

- GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (e.g., LOCAs).

- GDC 14 requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 31 requires that the design of the RCPB include sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, it will behave in a non-brittle manner and the probability of rapidly propagating fracture will be minimized.

- GDC 35 requires a system to provide abundant emergency core cooling. GDC 35 also requires that, during activation of the system, clad metal-water reaction will be limited to negligible amounts.

- GDC 41 requires that the design provide containment atmosphere cleanup systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment. The staff limited its review of the ESF structural materials to ensuring that they meet the requirements of GDC 41 with respect to corrosion rates related to hydrogen generation in post-accident conditions.

- The regulation in 10 CFR Part 50, Appendix B, mandates that applicants establish quality assurance (QA) requirements for the design, construction, and prevention or mitigation of the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

6.1.1.4 Technical Evaluation

The staff reviewed the information provided in the DCD to determine if the ESF systems comply with the SRP Section 6.1.1 and the applicable requirements of GDC 1, GDC 4, GDC 14, GDC 31, GDC 35, GDC 41, and 10 CFR Part 50, Appendix B. After the acceptance review, the staff reviewed Revision 0 and Revision 1 of the APR1400 DCD and the referenced technical reports.

6.1.1.4.1 Materials and Fabrication

To meet the requirements of GDC 1 and 10 CFR 50.55a, SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the safety functions to be performed. An applicant shall meet this requirement by utilizing material specifications for the ESF components in accordance with ASME Code, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” ASME Code, Section III, Division 2, “Code for Concrete Containments,” RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” or an alternative meeting the requirements of 10 CFR 50.55a(z).

The applicant has committed to using the ASME Code, Section III as the code of construction for ESF components that are important to safety. The general description of ASME Code compliance is described in DCD Tier 2, Section 5.2.1, “Conformance with Codes and Code Cases,” and is referenced in DCD Tier 2, Section 6.1.1.

The base material specifications in DCD Tier 2, Tables 6.1-1 and 6.1-2 have the prefix of “SA-,” and weld material specifications in Tables 6.1-1 and 6.1-2, have the prefix of “SFA-.” These prefixes indicate that the base metal and weld filler materials used in the ESF systems will be certified to the requirements of the ASME Code, Section II. The staff compared the list of specifications in Tables 6.1-1 and 6.1-2 to DCD Tier 2, Section 3.2, “Classification of Structures, Systems, and Components,” Table 3.2-1, “Classification of Structures, Systems, and Components.” The staff concluded that Tables 6.1-1 and 6.1-2 were consistent with the design criteria specified in Table 3.2-1.

The staff issued Request for Additional Information (RAI) 403-8454, Question 06.01.01-1, dated February 10, 2016 (ML16041A094) requesting the applicant to clarify how the material chosen for the IRWST and Holdup Volume Tank conforms to the code of construction described in DCD Tier 2, Table 3.2-1. In its response to RAI 403-8454, Question 06.01.01-1, dated April 27, 2016 (ML16118A106), the applicant stated that the design of the IRWST liner complied with ASME Code, Section III, Division 2, with the exception of the QA and material testing requirements in ASME Code, Section III, CC-2000. On September 12, 2016, the staff held a public meeting with the applicant to discuss the safety functions of the IRWST liner and how the applicant has confidence that the liner will perform the design functions. During the public meeting the applicant agreed to supplement the QA and material testing requirements by requiring conformance with ASME NQA-1 and 10 CFR Part 50, Appendix B. The staff finds that conformance with the NQA-1 and Appendix B requirements provide sufficient assurance that the liner material meets the material specification and ensures that the material is controlled in a sufficient manner considering its safety function. Because the DCD Tier 2 has not been updated to reflect this information, **RAI 403-8454, Question 06.01.01-1, is being tracked as a confirmatory item.**

Inherently, the commitment to meet ASME BPV Code, Section III will also require conformance to the fracture toughness requirements contained in Subarticles NB/NC/ND/NE-2300. The SRP states that conformance to Subarticles NB/NC/ND/NE-2300 is an acceptable method to demonstrate adequate fracture toughness. The applicant will not use any ASME BPV Code Cases for ESF components and, as such, all materials in the ESF system will conform to the requirements of ASME Code, Section III and will have sufficient fracture toughness to prevent rapid propagating failure or brittle fracture. The material specifications, types, and grades that have been selected for the ESF components are consistent with ESF and RCS components in operating reactors. Where applicable, Alloy 690/52/152 is used instead of Alloy 600/82/182 due to SCC operational experience. Controls for welding Alloy 52/152 are documented in DCD Section 3.6.3 “Leak-Before-Break Evaluation Procedures,” which is referenced in DCD Section 6.1.1.

The applicant committed to meeting RG 1.71., “Welder Qualification for Areas of Limited Accessibility.” The commitment to this RG conforms to welding practices which are accepted by the staff as described in the SRP.

6.1.1.4.2 Austenitic Stainless Steel

To meet the requirements of GDC 4 and GDC 14, SSCs important to safety shall be compatible with the operating environment (including normal, maintenance, testing, and accident scenarios), and the ESF SSCs shall not impact the extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture of the RCPB. The regulations 10 CFR Part 50, Appendix B require that the applicant include QA controls for safety-related components, which provide adequate assurance that SSCs will perform satisfactorily in service.

The applicant provided supplemental information stating that austenitic stainless steel ESF components are not designed with additional thickness to compensate for erosion, corrosion, abrasion, or other environmental effects. Additional corrosion allowance for austenitic stainless steel components is not necessary because the ESF systems are normally in a standby, non-operating mode. Because the ESF components do not experience significant flow of water on a regular basis, erosion and abrasion are not significant concerns. The stagnant water conditions in the ESF system will result in a stable passive oxide layer on the inside of the ESF piping, which eliminates the need for a general corrosion allowance thickness. The water chemistry of the ESF systems is controlled to ensure that the water conditions necessary for corrosion (e.g., the presence of chlorides, pH control, etc.) are precluded. The controls on water chemistry are evaluated by the staff in Section 6.1.1.4.4, “Composition and Compatibility of ESF Fluids,” of this SE.

Provided that sufficient water chemistry controls are established and maintained, austenitic stainless steel components in the ESF systems do not require a corrosion allowance. Additionally, because the piping systems are designed in accordance with ASME Code, Section III, there exists significant safety margin in the piping design. The ASME Code safety margin ensures that normal inspection of the ESF systems is sufficient to detect pipe thinning prior to failure of an ESF component.

The staff’s review focused on five topics that can impact the integrity of austenitic stainless steels: intergranular failure due to sensitization, thermal embrittlement, cracking caused by low delta ferrite levels, cold work, and contamination of materials.

The staff’s guidance for the prevention of austenitic stainless steel sensitization is given in RG 1.44, “Control of the Processing and Use of Stainless Steel.” DCD Tier 2, Section 6.1.1 states that stainless steel controls for ESF components is described in DCD Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.”

As stated in RG 1.44, austenitic stainless steel casting grades (such as CF3, CF8, CF3M, and CF8M) and austenitic stainless steel weld materials (such as E308/L and E309/L) with sufficient delta ferrite are not susceptible to sensitization. Austenitic stainless steel components manufactured to specifications SA-240, SA-182, SA-312, SA-213, SA-358, and SA-479 are required by the ASME Code to undergo a heat treatment sufficient to dissolve carbides and quenching, which is sufficient to prevent carbide formation during cooling. The ASME Code requirements provide sufficient basis that the base material and components will be supplied in an annealed state and free from sensitization.

DCD Tier 2, Section 5.2.3, describes testing that is necessary to verify if sensitization has occurred during fabrication. Components that are inadvertently subjected to, or suspected of being subjected to, temperatures between 800-1500 degrees Fahrenheit (°F) (427-816 degrees Celsius (°C)) will require testing in accordance with ASTM A262, Practices A or E. Retesting the material using ASTM A262 is consistent with RG 1.44 and would confirm the existence and extent of sensitization. DCD Tier 2, Section 5.2.3.4.1, "Avoidance of Stress Corrosion Cracking of Nuclear Steam Supply System Components," states that "[s]olution heat treatment is not performed on completed or partially fabricated components. Rather, the extent of chromium carbide precipitation is controlled during all stages of fabrication as described in Items b, c, and d below." As such, material that subsequently fails ASTM A262, Practice E would not be used.

Although weld material (with sufficient delta ferrite) is not susceptible to sensitization, operating experience has demonstrated that the heat affected zones of welds may experience significant sensitization. To prevent sensitization during welding, the applicant will: limit heat input to 60 kJ/in (23.6 kJ/cm), limit interpass temperature to 350 °F (177 °C), and limit carbon content to 0.065 percent. The limits on heat input and interpass temperatures are consistent with recommendations to prevent sensitization described in RG 1.44. The staff noted that the carbon content limitation was not the same as the guidance in RG 1.44.

In RAI 8351, Question 05.02.03-13, dated December 14, 2015 (ML15348A252), for DCD Tier 2, Section 5.2.3, the staff requested more information on how the 0.065 percent carbon limit prevented sensitization. On February 10, 2016, the staff issued RAI 403-8454, Question 06.01.01-3, dated February 10, 2016 (ML16041A094), requesting a rationale for a 0.065 percent carbon content in the ESF system. On May 8, 2017, (ML17128A420), the applicant submitted its response to RAI 403-8454, Question 06.01.01-3, supplementing DCD Tier 2 with a requirement to qualify all welding processes for austenitic stainless steels with carbon content above 0.03 carbon with ASTM A262 testing. The staff finds the applicant's response acceptable because testing during the weld procedure qualification in accordance ASTM A262 Practices A or E is endorsed in RG 1.44 and is sufficient to provide assurance that the fabrication process will not sensitize austenitic stainless steel. Therefore, **RAI 403-8454, Question 06.01.01-3, is being tracked as a confirmatory item.**

Operational experience with cast austenitic stainless steel components has shown that the base material can lose ductility when exposed to high temperatures and/or significant radiation fields over extended periods of time. Staff guidance on the embrittlement of cast austenitic stainless steel is found in an NRC letter (Grimes Letter), dated May 19, 2000 (ML003717179). The Grimes Letter is described as an acceptable method of meeting NRC regulations in the Generic Aging Lessons Learned (GALL) Report (NUREG-1801), Aging Management Plan (AMP) XI.M12. The Grimes Letter specifies screening criteria based upon radiation damage, design temperatures, fabrication methods, and delta ferrite content. By following the guidance in the Grimes Letter a designer may eliminate a material failure mode by specifying requirements on the fabrication of components. Cast austenitic stainless steel components in the ESF systems are not susceptible to radiation embrittlement due to the distance between the components and the reactor vessel. In the response to RAI 335-8351, Question 05.02.03-8, dated January 13, 2016 (ML16013A482), the applicant provided a rationale for screening thermal embrittlement of cast austenitic stainless steel components based upon a 260 °C (500 °F) maximum operating temperature. The Grimes Letter specifies that a screening temperature of 251 °C (484 °F) may be used to categorize components as potentially susceptible or not susceptible to thermal embrittlement. While there is a discrepancy between the applicant's screening criteria (260 °C) and the Grimes Letter (251 °C), the discrepancy does not impact the ESF system. There are two ESF components that use austenitic stainless steel castings: the SIP and the containment spray pump. The SIP has a design temperature of 177 °C (350 °F) as stated in DCD Tier 2, Table 6.3.2-1, "SIS Component Parameters," and the containment spray pump has a design temperature of 204 °C (400 °F) as stated in DCD Tier 2, Table 6.2.2-2, "Containment Spray System Design Parameters." The design temperatures for the ESF system is significantly lower than the screen criteria temperatures in both the Grimes Letter and the applicant's RAI response. As such, the cast austenitic stainless steel components in the ESF system will not experience either thermal or radiation embrittlement.

The applicant stated that welding material for austenitic stainless steel will meet the delta ferrite guidance set forth in RG 1.31. Conformance to this RG is sufficient to prevent hot cracking of austenitic stainless steel weldments and meets the acceptance criteria in the SRP.

In DCD Tier 2, Section 6.1.1 the applicant committed to avoiding cold work of austenitic stainless steel materials. If cold work is performed, the cold work shall be identified and described and the component which is cold worked shall be subject to augmented in-service inspection. Implementation of the augmented in-service inspection program will be the responsibility of a combined license (COL) holder as is required by COL Information Item (COL Item) 6.6(4).

Surface contamination of austenitic stainless steels can introduce chemicals that are known to cause corrosion. SRP Section 6.1.1 recommends cleaning procedures and processes based on RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." However, RG 1.37 has been withdrawn by the staff (ML13345A259) and guidance is now found in RG 1.28, "Quality Assurance Program Criteria (Design and Construction)." RG 1.28 endorses the ASME nuclear QA program (QAP) standard NQA-1 including the cleanness and contamination controls specified in ASME NQA-1, Subpart 2.1. The applicant committed to a QAP in conformance with ASME NQA-1 and RG 1.28. The selection of degreasing solvents, the quality of water, and inhibiter selection is consistent with the guidance in ASME NQA-1-year, Subpart 2.1. Additionally, in DCD Section 6.1.1.1, "Materials Selection and Fabrication," the applicant committed to "protection… for contamination during fabrication, shipment, and storage as recommended in ASME NQA-1." The commitment to ASME NQA-1 includes meeting the requirements for shipping and handling as stated in ASME NQA-1, Subpart 2.2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants" and housekeeping stated in ASME NQA-1, Subpart 2.3, "Quality Assurance Requirements for Housekeeping for Nuclear Power Plants." The staff concludes that the cleanness and contamination controls during component fabrication, installation, and construction are sufficient to prevent the initiation of corrosion of ESF materials because the applicant committed to meeting ASME NQA-1 consistent with staff guidance and the SRP.

6.1.1.4.3 Ferritic Steels

To meet the requirements of GDC 4 and GDC 14, SSCs important to safety shall be compatible with the operating environment (including normal, maintenance, testing, and accident scenarios) and the ESF SSCs shall not impact the extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture of the RCPB. The regulations in 10 CFR Part 50, Appendix B require that the applicant include QA controls for safety related equipment that provides adequate assurance that SSCs will perform satisfactorily in service.

Ferritic steel components have the general classifications of carbon steels and low alloy steels (including chrome-moly steels). The material specifications types and grades for the ESF components in the APR1400 design are consistent with similar components and systems in operating reactors. As such, the staff's review focused on applicable operational experience in the current operating fleet and regulatory guidance related to this material.

The applicant committed to good welding practices for preheat and moisture controls of ferritic steels. Specifically, the applicant committed to meeting:

(1) RG 1.50 and ASME Code, Section III, Appendix D for minimum preheat;

(2) ASME Code requirements of Articles NB/NC/ND-2000 and -4000 for moisture control.

The SIP for the APR1400 design includes a low-alloy steel forging with stainless steel cladding. The staff reviewed RG 1.42, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," and determined that the guidance regarding cracking is not applicable because the applicant utilizes SA-508 Grade 3, Class 1 material. RG 1.42 applies to SA-508 Grade 2, Class 1 material for which underclad cracking is a significant concern.

The staff concludes that GDC 1 and GDC 4 are met because the ferritic steel components proposed by the applicant are consistent with components and systems in operating plants and the applicant adequately addressed historical degradation of ferritic materials and takes appropriate controls to prevent degradation.

The applicant committed to the same cleanness controls on the ferritic steel materials as the austenitic stainless steel materials. The cleanness and contamination controls will meet the requirements found in ASME NQA-1, Subpart 2.1. The applicant committed to meeting ASME NQA-1, Subparts 2.2 and 2.3, which relate to controls on shipping, handling, and housekeeping of SSCs. Based upon the information provided by the applicant, the staff concludes that the cleanness and contamination controls are sufficient to prevent the initiation of corrosion of ferritic steel ESF components caused by contamination during component fabrication, installation, and construction.

6.1.1.4.4 Composition and Compatibility of ESF Fluids

In DCD Tier 2, Section 6.1.1.2, "Composition and Compatibility of Core Cooling Coolants and Containment Sprays," the applicant describes the water chemistry controls for the RCS and the ESF systems. Water chemistry controls for the RCS are evaluated by the staff in Section 5.2.3 of this report. The ESF systems will comply with the Electric Power Research Institute (EPRI) "PWR Primary Water Chemistry Guidelines (EPRI Report 1014986)." Appendix B of the EPRI guidelines contain water chemistry parameters and best practices for refueling water storage tanks which is applicable to the IRWST in the APR1400 design. Sampling of the IRWST will be conducted on a weekly basis to ensure that the levels of chloride, fluoride, boron, and sulfate as well as the pH conform to the values described in the EPRI Guidelines. The staff has reviewed the applicant's conformance with the EPRI Guidelines in SE section 9.3.4 and concludes that the use of the EPRI Primary Water Chemistry Guidelines is an acceptable method to prevent degradation of SSCs in the RCS and ESF system. Weekly monitoring of the water chemistry is sufficient to ensure that material degradation is not occurring.

The applicant stated that zinc and mercury are excluded from the ESF systems. Zinc is excluded to preclude chemical reactions that produce hydrogen, and mercury is excluded to preclude possible interactions with aluminum, stainless steel, Alloy 690, and other materials containing copper.

In order to reduce the probability of SCC of austenitic stainless steel components, containment and core coolants should be maintained at a pH level of at least 7.0. DCD Tier 2, Section 6.5.2, "Containment Spray Systems," states that the pH of ESF fluids is controlled during a DBA using TSP baskets as a buffering agent, which results in a post-LOCA pH of at least 7.0. Control of pH in ESF fluids is consistent with Branch Technical Position (BTP) 6-1, "pH for Emergency Coolant Water for Pressurized Water Reactors" and, therefore, is acceptable to the staff. The staff also performed a confirmatory calculation of the containment water pH following a LOCA that confirmed that the pH will be adjusted to a value greater than 7.0. The acceptability of the applicant's calculation of the post-accident pH is addressed in SER Section 6.5.2 and 15.0.3, "Design Basis Accident Radiological Consequence for Advanced Light Water Reactors."

6.1.1.5 Combined License Information Items

DCD Tier 2, Section 6.6.9, "Combined License Information," describes the information that shall be submitted to the staff by a COL applicant referencing the APR1400 design. This information is duplicated in DCD Tier 2, Table 1.8-2, "Combined License Information Items."

None of the COL items in DCD Tier 2 pertain to the selection of materials for the ESF system.

6.1.1.6 Conclusion

Based on the analysis above, the staff finds that the selection of materials for the ESF system are commensurate with the safety-related functions that the system performs. The staff finds that the fabrication and process controls for austenitic stainless steels, ferritic steels, and nickel alloys are sufficient to ensure that the system will perform its safety functions during all normal, off-normal, and accident conditions. The staff finds that the water chemistry controls and the material selection is sufficient to prevent degradation of the system in service. The staff finds that the selection of materials in the ESF system provides a low probability of rapid failure for the ESF system. The staff finds that the selection of materials in the ESF system supports the control of hydrogen in containment during accident conditions. Based upon these findings the staff concludes that the material selection for the ESF system meets the criteria in the SRP and provides adequate assurance that the ESF system design meets NRC regulations.

6.1.2 Protective Coatings Systems (Paints) - Organic Materials

6.1.2.1 Introduction

Organic and inorganic coatings are used in the APR1400 plant to provide corrosion protection or facilitate surface decontamination. In some locations, coating failure can be a source of debris that could prevent the Emergency Core Cooling System (ECCS) from performing its safety-related function. Coating degradation could result from conditions such as physical wear, erosion from a high-energy spray jet (e.g., LOCA), chemical attack, high temperature, radiation, or application deficiencies. Conditions causing degradation can be present during operation, maintenance, or accident conditions.

A coatings program that addresses classification and quality requirements for coatings provides assurance that coatings will perform as expected under operating and accident conditions. The coatings program includes tracking of the coating materials used in containment and evaluation of their potential interactions with the ECCS.

Other organic materials may be used but are not evaluated as part of this topic. Combustible gas and organic iodides could be generated by decomposition of organic materials, and these topics are evaluated in Section 6.2.5, “Combustible Gas Control in Containment” and Section 6.5.2, “Containment Spray System (CSS),” of this SE.

6.1.2.2 Summary of Application

DCD Tier 1: The only Tier 1 entry for this topic is for Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC), as described below.

DCD Tier 2: This summary of the DCD is based on revisions that the applicant proposed in RAI responses, including the division into Sections 6.1.2.1 and 6.1.2.2. The staff will confirm these changes in the next revision of the DCD. The details of the RAI and applicant’s responses are described in the staff’s technical evaluation in Section 6.1.2.4 below. DCD Tier 2, Revision 1, Section 6.1.2, “Protective Coatings and Organic Materials,” describes the use of protective coatings and other organic materials inside containment, coatings outside containment that could affect the operation of safety systems, and coatings outside containment that would not affect the operation of safety systems. DCD Tier 2, Section 6.1.2 is divided into Section 6.1.2.1, “Protective Coatings,” and Section 6.1.2.2, “Organic Materials.” Section 6.1.2.1 describes the types of coating systems according to substrate material and temperature and it lists the definitions for Service Level (SL) I, II, and III coatings. Section 6.1.2.2 describes the organic materials other than protective coatings, including lubricant for the reactor coolant pump (RCP) and cable insulation and jacket materials. Three COL items are included in order to identify non-conforming coatings in containment, describe the coatings program description and milestones, and identify the amount of organic cable insulation and jacket material in containment.

ITAAC: There is one ITAAC item related to this review topic. That item is evaluated as part of the In-containment Water Storage System (IWSS), DCD Tier 1, Section 2.4.2, Item 9.c.iv in Table 2.4.2-4, “In-containment Water Storage System ITAAC.” The purpose of the ITAAC is to confirm that the as-built coatings for the IRWST are consistent with the analyses of coatings as a debris source. Those analyses are primarily conducted in the evaluation of ECCS suctions strainers and long-term cooling, but they depend, in part, on the coating materials information in DCD Section 6.1.2. For example, the coating types and SL classifications described in DCD Section 6.1.2 are inputs to the coating debris evaluation. The staff finds this ITAAC acceptable because it confirms that the coating materials and the quality assurance applied to the coatings meet the requirements in DCD Section 6.1.2 consistent with SRP 6.1.2.

Technical Specifications (TS): There are no TS for this review topic.

6.1.2.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 6.1.2, “Protective Coating Systems (Paints) – Organic Materials,” of NUREG–0800, and are summarized below. Review interfaces with other SRP Sections are listed in SRP Section 6.1.2.

- The regulations in 10 CFR Part 50, “Domestic Licensing of Product and Utilization Facilities,” Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” as it relates to the QA requirements for the design, fabrication, and construction of safety-related SSCs.

- The regulations in 10 CFR 52.47(b)(1), requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC regulations.

Acceptance criteria adequate to meet the above requirements include:

- A coating system to be applied inside a containment is acceptable if it meets the regulatory positions of RG 1.54, Revision 2, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” and the criteria of ASTM International (ASTM) standards D5144 and D3911,“Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants” and “Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design-Basis Accident (DBA) Conditions,” respectively.

6.1.2.4 Technical Evaluation

The staff reviewed DCD Tier 2, Section 6.1.2, “Organic Materials,” in accordance with SRP Section 6.1.2, “Protective Coating Systems (Paints) – Organic Materials,” Revision 3, issued March 2007. The staff also reviewed supplemental information provided in the applicant’s response to RAI 305-8375, Question 06.01.02-1, dated December 3, 2015 (ML15337A125). The purpose of the staff’s evaluation was to determine if the design meets the requirements of 10 CFR Part 50, Appendix B as it relates to the QA requirements of the design, fabrication, and construction of safety-related SSCs. The review was based on whether the design conforms to the guidance in RG 1.54, Revision 2. The failure of coatings can prevent safety-related SSCs from performing their safety-related functions. In addition, coatings could decompose under conditions following a LOCA to form combustible gas or radioactive organic iodides. Potential solid debris effects from coatings are evaluated in Section 6.2.2, “Containment Heat Removal Systems,” in this SE. The effects and control of combustible gases and organic iodides are reviewed under Sections 6.2.5 and 6.5.2, respectively, in this report.

In RAI 305-8375, Question 06.01.02-1, dated November 12, 2015 (ML15316A231), the staff requested that the applicant revise the original title of DCD Section 6.1.2, “Organic Materials,” to accurately reflect the scope, which includes coatings and other organic materials. In its response to RAI 305-8375, Question 06.01.02-1, dated December 3, 2015 (ML15337A125), the applicant proposed “Protective Coatings and Organic Materials” as the Section 6.1.2 title. The applicant’s response also proposed a new DCD Tier 2, Section 6.1.2.1, “Protective Coatings.” The staff determined that the proposed changes are acceptable because the titles accurately describe the content in DCD Tier 2, Section 6.1.2. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-1. Therefore, RAI 305-8375, Question 06.01.02-1 is resolved and closed.

In the original submittal, the applicant stated that no organic materials will be used in containment. However, because some coatings to be used are organic materials, in RAI 305-8375, Question 06.01.02-2, dated November 12, 2015, the staff requested that the applicant provide additional descriptions regarding the organic materials in containment. In its response to RAI 305-8375, Question 06.01.02-2, dated December 3, 2015 (ML15337A125), the applicant proposed a new DCD Tier 2, Section 6.1.2.2, “Organic Materials,” and COL Information Item 6.1(3) to describe the organic materials in containment other than coatings. These materials are cable jacketing, cable insulation, and RCP lubricating oil. The proposed DCD revision includes Table 6.1-4, “Other Organic Materials inside Containment,” which lists the materials and approximate amount of the lubricating oil. The amount of cable insulation and jacketing is not known in advance, so COL Information Item 6.1(3) requires a COL applicant to determine the amount of organic cable insulation and jacket material. The staff determined that the proposed DCD Tier 2, Table 6.1-4, and COL Information Item 6.1(3) are acceptable because they describe the organic materials expected to be in containment other than coatings. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-2. Therefore, RAI 305-8375, Question 06.01.02-2 is resolved and closed.

In DCD Section 6.1.2.1, the applicant stated that the protective coatings used inside containment are designated SL I, meet the guidance in RG 1.54, Revision 2, and have been demonstrated to withstand design basis conditions. In RAI 305-8375, Question 06.01.02-4, dated November 12, 2015, the staff requested that the applicant add the word “accident” to clarify that demonstration to be with respect to DBA conditions. In its response to RAI 305-8375, Question 06.01.02-4, dated December 3, 2015 (ML15337A125), the applicant proposed to include this correction. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-4. Therefore, RAI 305-8375, Question 06.01.02-4 is resolved and closed. In RAI 305-8375, Question 06.01.02-3, dated November 12, 2015, the staff asked the applicant to clarify the use of coatings and substrates in containment. In its response to RAI 305-8375, Question 06.01.02-3, dated December 3, 2015, the applicant proposed clarifying changes to DCD Tier 2, Section 6.1.2 and Table 6.1-3, “Coating Material Used in the Containment Structure,” which identify the types of substrates that will be coated and the types of coatings that will be selected for them. The applicant’s response stated that concrete will be coated with epoxy primer, surface, and finish coat. Inorganic zinc (IOZ) primer with an epoxy finish coat will be used for steel structures, including the containment liner plate, equipment hatch, and personnel airlock. Epoxy primer and epoxy finish coat will be used for steel equipment and components if the surface temperature is less than 93.3 ºC (200 ºF). IOZ primer without a finish coat will be used on equipment and components if the surface temperature is between 93.3 ºC (200 ºF) and 399 ºC (750 ºF). Stainless steel, galvanized steel, and machined or wearing surfaces are not to be coated. The staff determined that the proposed revisions are acceptable because they clarify the use of coating materials and substrates in the containment, enabling analyses of the suitability of containment coatings for DBA conditions in accordance with SRP Section 6.1.2. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-3. Therefore, RAI 305-8375, Question 06.01.02-3 is resolved and closed.

In its response to RAI 305-8375, Question 06.01.02-4, dated December 3, 2015 (ML15337A125), applicant proposed DCD Tier 2 revisions in new Section 6.1.2.1 to define the use of SL I, II, and III coatings with respect to the positions in RG 1.54, Revision 2. The staff finds these revisions acceptable because they meet the positions of RG 1.54, Revision 2. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-4. Therefore, RAI 305-8375, Question 06.01.02-4 is resolved and closed. The applicant stated that selection of SL I coating is based on ASTM Standards D5144-08, D3911-08, and D3843-08, "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities." The versions of the ASTM standards are different than those endorsed by the staff through RG 1.54, Revision 2. The differences include the acceptance criteria used to evaluate whether the coatings withstand DBA conditions. The 2008 revision of D3911 (D3911-08) eliminated the specific acceptance criteria to allow the user to specify the criteria. In RG 1.54, Revision 2, the staff accepted D3911-08 with the exception that licensees and applicants should meet the following acceptance criteria or criteria that are more stringent:

- Peeling and delamination shall not be permitted

- Cracking is not considered a failure unless it is accompanied by delamination or loss of adhesion

- Blisters shall be limited to intact blisters that are completely surrounded by sound coating bonded to the surface

Therefore, on November 12, 2015, the staff issued RAI 305-8375, Question 06.01.02-4, requesting that the applicant address the differences and confirm that the acceptance criteria for ASTM D3911-08 will meet the regulatory positions of RG 1.54, Revision 2. In the same question the staff requested that the applicant identify the DBA conditions used to qualify the coatings. As part of the response to RAI 305-8375, Question 06.01.02-4, dated December 3, 2015 (ML15337A125), in the proposed DCD Section 6.1.2.1, the applicant stated that the qualification of SL I coatings will meet the minimum acceptance criteria from RG 1.54, Revision 2, and that they would be qualified based on DBA conditions for a LOCA and MSLB combined environment. The staff finds this acceptable because it meets the positions in RG 1.54, Revision 2. The same response also proposed an editorial correction to the notation in the DCD for ASTM D3843, "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities," to change the notation "ASTM D3843-08" to "ASTM D3843-00 (reapproved 2008)." The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-4. Therefore, RAI 305-8375, Question 06.01.02-4 is resolved and closed.

Some components for containment are coated by the manufacturer. Therefore, on November 12, 2015, the staff issued RAI 305-8375, Question 06.01.02-5, asking the applicant how SL I will be ensured for coatings inside containment on manufactured components that cannot be procured with coatings meeting SL I requirements. In its response to RAI 305-8375, Question 06.01.02-5, dated December 3, 2015 (ML15337A125), the applicant stated that all purchase specifications will require that all structures, equipment, and components inside the containment building be procured with coatings meeting the SL I requirements. The response included proposed DCD revisions to require COL applicants to manage any non-conforming coatings according to ASTM D7491-08, "Standard Guide for Management of Non-Conforming Coatings in Coating Service Level I Areas of Nuclear Power Plants." The revisions included adding this requirement as COL 6.1(1) and changing the number of the original COL 6.1(1) to COL 6.1(2). The staff finds this acceptable because the intent is to use only SL I coatings inside containment and COL applicants will manage any exceptions according to an industry standard

(ASTM D7491-08) accepted for this purpose by the staff in RG 1.54, Revision 2. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-5. Therefore, RAI 305-8375, Question 06.01.02-5 is resolved and closed.

The applicant did not address coatings other than those inside containment in DCD Section 6.1.2. Since RG 1.54, Revision 2, addresses coatings other than SL I, including coatings outside containment that could affect safety systems (SL III), the staff requested, in RAI 305-8375, Question 06.01.02-4, dated November 12, 2015, that the applicant address the coatings other than SL I. In its response to RAI 305-8375, Question 06.01.02-4, dated December 3, 2015 (ML15337A125), the applicant addressed this RAI as part of the proposed new DCD Section 6.1.2.1, "Protective Coatings." This new section is divided into three parts that define the coating SLs and their associated requirements. The staff finds this acceptable because the proposed revision addresses both SL II and SL III coatings, and the definitions are consistent with those in RG 1.54, Revision 2. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-4. Therefore, RAI 305-8375, Question 06.01.02-4 is resolved and closed.

The applicant stated that the coatings program to monitor effectiveness of maintenance for SL I coatings includes materials, storage, equipment, application, inspection, and training as required by the Maintenance Rule, 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." The applicant proposed a COL item, COL 6.1(1), for providing the implementation milestones of the coatings program, but the applicant did not provide a description of the coatings program. In RAI 305-8375, Question 06.01.02-6, dated November 12, 2015, the staff requested that the applicant change the COL item to include a description of the coatings program and expand the scope to all safety-related coatings. In its response to RAI 305-8375, Question 06.01.02-6, dated December 3, 2015 (ML15337A125), the applicant renumbered this COL item to COL 6.1(2) and revised it to require the COL applicant to both describe the coating program and identify the implementation milestones. The applicant also revised the paragraph in DCD Tier 2, Section 6.1.2 that describes the basis for this COL item (10 CFR Part 50, Appendix B and 10 CFR 50.65, "Maintenance Rule"). The staff finds these changes acceptable because they provide a means for ensuring a plant will have a coatings program that meets the requirements of 10 CFR Part 50 and 10 CFR 50.65. Verification of these changes in the next revision of the applicant's DCD is being tracked as **Confirmatory Item MCB-6.1.2-6**.

The applicant's original submittal included information about the transport of particulate debris, presumably coatings debris generated during a LOCA. In RAI 305-8375, Question 06.01.02-7, dated November 12, 2015, the staff requested that the applicant remove this information from the DCD because it is inconsistent with the applicant's evaluation of coating debris in its Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance," analysis. In the same question the staff requested that the applicant replace the original information in DCD Tier 2, Section 6.1.2, or provide a pointer to another DCD Section, to describe how coatings are evaluated as a potential debris source in addressing GSI-191. In its response to RAI 305-8375, Question 06.01.02-7, dated December 3, 2015 (ML15337A125), the applicant proposed deleting the original sentence about debris settlement and adding a new sentence to state that evaluation of coatings as a potential debris source in the GSI-191 evaluation is described in DCD Tier 2, Section 6.8.4.5. The staff finds this revision acceptable because it provides a consistent description of how coatings are evaluated as a potential debris source in addressing GSI-191. The staff confirmed that DCD Tier 2, Revision 1, dated March

10, 2017, was revised as committed in the response to RAI 305-8375, Question 06.01.02-7. Therefore, RAI 305-8375, Question 06.01.02-7 is resolved and closed.

6.1.2.5 Combined License Information Items

The applicant proposed a COL item, COL 6.1(2), for describing the coatings program and providing the implementation milestones. As described in Section 6.1.2.44 above, this change is being tracked as **Confirmatory Item MCB-6.1.2-6** pending incorporation of the proposed change in the next revision of the DCD.

6.1.2.6 Conclusion

Pending incorporation of the confirmatory items in the next revision to the DCD, the staff concludes that the protective coating systems and other organic materials are acceptable and meet the requirements of 10 CFR Part 50, Appendix B. This conclusion is based on the applicant conforming to the regulatory positions in RG 1.54, Revision 2, including the QA standards in ASTM D5144 and the DBA qualification testing in ASTM D3911. The applicant also provided requirements for organic materials other than coatings. Therefore, the applicant's description of protective coating systems, and of the other organic materials inside containment, provides reasonable assurance that they will not prevent the function of safety-related systems.

6.2 Containment Systems

6.2.1 Containment Functional Design

The APR1400 containment is a PWR dry containment, which consists of a prestressed cylindrical concrete shell and dome with a steel liner plate. The containment structures, systems, and components that are important to safety are designed to withstand the environmental and dynamic effects associated with both normal plant operation and postulated accidents. The containment and its associated systems establish a barrier to limit the uncontrolled release of radioactivity to the environment and to remain functional during a DBA by incorporating sufficient margin so that conditions important to safety are not exceeded throughout the postulated accident. The containment is designed as an essentially leak-tight barrier that accommodates the calculated temperature and pressure conditions resulting from the complete spectrum of postulated DBAs, up to and including a double-ended slot break in the RCS or secondary system piping, without exceeding the design leakage rate. The containment spray system (CSS), containment heat removal system (CHRS) and the associated IRWST, taken together, act to rapidly reduce the containment pressure following a LOCA or secondary system piping rupture and maintain containment pressure and temperature within acceptable limits. The design containment integrity analyses are performed at Mode 1 or Mode 2 in which the total energy from the reactor coolant, secondary system metal, and decay heat is greater than that at lower modes of operation (Mode 3 through Mode 5). However, the containment functional design is also required in lower-mode operations as postulated accidents at lower modes could still release radioactive material and increase the containment pressure and temperature.

The APR1400 containment functional design is evaluated in detail in the following five Sections of the current SE:

- Section 6.2.1.1, "Containment Structure," describes and evaluates the containment pressure and temperature calculations performed by the applicant, and discusses the effectiveness of the CHRSs. The review was conducted in accordance with NUREG–

0800, Section 6.2.1.1.A “PWR Dry Containments, Including Subatmospheric Containments.”

- Section 6.2.1.2, “Subcompartment Analysis,” reviews accident differential pressure calculated for individual containment subcompartments that might result from a high energy line break within the subcompartment. The review was conducted in accordance with NUREG–0800, Section 6.2.1.2.

- Section 6.2.1.3, “Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents,” discusses mass and energy (M&E) release calculated for LOCAs, and evaluates the assumptions and models used for the M&E release calculations. The review was conducted in accordance with NUREG–0800, Section 6.2.1.3.

- Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures inside Containment,” presents an evaluation, similar to that in Section 6.2.1.3, for M&E release calculations performed for steam and feedwater line break accidents. The review was conducted in accordance with NUREG–0800, Section 6.2.1.4.

- Section 6.2.1.5, “Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies,” addresses the calculation of minimum containment pressure used to show the adequacy of the ECCS analysis and conservatism used by the applicant in the analysis. The review was conducted in accordance with NUREG–0800, Section 6.2.1.5.

The containment heat removal functions of the CSS are discussed in detail in Section 6.2.2, “Containment Heat Removal Systems,” of this SE. Other areas related to the evaluation of the APR1400 containment functional design include Section 6.2.4, “Containment Isolation System,” Section 6.2.5, and Section 6.2.6, “Containment Leakage Testing,” of this SER.

6.2.1.1 Containment Structure

6.2.1.1.1 Introduction

The primary functions of the reactor containment building (RCB) are to protect the safety-related SSCs located within it and to establish an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment during normal plant operation and accidents. The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure shall be designed to withstand, without loss of function, the impact of the postulated accidents involving the release of high energy fluids from the RCS and secondary systems. The containment structure shall also maintain functional integrity in the long term following a postulated accident, i.e., it shall remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require. The design and sizing of containment systems are largely based on the pressure and temperature conditions that result from release of the reactor coolant in the event of a LOCA. The containment design basis includes considerations of the effects of stored energy in the RCS, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen. The containment system is not required to be a complete and independent safeguard against a LOCA by itself but functions to contain any fission products released while the ECCS cools the reactor core.

6.2.1.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this evaluation is provided in Tier 1, Section 2.2.1, "Nuclear Island Structures," Section 2.11, "Containment System," and Section 2.11.1, "Containment Structure." A summary of the technical information is as follows.

The APR1400 containment consists of a seismic Category I, safety-related, steel-lined prestressed concrete structure of a cylindrical shell and hemispherical dome built on a safety-related reinforced concrete common basemat. The containment is an essentially leak tight barrier designed to retain its structural integrity during all postulated DBAs. The containment is designed and constructed to meet the requirements of ASME Section III, Division 2. The containment internal structures consist of reinforced concrete and structural steels that support the reactor vessel and RCS. The containment internal structures enclose a reactor cavity area below the reactor vessel, which can be flooded in the event of an accident. Further description of the containment layout, including internal structures, is located in DCD Tier 1, Section 2.2.1.

DCD Tier 2: The Tier 2 information associated with this evaluation is provided in DCD Tier 2, Section 6.2.1.1. A summary of the technical information follows.

The APR1400 containment is a PWR dry containment with some differences when compared to conventional dry containments. The water source needed by the ECCS and spray systems for DBA mitigation and post-accident heat removal is from the IRWST, which is a reinforced concrete structure with a stainless steel inside liner located near the bottom of the containment. The IRWST is designed to provide adequate water level to SI, containment spray, and shutdown cooling (SC) pumps to supply continuous subcooled water to cool the containment in the event of abnormal events such as a LOCA or secondary system piping rupture. This configuration has the advantage of not being required to switch the ECCS water source from an external water tank to the containment sump when the water in the tank is depleted and water recirculation starts. In case of an accident, water spilled from the RCS, ECCS, or from the sprays is condensed on the containment internal surfaces and eventually drains back to the IRWST. The APR1400 containment has safety-related spray systems and nonsafety-related fan coolers, whose function is to remove heat from the containment and, in the case of sprays, also scrub fission products released into the containment atmosphere following an accident. The CSS and CHRS act to rapidly reduce the containment pressure following a LOCA or secondary system piping rupture and maintain containment pressure within acceptable levels. Each train of the CSS rejects thermal energy in IRWST water to the component cooling water system (CCWS) through the containment spray heat exchanger. As a result, the containment accommodates, without exceeding the design leakage rate and with sufficient margin to the established pressure limit, the calculated pressure and temperature conditions resulting from a DBA.

The APR1400 design does not utilize a secondary containment. The containment's inside dimensions are 45.72 m (150 ft) in diameter and 76.66 m (251.5 ft) in height. The lateral walls of the cylindrical structure and dome are 1.37 m (4 ft. 6 in) and 1.22 m (4 ft.) thick, respectively. The inner surface of the containment wall and dome is covered with a 6.0 mm (0.25 in) thick leak-tight welded steel liner plate. The primary containment completely encloses the steam supply system, which includes the reactor vessel, steam generators (SGs), RCPs, pressurizer, and the associated piping.

The containment is designed to accommodate conditions resulting from a spectrum of postulated DBAs outlined in DCD Tier 2, Table 6.2.1-1, "Spectrum of Postulated Accidents." These conditions include elevated temperature, pressure, and humidity. DCD Tier 2, Tables

6.2.1-2, “Calculated Values for Containment and Subcompartment Pressure Parameters,” and 6.2.1-3, “Principal Containment Design Parameters,” provide the calculated and design values of the peak containment parameters, respectively. DCD Tier 2, Tables 6.2.1-4, “Double-Ended Suction Leg Slot Break – Maximum SIS Flow (0.9121m^2 (9.8175ft^2) Total Break Area,” through 6.2.1-18, “Main Steam Line break, 0% Power – MSIV Failure (0.381 m^2 (4.1 ft^2) Total Break Area),” provide the M&E release data for the postulated LOCAs and MSLBs listed in DCD Tier 2, Table 6.2.1-1. The applicant performed the containment analysis subsequent to the M&E analysis using the computer code GOTHIC (Reference 1), which calculates containment pressure and temperature during the course of the transient. Appendices A through G to the Technical Report (TR) APR1400-Z-A-NR-14007-P, Revision 0, “LOCA Mass and Energy Release Methodology,” (Reference 2) describe the methodology and the GOTHIC containment model used to predict the containment pressure and temperature response to a spectrum of high-energy line break DBAs in the large, dry APR1400 containment building. The sources of generated and stored energy in the RCS and secondary coolant system considered in the LOCA analyses include: primary coolant, secondary coolant, primary walls (including reactor internals), secondary walls, SI water, core power, and decay heat.

The containment design pressure of 4.218 kg/cm^{2}G (60 psig or 515 kPa) is based on the worst-case LOCA and bounds all of the postulated secondary system piping rupture events for peak containment pressure. The DCD Tier 2, Section 6.2.1.1.1 reports that a design margin of 10 percent has been taken into account in the determination of this value. The containment design temperature is 143.3 °C (290.0 °F). Per Reference 2, the applicant calculated a DBA LOCA peak saturation temperature of 134.59 °C (274.25 °F). The APR1400 containment is also designed for a limiting containment pressure reduction event to withstand an external pressure loading of 0.28 kg/cm^{2}G (4.0 psig or 27.5 kPa) relative to ambient pressure.

ITAAC: The ITAAC associated with this evaluation of APR1400 DCD Tier 2, Section 6.2.1.1 are provided in DCD Tier 1, Table 2.2.1-2, and Table 2.11.1-2, “Containment Structure ITAAC.”

TS: The TS associated with this evaluation of DCD Tier 2, Section 6.2.1.1 are provided in DCD Tier 2, Chapter 16, “Technical Specifications,” Section 3.6, “Containment Systems.” The TS that apply specifically to this review area include Sections Sections 3.6.1, “Containment,” 3.6.4, “Containment Pressure,” and 3.6.5, “Containment Air Temperature.”

6.2.1.1.3 Regulatory Basis

The Commission regulations and the associated acceptance criteria for this area of review are given in NUREG–0800 Section 6.2.1.1.A. Review interfaces with other SRP Sections can also be found in NUREG–0800, Section 6.2.1.1.A. The acceptance criteria are based on meeting the relevant regulatory requirements as summarized below.

- GDC 13, “Instrumentation and control,” requires instrumentation to be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to assure adequate safety. This means that instrumentation shall be capable of operating in the post-accident environment in order to monitor the containment atmosphere pressure and temperature, and the sump water level and temperature following an accident. It shall have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. Meeting this requirement helps ensure that the containment precludes the release of radioactivity to the environment.

- GDC 16, “Containment design,” as it relates to the reactor containment and associated systems being designed to ensure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Since the primary reactor containment is the final barrier of the defense-in-depth concept to protect against the uncontrolled release of radioactivity to the environment, preserving containment integrity under the dynamic conditions imposed by postulated LOCAs is essential. This means that the primary reactor containment shall be designed as an essentially leak tight barrier that will withstand the most extreme accident conditions for the duration of any postulated accident.

- GDC 38, “Containment heat removal,” as it relates to the CHRS(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels. The CHRS supports the containment function by minimizing the duration and intensity of the pressure and temperature increase following a LOCA, lessening the challenge to containment integrity. Meeting GDC 38 helps ensure that the containment can fulfill its role as the final barrier against the release of radioactivity to the environment.

- GDC 50, “Containment design basis,” as it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any LOCA without exceeding the design leakage rate and with sufficient margin.

- GDC 64, “Monitoring radioactivity releases,” requires that the reactor containment atmosphere be monitored for the release of radioactivity from normal operations, anticipated operational occurrences (AOOs), and postulated accidents. Meeting this requirement allows operators to ensure that containment meets its safety function of preventing a release of radioactivity to the environment.

- The regulations in 10 CFR 52.47(b)(1) requires that the DC application contain proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and acceptance criteria met, a plant that incorporates the certified design is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC’s regulations.

6.2.1.1.4 Technical Evaluation

GDC 16 and GDC 50 of Appendix A to 10 CFR Part 50, “ECCS Evaluation Models” require, in part, that the reactor containment structure and associated heat removal system shall be designed with sufficient margin to accommodate the calculated pressure and temperature conditions resulting from any LOCA. NUREG–0800, SRP Section 6.2.1.1A, “PWR Dry Containments, including Subatmospheric Containments,” specifies that the containment design pressure should provide at least a 10 percent margin above the accepted peak calculated containment pressure following a LOCA, a steamline or feedwater line break, to satisfy the GDC 16 and GDC 50 requirements for sufficient design margin.

The calculations referenced in DCD Tier 2, Section 6.2.1.1, report a greater than 10 percent margin to the containment design pressure 4.218 kg/cm^{2}G (60 psig or 515 kPa) for all five LOCA and ten MSLB cases of the break spectrum analysis. However, preliminary confirmatory calculations performed by the staff for the limiting double-ended discharge leg slot break

(DEDLSB) LOCA case using a multi-node MELCOR model yielded a peak pressure higher than the value (51.09 psig) calculated by the applicant's single-node GOTHIC model for the containment atmosphere region, such that the resulting margin to the design pressure is significantly less than 10 percent. The staff's multi-node confirmatory calculations using MELCO computer code also resulted in a much higher (193.8°C (380.9°F)) peak containment temperature for the limiting MSLB case, as compared to 167.4°C (333.4°F) from the applicant's single-node model, as documented in DCD Tier 2, Table 6.2.1-2. DCD Tier 2, Section 6.2.1.1.3.3, "Capability for Energy Removal from the Containment," acknowledges that the MSLB containment temperature exceeds the containment design temperature (290.0 °F) for a period prior to containment spray (CS) actuation. Figures 6.2.1-6, "Containment Pressure and Temperature vs. Time; MSLB – 102% Power with Loss of a CSS Train," through 6.2.1-15, "Containment Pressure and Temperature vs. Time; MSLB – 0% Power with an MSIV Single Failure," show that the containment temperature exceeds the containment design temperature for a couple of minutes for all MSLB cases analyzed. A related concern is that a single-node variant of the staff's multi-node MELCOR model is likely to predict even higher containment peak pressure and temperature than the values predicted by the staff's multi-node MELCOR model, which as reported earlier in the paragraph, are already much higher than the applicant's single-node GOTHIC model results reported in the DCD Section 6.2.1.1.3. On January 28, 2016, the staff issued RAI 378-8442, Question 06.02.01.01.A-9, to address these issues. In order to resolve the differences between the applicant's calculations as reported in the DCD and the staff's confirmatory calculations, the staff also requested the applicant to explain the selection and conservatism of the model inputs, and provide an electronic copy of the GOTHIC input decks for the APR1400 containment peak pressure and temperature calculations, along with the applicable reports. On November 11, 2015, the staff issued RAI 378-8342, Question 06.02.01.01.A-3, to this effect.

The applicant's response to RAI 378-8442, Question 06.02.01.01.A-9 (ML16064A417), dated March 4, 2016, provided a summary of the plant-specific design parameters and modeling assumptions that impact the containment peak pressure and temperature, such as containment/IRWST volumes, passive and active heat sinks, and initial and boundary conditions. The applicant's response to RAI 378-8442, Question 06.02.01.01.A-9 was complimented by the applicant's response to RAI 296-8342, Question 06.02.01.01.A-3 (ML15350A048), dated December 3, 2015, that submitted the APR1400 GOTHIC input decks for the containment peak pressure and temperature analyses. The staff's review of the GOTHIC models showed that the containment building is modeled as three lumped-parameter volumes that represent the containment atmosphere region, IRWST, and water trap spaces in containment. The water trap includes volumes such as the reactor cavity and holdup volume (HVT) that may be filled with water due to containment internal flooding during an accident. The outer surfaces of the containment shell (wall and dome) and inner surface of the containment floor are conservatively modeled adiabatically to prevent heat release to the atmosphere through the containment structure. The CSS is modeled using the pump, heat exchanger, and spray nozzle components of the GOTHIC code to transfer water from the IRWST to the containment atmosphere as liquid droplets. GOTHIC boundary conditions are used to model M&E release to the containment from the RCS by specifying the time history of break flow and enthalpy data. Table 4-2, "Double-Ended Discharge Leg Slot Break – Maximum SIS Flow," of the TR (Reference 2) presents M&E releases on each phase (blowdown, reflood, post-reflood and decay heat period) of the limiting LOCA (Double-Ended Discharge Leg Slot Break with maximum SIS flow). The NRC staff's review of the applicant's GOTHIC models for the LOCA and MSLB analyses enabled the staff to inspect the input values and modeling assumptions summarized in the response to RAI 378-8442, Question 06.02.01.01.A-9, dated March 4, 2016, and run sensitivity cases, which helped identify two non-conservatisms in the GOTHIC model used by the applicant with respect to the containment peak pressure and temperature, as described in the following paragraphs.

GOTHIC offers two types of heat transfer coefficients (HTCs); convection and condensation of steam. The HTC for the convection heat transfer can be chosen as Direct or Tagami. Likewise, the HTC for steam condensation can be specified as DLM (Diffusion Layer Model) or Uchida option. Tagami and Uchida are empirical heat transfer correlations that are based on experimental condensation data, whereas the Direct/DLM calculates the condensation rate and sensible heat transfer rate directly on the structure's surface using heat/mass transfer analogy. The Tagami correlation is most suited for LOCA blowdowns. NUREG-0588 suggests the Tagami and Uchida condensation heat transfer correlations should be used for LOCA blowdowns & MSLBs, respectively. Determination of which combination would result in the highest containment peak pressure depends on the specific plant. The staff determined that the applicant had used the Direct-DLM HTC model in the APR1400 licensing GOTHIC models and had not analyzed other combinations. The staff also determined that the submitted GOTHIC models used an inertial length of 1 ft, which could be made more conservative by using a value up to the full containment height of 166 ft. As GOTHIC uses inertial length in the momentum equation, its selection impacts the containment peak pressure and temperature calculations. The staff determined that the applicant needed to update the analyses to a justifiable inertia length.

In the July 7, 2016, public teleconference, the staff asked the applicant to perform sensitivity analyses to justify their HTC model and inertial lengths. The applicant performed the sensitivity analyses and submitted the results in a revised response to RAI 296-8342, Question 06.02.01.01.A-3 (ML16210A347), dated July 28, 2016, along with the modified GOTHIC models for the sensitivity analyses. The results confirmed that the Tagami-Uchida HTC model and 166 ft inertial length to be more conservative with respect to the containment peak pressure and temperature. The staff asked the applicant to update the DCD results (graphs, tables, etc.) for the revised licensing basis calculations for the combined Tagami-Uchida HTC model and 166 ft inertial length. The updated DCD results will allow the staff to make the safety findings regarding the calculated containment peak pressure, peak temperature, and available containment pressure margins during first 24 hours after the DBA initiation. The applicant addressed the non-conservatisms in the GOTHIC model in a revised supplemental response to RAI 296-8342, Question 06.02.01.01.A-3, dated March 16, 2017 (ML17081A282). The response provided the revised limiting GOTHIC analyses, and updated the DCD and TR to reflect the revised licensing basis calculations. The applicant also submitted the revised GOTHIC input decks on a CD media, and the description of the APR1400 containment peak pressure and temperature calculations on March 21, 2017 (ML17082A188). KHNP performed sensitivity analyses to estimate the effects of using the two GOTHIC condensation HTC options (Tagami/Uchida vs. Direct/DLM) and junctions' inertial length on the containment peak pressure and temperature. The containment maximum pressure and temperature from the DBA LOCA case (Double ended discharge leg slot break with maximum SI), which use the Tagami/Uchida option, are higher than that of the Direct/DLM option by 0.3 psi and 0.4 °F, respectively. Even though the Tagami option produces higher containment peak pressure and temperature, they are well bounded by the containment design pressure (74.7 psia) and design temperature (290 °F).

KHNP performed a sensitivity analysis for the inertial length of each junction to estimate the impact of the inertial length on the containment peak pressure and temperature using various inertial lengths. In the APR1400 containment GOTHIC model, a total of 17 flow paths are included and a nominal value of 1.0 ft is used for the inertial length of each flow path since most of the flow paths connecting to the containment volume are used as the flow boundary input for the M/E release. Using an inertial length of 166 ft, which is the longest height among the volumes used in the containment model for all flow paths, increases the containment maximum pressure and temperature by 0.34 psi and 0.48 °F to 66.14 psia and 274.64 °F, respectively.

Based on the applicant's sensitivity analyses, the case that uses the GOTHIC Tagami/Uchida condensing options with maximum inertial length (166 ft) for each flow path results in the highest containment peak pressure, which is greater than the case that uses GOTHIC Direct/DLM with minimum inertial length (1.0 ft) by approximately 0.6 psi. The applicant revised the GOTHIC containment model for more conservative Tagami and Uchida condensation heat transfer correlations for LOCA and MSLB, and a higher junction's inertial length of 166 ft. The staff concludes that the peak calculated containment pressure remains below the containment design pressure of 74.7 psia with a sufficient pressure margin of greater than 10 percent. The peak containment temperature is also less than the containment design temperature. The staff's MELCOR confirmatory calculations also support the safety finding. The applicant submitted the mark-ups to appropriately revise the DCD Tier 2, Section 6.2.1 and the TR (Reference 2) as a part of the RAI response. The staff determined that the proposed revisions are acceptable because they provide the details and clarity needed to reach the safety findings, and update the DCD and TR to reflect the GOTHIC model description, changes, and results.

In addition to the modification of GOTHIC model for the wall condensation and junctions' inertial length, the staff also reviewed the following additional GOTHIC containment model revisions made by the applicant and presented in the supplemental RAI response to RAI 296-8342, Question 06.02.01.01.A-3, dated March 16, 2017 (ML17081A282). Some additional details of the GOTHIC model and its modifications were provided in the supplemental RAI response to RAI 378-8442, Question 06.02.01.01.A-9, dated July 5, 2017 (ML17186A379).

Break Flow Model

In the previous GOTHIC model, the break liquid after phase split was assumed to release as droplets during the entire LOCA blowdown. In the revised GOTHIC model, the break liquid discharge as drops is limited to the condition that the droplet remains superheated liquid relative to the containment atmosphere, even in blowdown phase. The staff accepts that this model revision does not impact the maximum containment pressure and temperature as the containment atmosphere is expected to remain saturated during the entire LOCA blowdown.

Water Trap Model

The water discharged to the containment during an accident, which includes break spillage and spraying flow, is not fully returned to the IRWST due to trap spaces in containment that may block the water going to the IRWST. The previous GOTHIC containment model did not take credit for the water hold-up volume capable of being filled with the water released into the containment during the transients and not returning to the IRWST. Instead, the same size of water volume corresponding to the sum of all trap spaces was conservatively excluded from the initial IRWST water volume. In the revised GOTHIC containment model, water trapped in the containment is taken into account to estimate the containment peak pressure and temperature based on reasonable prediction of the IRWST water temperature during the transient. The use of the water trap model decreases temperature of the water supplying to the SI pumps and CS pumps for core and containment cooling in comparison to the previous model that has a reduced IRWST water volume without the water trap, consequently decreasing the containment peak pressure and temperature. The applicant estimated the effect of the water trap model on the peak containment pressure to be about 0.5 psi of pressure decrease, which is not significant compared to the containment peak pressure margin, and does not affect the staff's safety finding about the containment design.

Energy Release Model (Decay heat phase)

During the decay heat phase (long-term cooling period), the metal and coolant stored energy in the SG(s) secondary side is released to the RCS primary side following decrease of the RCS coolant temperature. In the previous GOTHIC model, the energy stored in metal and coolant in the SGs secondary side was assumed to be released at a constant rate during a 24-hour period of a LOCA. The SGs secondary side's energy release with a constant rate is less conservative than that with a decay decrease rate in view of the maximum IRWST water temperature, since the IRSWT water reaches the maximum temperature within approximately 3 hours of the accident. Therefore in the revised GOTHIC model, the residual stored energy in SGs secondary side is assumed to be released in a decreasing rate aligned with decay heat and conservatively completed within 24 hours. This model revision does not impact on the maximum containment pressure and temperature; however, it increases the containment pressure at 24 hours, as well as the maximum IRWST water temperature. The staff finds the change to be conservative and does not affect the safety finding about the containment design.

Decay energy curve

The GOTHIC containment model additionally takes into account the decay heat contribution from actinides other than U-239 and Np-239 based on the decay energy associated with the

ANSI/ANS 5.1-1979 with 2σ uncertainties for full reactor power. An error in calculating additional decay energy from actinides other than U-239 and Np-239 was found in the previous GOTHIC model and was addressed in the revised GOTHIC model. The staff agrees that the model revision does not affect the peak containment pressure and temperature since the GOTHIC decay energy model is used only for the mass and energy release calculation during the decay heat phase.

Per the submitted mark-ups of the supplemental RAI response to RAI 296-8342, Question 06.02.01.01.A-3, the revised peak containment pressure is 3.60 kg/cm^{2}G (51.21 psig or 454.41 kPa) under the DBA LOCA. The staff determined that the APR1400 containment design pressure meets the GDC 15 and 50 requirements for sufficient design margin by having at least a 10 percent margin above the maximum peak calculated containment pressure. The revised DBA LOCA peak saturation temperature is 134.95 °C (274.95 °F), which is less than the containment design temperature of 143.3 °C (290.0 °F). The maximum peak containment atmosphere temperature of 171.55 °C (340.78 °F) occurs in the limiting DBA MSLB. The DBA MSLB containment temperature does exceed the containment design temperature for less than 2 minutes, but the superheated vapor condenses rapidly after coming into contact with the subcooled surface of structures within the containment. In addition, the calculated maximum MSLB containment surface temperature of 133.8 °C (272.9 °F) does not exceed the containment design temperature of 143.3 °C (290.0 °F). Therefore the staff accepts that the superheated condition has an insignificant impact on containment integrity. The RAI supplemental response to RAI 296-8342, Question 06.02.01.01.A-3 also submitted the revised LOCA mass and energy release Tables 6.2.1-4 through 6.2.1-8 used in the updated GOTHIC model. The response also included the updated DCD Chapter 6.2.1 figures with the revised GOTHIC model containment pressure and temperature results for various DBA LOCA and MSLB analyses, as well as the revised TR mark-ups that provide the details of the above-described GOTHIC model revisions. The staff finds the applicant's response acceptable, and RAI 296-8342, Question 06.02.01.01.A-3 and RAI 378-8442, Question 06.02.01.01.A-9, are resolved, and the verification of the proposed DCD and TR revisions is being tracked under **Confirmatory Item 06.02.01.01.A-3.** Apart from the discussion above, the staff identified a non-conservatism in that the potential impact of the burnup dependent thermal conductivity degradation (TCD) was not accounted for in the original M&E release used in the APR1400 containment peak pressure and temperature calculations. The lack of a burnup dependent TCD model in the computer codes resulted in over-prediction of the fuel thermal conductivity at higher burnups. The staff expected that if the TCD is duly accounted for, then the fuel thermal conductivity would decrease, which would result in a higher initial core stored energy and rod temperatures, and would lead to more M&E release into the containment and, thus, higher peak containment pressure and temperature. The applicant's accounting for the TCD resulted in increasing the fuel enthalpy and the fuel centerline temperature which impacted several DCD sections (Chapters 4, 6, 9, 15), and topical and technical reports (References 4-6). For the containment functional design, the staff needed to review the potential impact of the acceptable TCD model on the limiting peak containment pressure and temperature analyses results and the corresponding M&E release tables. The applicant re-performed the M&E and containment analyses at the limiting burnup using the modified thermal conductivity for all rods and fuel rod gap properties to identify the changes from the base cases. The TCD issue was tracked under RAI 5-7954, Question 26895 (TR PLUS7 Fuel Design for the APR1400-11). The applicant's supplemental response to RAI 5-7954, Question 26895, dated August 11, 2017 (ML17223B385) stated that although the core stored energy may be increased by the TCD, the containment peak pressure with the TCD effect is slightly higher by 0.63 psi for LOCA and 0.33 psi for MSLB, than the respective cases without accounting for the TCD. The response also submitted the revised mass and energy release Tables 6.2.1-4 through 6.2.1-18, Table 6.2.1-37, and Figures 6.2.1-33 through 6.2.1-36, and provided a DCD Tier 2 Section 6.2.1.3.2 mark-up. The supplemental response to RAI 296-8342, Question 06.02.01.01.A-3, shows that these tables are also updated in the TR APR1400-Z-A-NR-14007-P, Revision 1, "LOCA Mass and Energy Release Methodology," with the TCD effect. The staff determined that the proposed revisions are acceptable because they provide the details and clarity needed to update the M&E release analyses for the TCD. The staff determined that the TCD effect on the limiting LOCA and MSLB cases is not significant compared to the safety margin in the containment design that the applicant demonstrated in the DCD. The detailed methodology descriptions and M&E and P/T results of the TCD effect analysis are provided in Section 3 of the revised M&E Release TR (Reference 3). The staff finds the response to RAI 5-7954, Question 26895 acceptable for the applicant's assessment of the TCD impact on the containment functional design. The verification of the submitted DCD and TR revisions is being tracked under **Confirmatory Item 06.02.01.01-7954**. In addition, the supplemental RAI response to RAI 296-8342, Question 06.02.01.01.A-3 and the associated DCD and TR revisions, M&E tables, and GOTHIC model update incorporate the TCD effect on the containment temperature and pressure.

GDC 38 of Appendix A to 10 CFR Part 50 requires, in part, that the CHRS shall rapidly reduce the containment pressure following any LOCA, lessening the challenge to the containment integrity. NUREG–0800, SRP Section 6.2.1.1A specifies that the containment pressure should be reduced to less than 50 percent of the peak calculated pressure for the design basis LOCA within 24 hours after the postulated accident, in order to satisfy the GDC 38 requirements for the containment as the final barrier against the release of radioactivity to the environment. DCD Tier 2, Table 1.9-2, "APR1400 Conformance with the Standard Review Plan," states that "the APR1400 conforms to SRP Section 6.2.1.1.A Acceptance Criterion No. 2. DCD Tier 2, Section 6.2.1.1.3.2, "Containment Response Analyses to Loss-of-Coolant Accidents," states that "[t]he calculated containment pressure at 24 hours, 1.795 kg/cm^{2}G (25.54 psig), is 42.35 percent of the peak calculated pressure for the limiting LOCA and thus meets the requirements of GDC 38." Using the values provided in the DCD, the staff calculated the pressure of 40.24 psia at 24 hours to be equal to 61.2 percent of the peak calculated pressure of 65.79 psia occurring at 324 seconds into the event. The staff issued RAI 8442, Question 06.02.01.01.A-10, to ask for justification about how the calculated containment pressure at 24 hours, 40.24 psia (25.54 psig), is considered reduced to less than 50 percent of the peak calculated pressure, 65.79 psia (51.09 psig). The peak vapor temperature was determined to be just below 135°C (275°F).

The applicant's response (ML16064A417), dated March 4, 2016, stated that the value of containment pressure at 24 hours, 1.795 kg/cm^{2}G (25.54 psig), documented in DCD Tier 2, Section 6.2.1.1.3.2 is incorrect, which will be revised to 1.521 kg/cm^{2}G (21.64 psig), as shown on the associated mark-up. The revised pressure at 24 hours, 1.521 kg/cm^{2}G (21.64 psig), is estimated to be 42.35 percent of the peak calculated pressure, 3.59 kg/cm^{2}G (51.09 psig), which would meet the GDC 38 requirement of "less than 50%." The applicant also clarified that using values in the British pressure gauge unit, psig, to determine the pressure margin at 24 hours after the accident, is in accordance with the basic unit requirement described in the RG 1.206 Table 6-1. The staff finds the applicant's use of gauge pressures in the determination of margins acceptable. However, RAI 378-8442, Question 06.02.01.01.A-10 was not resolved as the staff needed to verify the calculated containment peak pressure and peak-pressure margin at 24 hours, and associated DCD graphs and tables in the approved licensing calculations that were originally tracked under RAI 296-8342, Question 06.02.01.01.A-3. The supplemental response to RAI 378-8442, Question 06.02.01.01.A-10, dated June 27, 2017 (ML17178A283), shows that the recalculated containment pressure at 24 hours after the postulated accident was reduced to 1.336 kg/cm^{2}G (19.0 psig), which was 37.1 percent of the peak calculated pressure (51.21 psig) for the design basis LOCA (DEDLSB with Max. SI flow). The staff determined that this information is consistent with the revised results presented in the RAI 296-8342, Question 06.02.01.01.A-3 supplemental response obtained from the updated licensing basis GOTHIC model for containment pressure and temperature calculations. The staff finds the applicant's response acceptable because it demonstrated that the APR1400 containment design satisfies the GDC 38 requirements and SRP Section 6.2.1.1A guidance for containment heat removal. Therefore, RAI 378-8442, Question 06.02.01.01.A-10 is resolved, and the verification of the related DCD and TR (Reference 2) revisions submitted as mark-ups as part of the RAI 296-8342, Question 06.02.01.01.A-3 supplemental response is being tracked under **Confirmatory Item 06.02.01.01.A-3.**

On February 27, 2016, the staff issued RAI 378-8442, Questions 06.02.01.01.A-9 and 06.02.01.01.A-10, to gain safety insights into the applicant's limiting peak pressure/temperature analyses for the containment by resolving the differences between results of the applicant's calculations as reported in the DCD and the staff's confirmatory calculations showing insufficient design margins. The staff was aware of the conservatisms used in calculating the heat sink parameters, containment volume, nodalization, spray characteristics, and assumptions listed in the "Assumptions and Input Parameters" section of DCD Tier 2, Section 6.2.1.1.3.2. However, in order for the staff to understand the discrepancies in the calculated pressure and temperature and reconcile the differences between the two models, the applicant needed to provide a full accounting of the input data, sensitivity coefficients (inertial lengths, loss coefficients, etc.), assumptions used in heat transfer correlations used for containment analysis or M&E release calculations, and any other assumptions or uncertainties not listed in Section 6.2.1.1 of the DCD that could adversely impact the design margins in the containment peak pressure or temperature. The applicant was also asked to address any potential impact of excessive heating on the containment integrity and equipment qualifications for other safety-related systems.

Summarized below are the conservatisms built into the APR1400 containment response analysis.

- Treating the containment shell as insulated, which is conservative in the decay heat phase during long term cooling.

- Accounting for the thermal resistance of the air gap between the steel liner and the concrete of the containment shell, which reduces heat loss to the heat sinks and increases peak containment pressure.

- Not crediting any containment fan coolers.

- Assuming only one train of CSS.

- Using a one-dimensional heat conduction equation, which is more conservative than accounting for multi-dimensional heat conduction.

- Using a single lumped-volume for containment, which is more conservative than subdivided volume as shown in containment analysis of Carolina Virginia Tube Reactor (References 2 and 3).

- Accounting for pump heat addition to the working fluid such as CS.

- Ignoring forced convection and radiation heat transfer mechanisms from gases in the containment atmosphere to the containment structures (heat conductors) in both LOCA and MSLB response analyses.

- Ignoring heat loss from the containment atmosphere gases to the IRWST pool surface by setting the interfacial area equal to zero to isolate the cooler IRWST water from the containment atmosphere in both the LOCA and MSLB response analyses.

Heat sinks are an important parameter in calculating the containment pressure and temperature transients. Assumptions used in modeling passive heat structures are described in Section A.2.3.3, "Thermal Conductors," of the TR (Reference 2). APR1400 DCD Tier 2, Table 6.2.1-23, "Passive Heat Sink Data," contains the passive heat sink data used in the containment peak pressure and temperature analysis. A total of 18 passive heat structures are considered in the containment atmosphere volume. Table 6.2.1-23 provides the material types, thicknesses, surface areas, and surface conditions of each of the structures in detail. However, the table contains heat sink thickness values tabulated with inconsistent units and inconsistent unit conversions. These values also appear to be inconsistent with the values provided to the staff in the RAI 49-7825, Question 06.02.01.05-1 response (ML15267A720), dated September 24, 2015, which updates the input parameters used to perform the minimum containment pressure analysis. The staff determined that the applicant needs to update the DCD with appropriate values for the containment heat sinks and other input parameters applicable to the containment peak pressure and temperature analyses, and the applicant was requested to report the updates and their impact on the analysis. Therefore, on November 5, 2015, the staff issued RAI 296-8342, Question 06.02.01.01.A-2, to address this issue to help inform the staff's confirmatory analyses and ensure consistency with other RAI responses, with the overall objective of meeting the GDC 16 and GDC 50 requirements for sufficient design margin. The applicant's response to RAI 296-8342, Question 06.02.01.01.A-2 (ML15350A048), dated December 3, 2015, provided a description of the GOTHIC input decks for the LOCA and MSLB containment analyses. However, the applicant's response did not confirm the validity of the heat sink values in Table 6.2.1-23, which the staff informed the applicant of in the July 7, 2016, public teleconference. The staff review of the GOTHIC licensing decks also revealed the following two major issues that were communicated to the applicant during the May 17, 2017, public teleconference.

- Three additional heat sinks were specified in the licensing basis GOTHIC deck that were not included in the DCD, Tier 2, Table 6.2.1-23 which specifies the passive heat sink data (material types, thicknesses, surface areas, boundary conditions) for a total of 16 passive heat structures, which would be less conservative.

- In DCD Table 6.2.1-23, the tabulated heat sink thickness is half of the physical thickness of the actual heat sink and the tabulated surface area is twice the surface area of one side of the two-sided heat sink, which suggested that the table was populated to make it consistent with the GOTHIC decks rather than describing the actual physical dimensions of the heat sinks, as had been typically done in other DCAs. During the subsequent staff confirmatory calculations, an uncertainty was identified around the reported safety margin due to the ambiguity over whether the surface area corresponded to one or both sides of the heat sink.

The applicant's supplemental response to RAI 296-8342, Question 06.02.01.01.A-2, dated June 19, 2017 (ML17170A217) verified that all the heat sinks in DCD Tier 2 Table 6.2.1-23 (Revision 1) have correct values and are consistent with the values provided in the response to RAI 49-7825, Question 06.02.01.05-1. The staff agrees that the analyses results were not affected, since all GOTHIC decks used for the analyses were prepared using British units, and the results were converted into metric units for consistency within the DCD. The staff verified that the applicant appropriately revised DCD Tier 2 Table 6.2.1-23 (Revision 1).

With regard to the definition of the thickness and surface area of passive heat sinks tabulated in the DCD Table 6.2.1-23, the response explained how various passive heat sinks were classified depending upon each surface's boundary condition and how the equivalent thicknesses of various passive heat sinks of arbitrary and complex shapes inside the APR1400 containment were calculated using the net volume occupied by the structure and the surface area exposed to the containment atmosphere. The surface area was the minimum value determined from consideration of the geometry uncertainty of each structure. The staff determined that the applicant's formulation of the equivalent heat sink thickness to be conservative because it would lead to a reduced thickness for the heat sink. The applicant also submitted the mark-ups to appropriately revise the DCD Tier 2, Table 6.2.1-23, and Table A-3A of the TR (Reference 2) as a part of the RAI response. The staff determined that the proposed revisions are acceptable because they provide the details and notes needed to interpret the tabulated heat sink data and boundary conditions. Now the table notes appropriately describe whether the tabulated heat sink surface sink is partially or completely exposed to the containment atmosphere and, thus, whether the surface area can be interpreted as one-sided or two-sided. This lack of clarity on the surface area had initially led the staff to undervalue heat sinks in the MELCOR confirmatory calculations that then underestimated the containment peak pressure margins. When corrected, the staff confirmatory analyses predict containment pressures and temperatures similar to the licensing basis results, such that the staff had no concerns about the safety margins. The staff finds the applicant's response acceptable, and RAI 296-8342, Question 06.02.01.01.A-2 is resolved, and the verification of the proposed DCD and TR revisions is being tracked under **Confirmatory Item 06.02.01.01.A-2.**

The applicant's June 19, 2017 supplemental response also stated that the three additional passive heat sinks specified in the GOTHIC that were not listed in the DCD Table 6.2.1-23 represent the basemat, IRWST slab & HVT in containment, and RCS metal. The response explained that the basemat and IRWST slab & HVT conductors were conservatively excluded from the GOTHIC analyses as the heat absorption on their exposed surfaces may be disturbed while these structures are flooded and filled with spilled water during an accident. Both sides of the inner structure surfaces are set to an adiabatic condition to prevent heat transfer from/to the structure. The response further explained that the RCS metal energy conductor (19th conductor in GOTHIC deck) is included as one of the GOTHIC thermal conductors. However, it doesn't have the role of a passive heat sink to absorb heat from the atmosphere during an accident, as this conductor represents the release of the metal energy stored in the RCS and SGs during long-term decay heat phase after the End-Of the Post Re-flood (EOPR). Therefore, it doesn't impact the calculation results until reaching the EOPR. Based on its own review of the GOTHIC deck, the staff concludes that the three heat sinks did not affect the GOTHIC calculations, and finds it acceptable that they were excluded from DCD Tier 2, Table 6.2.1-23.

The nominal reactor core thermal power for the APR1400 design is 3,983 MWt (3.775×10^6 Btu/sec). RCS parameters for the nominal core power of 3,983 MWt are given in DCD Tier 2, Table 6.2.1-20, "Initial Conditions for Containment Peak Pressure and Temperature Analysis." Regarding the initial reactor power (prior to the break initiation), an initial full reactor power level with 2 percent uncertainty plus the maximum RCP power, equates to 4091.86 MWt (3.878×10^6 Btu/sec). Even though, DCD, Tier 2, Table 6.2.1-24, "Initial Conditions for Containment Pressure Analysis," lists some initial conditions for containment pressure analysis, the application does not provide any details on how the initial conditions relevant to containment response analysis were established. Besides the GDC 16 and GDC 50 requirements, as described in NUREG–0800, SRP Section 6.2.1.1.A, ANSI/ANS 56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments," also has established detailed guidelines for containment response to DBAs, which explicitly require that the initial conditions shall be chosen to yield a conservatively high peak containment atmosphere pressure and temperature. In selecting the initial dry primary containment atmospheric conditions and structural temperatures, consideration should be given to the competing effects of the initial air mass and the active and passive heat sink thermal capacities. Initial pressure and degree of subcooling are also important as they affect the critical flow rate at the break. It is important to know the values of these parameters used in the LOCA M&E analysis to assess the design margins reported in the DCD to meet the GDC 16 and GDC 50 requirements for sufficient design margin to the design pressure and temperature for the reactor containment structure and associated heat removal system under the limiting DBA conditions. Therefore, on December 3, 2015, the staff issued RAI 327-8354 (ML15337A169) with five questions, to seek additional information about the initial and boundary conditions the applicant used for the LOCA analyses for the containment, as discussed below.

The initial containment atmosphere temperature is an important parameter in the calculation of the peak containment pressure and temperature as such, its upper value constitutes a limiting condition for operation (LCO) in the TSs. The initial atmosphere temperature not only affects the containment response to a design basis LOCA but also other aspects of the accident such as the SI water temperature from safety injection tanks (SITs). As discussed in Appendix C, (case studies for modeling characteristics) to the TR (Reference 2) and given in Table C-1A, "Containment Analysis Bounding the Initial Conditions," the maximum containment temperature is 48.89 ^{o}C (120 ^{o}F). If the value listed in Table C-1A includes an instrument uncertainty, then the initial atmosphere temperature of 46.1 ^{o}C (115 ^{o}F) would not be conservative; and if it does not include an instrument uncertainty, then an initial atmosphere temperature of 51.67 ^{o}C (125

°F) shall have been used for all the analyses. On December 3, 2015, the staff issued RAI 327-8354, Question 06.02.01.01.A-4, to ask the applicant to explain whether the containment initial atmosphere temperature is based on the typical value commonly used in the containment response analysis or obtained from an energy balance analysis for the RCS heat loss versus fan coolers heat removal. In its response to RAI 327-8354, Question 06.02.01.01.A-4 (ML15364A583), dated December 30, 2015, the applicant's response clarified that the value of 49 °C (120 °F) in Table C-1A is chosen from the temperature LCO of the TS Section 3.6.5. The response also confirmed that no uncertainty in the initial atmosphere temperature was assumed. The staff requested clarification about why instrument uncertainty was not applied to the temperature LCO of 120 °F in the technical specifications in the July 7, 2016 public teleconference. The applicant's supplemental response to RAI 327-8354, Question 06.02.01.01.A-4, dated October 28, 2016 (ML16302A415), provided two citations to justify not using instrument uncertainty in the initial temperature condition. The response noted that the NRC regulation 10 CFR 50.36 defines an LCO as the "lowest functional capability or performance levels of equipment required for safe operation of the facility." The response also quoted SECY-11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," that LCO limits shall be used for the bounding values as the initial conditions for containment accident analysis." The staff verified the two citations that document the bounding nature of the LCO limits for accident analysis. Therefor the staff found it acceptable that no instrument uncertainty is added to the temperature LCO limit for the containment initial temperature condition. Therefore the staff finds the applicant's supplemental response acceptable and RAI 327-8354, Question 06.02.01.01.A-4 is resolved and closed.

The APR1400 break spectrum analysis determined that a double-ended slot break on the pump discharge side of a cold leg (DEDLSB) with maximum safety injection pump (SIP) flow represents the limiting design basis LOCA. The maximum SI flow rate would result in a higher M&E release and subsequently, a higher containment pressure. In the case of a cold leg or hot leg LOCA with all SI systems available (referred to as Max-SI), generally a safety factor is applied for higher frequency of the emergency diesels that would lead to a higher ECCS pump flow rate. A higher EDG frequency would increase SIPs total dynamic head (TDH), resulting in higher flow rate, which would be conservative. Therefore, on December 3, 2015, the staff issued RAI 327-8354, Question 06.02.01.01.A-5, to ask the applicant to clarify whether the nominal flow rate of the SIPs has been increased to account for a potentially higher EDG frequency. In its response to RAI 327-8354, Question 06.02.01.01.A-5, dated February 3, 2016 (ML16034A202), the applicant stated that a 2 percent higher emergency EDG frequency is considered in the analysis that ensures a maximum SI rate for the limiting LOCA by having a 5 percent margin above the nominal flow rate of the SIP at the pump shutoff and runout head conditions. As the SI flow rate would increase proportionally to the EDG frequency, the effect of the 2 percent EDG frequency is enveloped by the 5 percent margin to maximum SI rate. Therefore, the 2 percent increase of the EDG frequency need not be additionally considered in the SI input of the LOCA analysis for the APR1400; the staff finds the response acceptable, and RAI 327-8354, Question 06.02.01.01.A-5, is resolved and closed.

As stated in the Reference 2, Section 3.6, "Description of Core Reflood Model," following the termination of critical flow, the containment backpressure is assumed to be a constant 4.078 kg/cm^2A (58 psia) throughout the reflood phase. On December 3, 2015, the staff issued RAI 327-8354, Question 06.02.01.01.A-6 asking the applicant to specify the basis for selecting this pressure for input to the FLOOD3 code and explain whether an even lower value for containment back pressure would be more conservative for the M&E release during the reflood phase of the design basis LOCA. In its response to RAI 327-8354, Question 06.02.01.01.A-6, dated February 3, 2016 (ML16034A202), the applicant stated that the assumed constant backpressure value is selected such that it is lower than the calculated containment pressure during the end of blowdown (EOB) and when the containment peak pressure is reached. The applicant also provided two tables summarizing the peak, minimum, and EOB pressures for DEDLSB and double-ended suction leg slot break (DESLSB) with minimum and maximum SI, calculated for APR1400 DCD design (using GOTHIC code) and Shin Kori 3&4 design (using CONTEMPT-LT/028 code). However, the staff noted that the lowest containment pressure is 57.157 psia, which is even lower than the 58 psia used for DEDLSB with maximum SI for APR1400 DC. In the July 7, 2016 public teleconference, the staff asked the applicant to justify not using a more conservative 57 psia value. The applicant's supplemental response to RAI 327-8354, Question 06.02.01.01.A-6, dated November 9, 2016 (ML16314E543), stated that the containment pressure does drop below 58 psia around 50 seconds for DEDLSB with maximum SI, but that the duration of time for the containment pressure drop below 58 psia is relatively short compared to the time period starting from the EOB to the time of peak containment pressure. In addition, the pressure difference between the assumed minimum pressure of 58 psia and the actual minimum pressure of 57.157 psia is relatively small compared with the pressure difference between the assumed minimum pressure and the containment peak pressure of 65.79 psia. The staff noted that the calculated minimum pressure of 57.157 psia occurs past the EOB and, thus, would not significantly affect the M&E release to the containment. The staff also agrees that the assumed input of 58 psia is lower than the containment pressure during most of the time period from the EOB to the time of peak containment pressure and, thus, is conservative. Even though the use of 57 psia as input for the containment back pressure in the LOCA M&E analysis would result in more conservative output as compared to the output when 58 psia is used, the staff concludes that additional conservatism is not necessary because the use of 58 psia as input is already conservative. The staff concludes that the applicant's analysis has justified conservatism in the assumed value of 4.078 kg/cm^{2}A (58 psia) of containment back pressure for the containment peak pressure analysis by using two different computer codes over a sufficient DBA analysis domain. Therefore, the applicant's supplemental response is acceptable and RAI 327-8354, Question 06.02.01.01.A-6 is resolved and closed.

Section A.2.3.1, "Control Volumes," of Reference 2 states that "[t]he containment free volume is calculated by subtracting the volume occupied by equipment inside containment from the gross volume calculated from the building dimensions." The section tabulates the containment atmosphere design value and the value used for LOCA that is conservatively smaller. On December 3, 2015, the staff issued RAI 327-8354, Question 06.02.01.01.A-7, asking the applicant to clarify the basis for the uncertainty that was applied to estimate the containment free volume. This clarification would demonstrate the level of conservatism exercised in the containment free volume for the calculated peak pressure and temperature analyses. In its response to RAI 327-8354, Question 06.02.01.01.A-7, dated December 30, 2015 (ML15364A583), the applicant described the additional safety margin of 1.4 percent it applied to account for the uncertainties in calculating the free volume due to complex shapes of the internal structures and components in the containment building, and the variance due to structural dimension changes caused by design and construction improvements. The design value (3,172,175 ft^3) presented in the containment atmosphere table in Section A.2.3.1 of the TR (Reference 2) is the estimated minimum containment free volume. The applicant used a containment free volume 1.4 percent smaller than the design value in the containment pressure and temperature analyses for additional conservatism. The staff finds the degree of conservatism in using 98.6 percent of the design value as free containment volume to be acceptable, as a smaller volume will tend to produce a higher peak containment pressure. RAI 327-8354, Question 06.02.01.01.A-7 is resolved and closed.

On December 3, 2015, the staff issued RAI 327-8354, Question 06.02.01.01.A-8, to ask whether heat transfer to the containment basemat is credited or is conservatively ignored. In its response to RAI 327-8354, Question 06.02.01.01.A-8, dated December 30, 2015 (ML15364A583), the applicant explained that the surface of the containment floor, including the basemat, is assumed to be insulated and heat transfer to the basemat is conservatively neglected in the safety analysis. Because it provides additional conservatism in the analyses, the staff finds the explanation acceptable and, therefore, RAI 327-8354, Question 06.02.01.01.A-8, is resolved and closed.

To satisfy the requirements of GDC 38 and GDC 50 with respect to the containment heat removal capability and design margin, the LOCA analysis should be based on the assumption of loss of offsite power (LOOP) and the most severe single failure (SF) in the emergency power system (e.g., a diesel generator failure), the CHRSs (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The DCD considers failure of one SIP and the failure of one EDG as two potential SFs in the M&E release analysis. A SF is assumed for the minimum SI flow, while no failure is assumed for the maximum SI flow. The failure of one train of CS is considered the SF for containment analysis for both maximum and minimum SI flows. A simultaneous LOOP was also assumed in the determination of the most severe LOCA case. The staff determined that the APR1400 LOCA analysis is based on a LOOP coincident with loss of a CSS train. From an EDG SF standpoint, consideration of no EDG failure, which results in maximum SI flow (max-SI), produces the most severe containment peak pressure and temperature. The case with minimum SI flow (min-SI) is not limiting for containment peak pressure. The staff agrees that the selection of SF for LOCA is conservative and will result in the highest calculated containment pressure.

In addition to satisfying the requirements of GDC 38 and GDC 50 with respect to containment heat removal capability and design margin, the containment response analysis for postulated secondary system pipe ruptures should be based on the most severe single active failure in the CHRSs (i.e., a fan, pump, or valve failure) or the secondary system isolation provisions (i.e., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should be based on a spectrum of pipe break sizes and reactor power levels. As discussed in Section 6.2.1.4 of this SER, the APR1400 MSLB accident analysis using the M&E release with offsite power available is more conservative than with a LOOP, including consideration of the delayed CS actuation due to EDG startup following a LOOP. Thus, loss of a CS train or a main steam isolation valve (MSIV) failure is assumed as the limiting single active failure, without a LOOP, during a postulated MSLB. From the spectrum analyses of break sizes, power levels, and SFs, the applicant determined the limiting MSLB accident with 102 percent reactor power level and one MSIV failure. The staff agrees that the selection of SF for MSLB is conservative and will result in the highest calculated containment pressure or temperature depending on the purpose of the analysis.

To satisfy the requirements of GDC 38 and GDC 50 with respect to the functional capability of the CHRSs and containment structure under LOCA conditions, provisions should be made to protect the containment structure against possible damage from containment external pressure loading that may result from inadvertent operation of CHRSs. The provisions should include conservative structural design to ensure that the containment structure is capable of withstanding the maximum expected external pressure or interlocks in the plant protection system and administrative controls to preclude inadvertent operation of the systems. NUREG–0800 Section 6.2.1.1.A. specifies that the external design pressure margin should be at least 10 percent. If the containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment will provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event. The APR1400 containment is designed to withstand an external pressure loading of 0.28 kg/cm^{2}G (4.0 psig or 27.5 kPa) relative to ambient pressure. An evaluation and associated analyses, provided in DCD Tier 2, Section 6.2.1.1.3.5, “Inadvertent Operation of the Containment Heat Removal Systems,” demonstrate that the containment structure integrity is maintained under maximum external pressure-loading condition. Containment systems that may lower the containment pressure to less than the external atmosphere pressure following inadvertent operation include the spray, purge, and fan cooler systems. The applicant determined that the pressure reduction effects due to the inadvertent operation of the reactor containment fan cooler (RCFC) units or the containment normal purging system are negligible, and the limiting event for minimum containment pressure design is an inadvertent actuation of the CS system in a sealed containment. The applicant’s analysis summarized in the DCD is based, conservatively, on no heat transfer from the containment structure, no volume reduction due to the addition of spray water, and disregarding all of the heat sources within containment. Dalton’s Law is applied to determine the final containment pressure. The assumed initial conditions for this analysis are listed in Table 6.2.1-24. Conservative values are used for initial pressure, temperature, and relative humidity, spray water temperature to minimize the resultant external pressure. Following a limiting containment pressure reduction event, the calculated maximum external pressure of 0.25 kg/cm^{2}G (3.54 psig or 24.52 kPa) is less than the external design pressure of 0.28 kg/cm^{2}G (4.0 psig or 27.5 kPa). The staff also performed an independent confirmatory analysis that validated the applicant’s maximum external pressure calculations and demonstrated that the results documented in the DCD were conservative. Therefore, the staff finds that, in accordance with the requirements of GDC 38 and GDC 50, it has been demonstrated that the APR1400 containment design external pressure provides more than a 10 percent margin above the calculated maximum external pressure.

In accordance with the requirements of GDC 13, GDC 64, and 10 CFR 50.34(f)(2)(xvii) (for those applicants subject to 10 CFR 50.34(f)), instrumentation capable of operating in the post-accident environment shall be provided to monitor the containment atmospheric pressure and temperature and the sump water level and temperature following an accident. The instrumentation shall have adequate range, accuracy, and response to ensure that the above parameters can be tracked and recorded throughout the course of an accident. Item II.F.1 of NUREG–0737 and NUREG–0718, and BTP 7-10, “Guidance on Application of Regulatory Guide 1.97,” needed to be referred to. DCD Tier 2, Section 6.3.5, “Instrumentation Requirements,” describes the instrumentation requirements that include the design criteria, system actuation signals, and instrumentation for operation. However, the accident monitoring instrumentation is described in DCD Tier 2, Section 7.5.1.1, “Accident Monitoring Instrumentation.” The technical review findings regarding the APR1400 post-accident containment and sump monitoring instrumentation are documented in Section 7.5.4.1 of this SER.

In accordance to 10 CFR Part 50, Appendix K, I.D.2, the minimum calculated containment pressure shall not be less than that used in the analysis of the ECCS capability. The minimum calculated containment pressure of the APR1400 containment design is reviewed under SRP Section 6.2.1.5 and the findings are documented in this Section 6.2.1.5 of this SER.

In accordance with GDC 4, containment internal structures and system components (e.g., reactor vessel, pressurizer, SGs) and supports shall be designed to withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment subcompartments. The related technical review for APR1400 was performed under SRP

Section 6.2.1.2, “Subcompartment Analysis,” and the findings are documented in Section 6.2.1.2 of this SER.

In meeting the requirements of 10 CFR 50.34(f)(3)(v)(A)(1), applicants subject to this section shall evaluate an accident that releases hydrogen generated from a 100 percent fuel clad metal-water reaction. The evaluation shall demonstrate that the appropriate article for SLSL C limits (considering pressure and dead load only), for either concrete or steel containments, from ASME Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from hydrogen burning in containment shall be analyzed. The related technical review for APR1400 was performed under SRP Section 6.2.5 and the findings are documented in Section 6.2.5 of this SER.

ITAAC: The ITAAC provided for containment, located in DCD Tier 1, Table 2.2.1-2, “Nuclear Island Structures ITAAC,” and Table 2.11.1-2, “Containment Structure ITAAC,” did not address some of the key physical parameters relied on in the safety analyses. Substantial changes to these parameters could adversely impact the safety analyses in Chapter 6. No ITAAC was provided to confirm that the as-built containment volume is conservative with respect to the value assumed in the containment peak pressure analyses in DCD Tier 2, Section 6.2.1. DCD Tier 2, Table 6.2.1-3, lists the principal containment design parameters, one of which is the net free volume of the containment (88,576 m^3 (3,128,000 ft^3)). Because the containment volume is a key assumption in the containment pressure analyses (both for calculating peak pressures in Section 6.2.1.1.A and the minimum pressure in Section 6.2.1.5), the as-built value shall be verified to be conservative with respect to the value assumed in the safety analyses. In addition, no ITAAC was provided to ensure that the as-built containment subcompartment accessway dimensions are at least as large as those assumed in DCD Tier 2, Section 6.2.1.2. Therefore, on December 9, 2015, the staff issued RAI 330-8418, Question 14.03.11-3 (ML15343A333), to address this issue.

DCD Tier 2, Section 6.2.1.1.1, “Design Bases,” states that the containment design pressure will be 4.218 kg/cm^2G (60 psig or 515 kPa). The containment design pressure will be verified by inspections and by a pressure test as required by ASME Code, Section III, as described in DCD Tier 1, Table 2.2.1-2 item 2.c. The allowable design leakage rate for containment is specified in DCD Tier 2, Table 6.2.1-3 as 0.1 weight percent of the containment air per day. This value is also reiterated in the TS, as required by 10 CFR Part 50, Appendix J, in DCD Tier 2, Section 5.5.16, “Containment Leakage Rate Testing Program,” and the bases for Section 3.6.1, “Plant Design for Protection against Postulated Piping Failures in Fluid Systems Inside and Outside the Containment.” This value is verified by ITAAC located in DCD Tier 1, Table 2.2.1-2 item 2.d, which involves inspections and leak rate testing in accordance with 10 CFR Part 50, Appendix J.

Following a high energy line break within containment, containment internal heat structures (heat sinks) are important in condensing steam from the containment atmosphere and storing energy. This phenomena influences the containment pressure and temperature following an accident. Although ITAAC for performing a peak pressure analysis with acceptable margin existed, no ITAAC was provided to verify that the heat sink inventory conforms to the analysis described in DCD Tier 2, Section 6.2.1, and would provide reasonable assurance that the key assumptions made in the containment pressure analyses, such as the area of the containment heat sinks, were valid for the as-built design. Therefore, on November 2, 2015, the staff issued RAI 286-8340, Question 06.02.01.01.A-1 (ML15306A470), to address this issue. In its response to RAI 286-8340, Question 06.02.01.01.A-1(ML15328A323), dated November 24, 2015, the applicant stated they would provide an additional ITAAC in Table 2.11.1-2 to ensure that the heat sink areas and compositions were conservative with respect to the containment pressure analyses. Further, in its response to RAI 330-8418, Question 14.3.11-3 (ML16012A559), dated January 12, 2016, the applicant revised the ITAAC in Table 2.11.1-2 to include inspections of the containment volume and subcompartment accessway dimensions, to ensure that the as-built design parameters are conservative with respect to the containment analyses. The applicant's revision of the ITAAC is acceptable because it ensures that the key design parameters (the efficacy of the heat sinks and various containment dimensions) in the as-built design will be acceptably conservative with respect to the design basis containment analyses; therefore, the applicant's responses to RAI 286- 8340, Question 06.02.01.01.A-1 and RAI 330-8418, Question 14.3.11-3, are acceptable to the staff and are considered resolved. The staff is tracking these revisions to the DCD as **Confirmatory Items 06.02.01.01.A-1** and **14.3.11-3**.

TS: Safety analyses in DCD Tier 2, Section 6.2.1, rely on initial containment conditions, including leakage, pressure, and temperature being within an assumed range. Regular verification of these parameters provides assurance that initial conditions at the outset of a postulated accident will remain within an acceptable range. The containment leakage rate is verified below the allowable value by surveillance requirement (SR) 3.6.1.1. Normal containment pressure, which should be between -0.1 psig and 1.0 psig (100.6 kPa to 108.2 kPa), is verified within range by SR 3.6.4.1. For the purposes of calculating peak containment pressure, which assumes an initial condition of 1.0 psig, staff finds this acceptable. Normal containment air temperature is verified by SR 3.6.5.1. Because the maximum allowed value is 49 °C (120 °F), which corresponds to the initial condition assumed in the containment peak pressure and temperature analyses, staff finds this acceptable.

6.2.1.1.5 Combined License Information Items

There are no COL information items associated with Section 6.2.1.1 of the APR1400 DCD.

6.2.1.1.6 Conclusion

The NRC regulations and the associated acceptance criteria for this area of review are given in NUREG–0800, Section 6.2.1.1.A. Review interfaces with other SRP sections can also be found in NUREG–0800, Section 6.2.1.1.A. The acceptance criteria are based on meeting the relevant regulatory requirements that include GDC 13, GDC 16, GDC 38, GDC 50, GDC 64, and 10 CFR 52.47(b)(1).

The staff reviewed the APR1400 standard design for RCB using the NRC regulations and the associated acceptance criteria provided in NUREG–0800, Section 6.2.1.1.A. The staff's review of the compliance of the APR1400 RCB design with the associated GDC and other regulatory requirements led to issuing several RAIs to the applicant. All RAI responses and subsequent supplemental responses were received and all open items were resolved. However, as discussed in Section 6.2.1.1.4 of this SER, several of the RAI responses are being tracked as confirmatory items. The staff concludes that the SER Section 6.2.1.1 remains incomplete pending satisfactory resolution of the confirmatory items identified in staff's technical evaluation in this section. The staff will update Section 6.2.1.1 of this SER to reflect the final disposition of the confirmatory items in the DCD application.

The staff finds that GDC 16 has been met since in the APR1400 DCD, the applicant has shown that the calculated containment pressure is always 10 percent below the containment design pressure, for all limiting breaks. Additionally, the applicant has also met GDC 38. The applicant has shown that containment pressure is reduced to less than 50 percent of the peak calculated pressure for the design basis LOCA within 24 hours of the postulated accident initiation. Confirmatory analyses performed by the staff, agree with the applicant's conclusions regarding the acceptability of the containment design.

GDC 50 has been satisfied because the applicant has assumed a LOOP, and has performed parametric evaluations on single failures in the emergency power system (LOCA), and single active failures (secondary system analysis) to determine the most limiting accident scenarios. In order to satisfy the requirements of GDC 38 and 50 to ensure the functional capability of the containment heat removal systems and containment structure from external pressure conditions, the applicant has performed a limiting analysis in order to comply with this requirement. The applicant has complied with all requirements established in SRP Section 6.2.1.1.GDC 13 requirements on instrumentation and control for the containment and associated systems are reviewed and findings are made in Section 7.3, "Engineered Safety Feature Systems," of this SER. Similarly, compliance to the GDC 64 requirements on monitoring radioactivity releases is covered under Sections 11.3, "Gaseous Waste Management System," and 11.5, "Process Effluent Radiation Monitoring and Sampling Systems," of this SER.

The staff has reviewed the containment systems ITAAC in the APR1400 DCD Sections 6.2.1.1, 6.2.1.3, and 6.2.1.4 to the requirements of 10 CFR 52.47 (b)(1). The detailed staff technical evaluation of the associated ITAAC is included in Sections 6.2.1.1, 6.2.1.3, and 6.2.1.4 of this SER. The staff used the acceptance criteria defined in SRP Sections 14.3 and 14.3.11, and the ITAAC related guidance in Regulatory Guide 1.206 Sections C.1.14.3 and C.II.1.2.11. The staff finds that the containment systems ITAAC provided in Tables 2.2.1-2, 2.11.1-2, 2.4.3-4 are necessary and sufficient to verify the DCD Tier 1 design commitments for the containment heat sink dimensions, safety injection tanks discharge and safety injection pumps flow characteristics, containment volume and leak rate testing. The staff finds reasonable assurance that if the proposed ITAAC are performed and the acceptance criteria are met, the as-built containment design parameters would be in conformity with the certified design with respect to the parameter values used in the design-basis containment safety analyses. Based on this review and a review of the selection methodology and criteria for the development of the Tier 1 information in the DCD, the staff concludes that the top-level containment functional design commitments and performance characteristics of the SSCs for the containment structural integrity and function as a barrier against uncontrolled fission products release are appropriately described in the DCD Tier 1, and the Tier 1 information is acceptable. Consequently the staff finds that the APR1400 containment systems design meets the requirements of 10 CFR 52.47 (b)(1).

6.2.1.1.7 References

1. GOTHIC Thermal Hydraulic Analysis Package Technical Manual, Version 8.0(QA), NAI 8907-06, Revision 19, Numerical Applications, Inc., January 2012.

2. KHNP Technical Report (TR) APR1400-Z-A-NR-14007-P/NP, "Mass and Energy Release Methodologies for LOCA and MSLB," Revision 1, March 31, 2017 (ML17193A959)

3. Schmitt, R. C., et al., "Simulated Design Basis Accident Tests of the Carolinas Virginia Tube Reactor Containment Final Report," IN-1403, Idaho Nuclear Corporation, Idaho Falls, ID, 1970

4. KHNP Topical Report APR1400-F-M-TR-13001-P, "PLUS7 Fuel Design for the APR1400," Revision 1, August 11, 2017 (ML17223B420)

5. KHNP Topical Report APR1400-F-A-TR-12004-P, "Realistic Evaluation Methodology for Large-Break LOCA of the APR1400," Revision 1, August 9, 2017 (ML17240A223)

6. KHNP Technical Report APR1400-Z-A-NR-14011-P, “Criticality Analysis of New and Spent Fuel Storage Racks,“ Revision 1, February 28, 2017 (ML17094A189)

6.2.1.2 Subcompartment Analysis

6.2.1.2.1 Introduction

Reactor containments generally contain a number of enclosed spaces, denoted here in this report as subcompartments, which house high energy piping and are not fully open or have limited flow area to the larger containment volume. In GDC 50, of Appendix A to 10 CFR Part 50, these are referred to as internal compartments. In the event of a pipe rupture, a short-term pressure pulse would exist inside a containment subcompartment. This pressure transient would produce a pressure differential across the walls of the subcompartment, which would generally reach a maximum value within the first second after blowdown begins. The magnitude of this maximum value is a function of several parameters, which include blowdown M&E release rates, subcompartment volume, vent area, and vent flow behavior. GDC 50 requires that subcompartments be designed to mitigate against the effect of this differential pressure and ensure they can withstand the resultant loading.

6.2.1.2.2 Summary of Application

DCD Tier 1, Section 2.11.1, contains two ITAAC related to this Section.

DCD Tier 2, Appendix 3.8A, “Structural Design Summary,” contains a description and diagrams of the containment arrangement, which provide an outline regarding the locations of the containment subcompartments. More detailed plan views are located in DCD Tier 2, Figures 6.2.1-21, “Horizontal Section View and Nodal Model of the Steam Generator Subcompartment,” through 6.2.1-25, “Plan Views and Nodal Models of the Letdown Heat Exchanger and Valve Subcompartments.” DCD Tier 2, Section 6.2.1.2, provides a description of the subcompartment structures inside the containment and the loadings they are subjected to. The APR1400 containment design contains multiple subcompartments, listed and discussed in further detail below in Section 6.2.1.2.4 of this SE. These subcompartments are designed to withstand the forces they are subjected to during a postulated pipe break, either by design of the components within the subcompartment or via openings from the subcompartment into the larger containment volume sized such that the differential pressure does not stress the design limits. Pipe restraints are not credited in reducing the area of high energy line breaks.

Subcompartments containing only piping segments subject to leak-before-break (LBB) considerations are omitted from the analysis as LBB piping is expected to indicate leakage before rupture. The M&E releases to the subcompartments are tabulated and described in DCD Tier 2, Section 6.2.1.2.3, “Design Evaluations.” Transient analyses are performed using the COMPARE-MOD1A code, and the resultant pressure loadings on the subcompartments are displayed in DCD Tier 2, Figures 6.2.1-26, “Steam Generator Subcompartment Pressure Response,” through 6.2.1-31, “Letdown Heat Exchanger Valve Subcompartment Pressure Response,” and the maximum differential pressures in each subcompartment are tabulated in DCD Tier 2, Tables 6.2.1-26 through 6.2.1-31 of the DCD. Subcompartments are nodalized so that the nodal boundaries coincide with flow restrictions, and the nodal models are summarized and displayed in Section 6.2.1.2.3 of the DCD Tier 2.

6.2.1.2.3 Regulatory Basis

The relevant requirements for the Commission regulations for subcompartment analysis, and the associated acceptance criteria, are given in the SRP Section 6.2.1.2, "Subcompartment Analysis," of NUREG–0800. The application is required to satisfy the following:

- GDC 4, "Environmental and Dynamic Effects Design Bases," of Appendix A to 10 CFR Part 50, which requires, in part, that the applicant take provisions to accommodate and appropriately protect SSCs important to safety against the environmental conditions, including dynamic effects, that may result from normal operation, maintenance, testing, equipment failures and postulated accidents.

- GDC 50, "Containment Design Basis," which requires, in part, that the reactor containment structure and its internal compartments accommodate the calculated pressure and temperature conditions resulting from any LOCA.

Acceptance criteria from SRP Section 6.2.1.2 for meeting the above requirements include:

- Nodalization Schemes. Subcompartment nodalization schemes should be chosen so that there is no substantial pressure gradient within a node. A sensitivity study which includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes should be used to verify the nodalization scheme.

- Initial Thermodynamic Conditions. The initial atmospheric conditions within a subcompartment should maximize the resultant differential pressure. An acceptable model would assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. If the assumed initial atmospheric conditions differ from this model, the selected values should be justified by the applicant.

- Vent Flow Path and Distribution of Mass and Energy Released. Assumptions with regard to the distribution of M&E release should be biased towards maximizing the subcompartment pressure. The vent flow behavior through all flow paths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100-percent water entrainment. In addition, the selected vent critical flow correlation should be conservative with respect to available experimental data.

- Break Flow (from SRP Section 6.2.1.3): The analytical approach used to compute the M&E release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved ECCS analysis.

The staff reviewed the analysis conducted to determine the maximum differential pressure loading the containment subcompartment walls would be subjected to as a result of the most limiting postulated line break within each given subcompartment. The review was conducted in accordance with the guidance in SRP Section 6.2.1.2, "Subcompartment Analysis," and SRP Section 6.2.1.3.

6.2.1.2.4 Technical Evaluation

In accordance with SRP 6.2.1.2, subcompartment analysis is performed by: selecting appropriate subcompartments within the containment, identifying high energy piping within each subcompartment, applying LBB criteria to narrow the selection of piping down to high energy lines in which a postulated rupture could occur, and analyzing the appropriate break(s) in each subcompartment to determine the differential pressure loading resulting from the break(s). Guidance in SRP Section 6.2.1.2 states that at the construction permit stage of a review, a 40-percent pressure margin should exist between the calculated pressure and the design pressure, and staff has found that this margin should be the target for the DCD stage of review. Performing subcompartment analysis in this fashion meets GDC 4 as it relates to the ability of SSCs in containment to withstand the dynamic effects resulting from a postulated break, and GDC 50 as it applies to the containment subcompartments withstanding calculated pressures following a postulated LOCA.

As discussed in DCD Tier 2, Section 6.2.1.2, the applicant applied the leak-before-break concept to the reactor coolant system (RCS) high-energy piping. Conceptually, LBB means that piping, for which LBB has been demonstrated to be applicable by deterministic and experimental methods, would show detectable signs of leakage from potential flaws before a catastrophic failure (break) of the pipe would occur. Application of LBB to the containment subcompartment analysis allows the postulated rupture of large pipes to be precluded from the spectrum of postulated breaks resulting from the potential loading experienced under normal, anticipated transient and safe-shutdown earthquake (SSE) conditions. DCD Tier 2, Section 3.6, “Protection against the Dynamic Effects Associated with the Postulated Rupture of Piping,” summarizes the LBB analysis.

This treatment of LBB is consistent with GDC 4, which states, in part, that “dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.” As such, the application of LBB to RCS piping for subcompartment analysis is acceptable given the applicant demonstrated that the probability of a rupture is extremely low under design-basis conditions. Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” of this SER discusses the staff's evaluation and acceptance of LBB for APR1400.

6.2.1.2.4.1 Selection of Postulated Breaks.

The applicant identified six locations within the containment that could be classified as subcompartments:

- Reactor cavity
- SG subcompartment
- Pressurizer subcompartment
- Pressurizer spray valve subcompartment
- Regenerative heat exchanger subcompartment
- Letdown heat exchanger and valve subcompartment

Because the applicant applied LBB to RCS piping, the reactor cavity contains no piping that requires analysis of dynamic effects of a rupture associated with subcompartment analysis. The LBB analysis for the piping is described in DCD Tier 1, Section 3.6.3.1. The staff found the applicant identified appropriate subcompartments containing high energy lines in the list above. Other potential high energy line breaks in the containment are sufficiently open to the larger bulk of the containment volume such that no dynamic effects due to local pressurization are expected to occur.

In the subcompartment analysis, most of the compartments have access doors or openings where the subcompartment interacts with the containment atmosphere. However, no additional information on the modeling of these pathways is provided (with regards to partial blockages due to signage or doors, or changes in the flow area as a result of blowout). Therefore, on June 23, 2015, the staff issued RAI 52-7832, Question 06.02.01.02-3 to address this issue (ML15174A090). In its response to RAI 52-7832, Question 06.02.01.02-3, August 6, 2015 (ML15218A302), the applicant documented the type of wire mesh doors used as well as the assumed blockage ratios (30 percent blocked for gratings and 35 percent blocked for doors). In addition, there are no features in the subcompartments that change based on loading (such as blowout dampers or overpressure panels). The staff used this information in the confirmatory analysis and views these as appropriate assumptions for the conditions described in the plant.

DCD Tier 2, Section 6.2.1.2.2, Revision 0, “Design Features,” was inconsistent with its description and depiction of the nodalization of the regenerative heat exchanger room when compared with DCD Tier 2, Tables 6.2.1-29, “Regenerative Heat Exchanger Subcompartment Nodal Description,” and 6.2.1-34, “Regenerative Heat Exchanger Subcompartment Vent Path Description.” The room is described as having only a single access to the containment atmosphere but is modeled as having no access to the containment atmosphere and instead accesses a room, the In-Core Instrumentation (ICI) chase, with two pathways to the containment atmosphere. Therefore, on June 23, 2015, the staff issued RAI 52-7832, Question 06.02.01.02-4 to address this issue (ML15174A090). In its response to RAI 52-7832, Question 06.02.01.02-4, dated August 6, 2015 (ML15218A302), the applicant clarified the nature of the subcompartment, renaming the accessway to “Corridor below Regenerative Heat Exchanger Subcompartment,” and revising DCD Tier 2, Section 6.2.1.2.2, to state that two access openings and one vent opening are present, as the nodalization suggests. These changes are reflected in Revision 1 of the DCD, and are acceptable, as they corresponds to the plant layout as specified in the documents audited by the staff (ML17037A756).

6.2.1.2.4.2 Calculation of Mass and Energy Release

In each of the subcompartments mentioned above, there is at least one high energy piping segment. Some of the high-energy piping is part of the RCS and therefore is subject to LBB considerations; the applicant is not required to analyze LBB piping for transient subcompartment pressurization. The applicant analyzed the following breaks in the following subcompartments:

- In the SG subcompartment, potential breaks include the economizer nozzle, downcomer nozzle, and blowdown nozzle. RCS lines and the pressurizer surge line, also present in the subcompartment, are not considered due to the application of LBB. Based on the evaluation, the economizer nozzle is the limiting break.

- In the pressurizer subcompartment, potential breaks include the pressurizer spray line and the pilot operated safety relief valve (POSRV) nozzle (limiting). The surge line is omitted due to the application of LBB.

- In the pressurizer spray valve subcompartment, the limiting postulated break occurs in the pressurizer spray line.

- In the regenerative heat exchanger subcompartment, the limiting postulated break is in the CVCS letdown line.

- In the letdown heat exchanger and valve subcompartment, the limiting postulated break occurs in the CVCS letdown line.

The staff conducted a regulatory audit from July 29, 2015 to August 26, 2016 to examine the subcompartment pressurization calculations for each subcompartment listed above (ML17037A756). During the audit, staff reviewed the assumptions, nodalization, junction data, and results. Based on the documentation in the DCD and the material that staff audited, the staff agrees that the spectrum of breaks listed above is a reasonable selection for the design, as the applicant's chosen subcompartments represent a bounding set for the APR1400 design.

No reference was made to the reactor power level or initial primary or secondary coolant conditions used for the M&E releases in DCD Tier 2, Section 6.2.1.2. Therefore, on June 23, 2015, the staff issued RAI 52-7832, Question 06.02.01.02-1 to address this issue (ML15174A090). In its response to RAI 52-7832, Question 06.02.01.02-1, dated August 6, 2015 (ML15218A302), the applicant stated the M&E values used in the subcompartment analysis were based on those used in DCD Tier 2, Section 6.2.1.3 (that is, conservatively biased for the highest M&E release, including an initial condition of 102 percent of rated power). The applicant updated DCD Tier 2, Revision 1, Table 6.2.1-25 to more accurately reflect the conditions used in the analysis.

The staff verified RCS initial conditions for the analyzed subcompartment breaks are consistent with those used in Section 6.2.1.3. These conditions are conservatively biased to yield a M&E release that results in a bounding differential pressure in the subcompartment. For two-phase water-steam mixture, the applicant stated they used the Moody critical flow model and, for subcooled water, the applicant used the Henry-Fauske critical flow model. The Moody model is consistent with the guidance associated with the approved model in SRP Section 6.2.1.2, and the Henry-Fauske model is suitable for subcooled flow, which the staff has found acceptable in similar previous analyses reviewed in other design certification applications by the staff. These methods provide conservative M&E releases for the breaks analyzed in this section.

For the purposes of the analyses conducted in Section 6.2.1.3, the fluid conditions warrant the use of the Henry-Fauske flow model, as all the breaks except the POSRV nozzle and one end of the pressurizer spray nozzle break are subcooled. In the case of the POSRV nozzle and the pressurizer side of the spray nozzle, the Henry-Fauske model yields a M&E flow that results in higher subcompartment pressures and, therefore, is conservative. The applicant updated DCD Tier 2, Section 6.2.1.2.3, Revision 0, to state that the CEFLASH-4A code was used to calculate the M&E releases by using the initial conditions provided and adding a one percent margin to the critical flow. The applicant's changes are acceptable as the staff found CEFLASH-4A acceptable for modeling similar mass and energy releases below in Section 6.2.1.3.4.

6.2.1.2.4.3 Nodalization Analysis

In general, the subcompartments denoted above are nodalized in the applicant's analysis in such a fashion that nodal boundaries occur at areas of flow restriction so that there is no pressure gradient within each node. Nodalization in such a fashion is in accordance with SRP Section 6.2.1.2, which also states that sensitivity studies should be conducted to verify convergence of the nodal scheme.

Sufficient information regarding the nodalization sensitivities was not provided. Although the applicant mentioned that sensitivity studies were performed to arrive at the nodalization depicted in the DCD, no information related to them was provided. Therefore, on June 23, 2015, the staff issued RAI 52-7832, Question 06.02.01.02-2 to address this issue (ML15174A090). In its response to RAI 52-7832, Question 06.02.01.02-2, dated August 6, 2015 (ML15218A302), the applicant stated that subcompartment nodalization sensitivity studies were not performed explicitly. The initial nodalization was deemed sufficient either due to the small resulting differential pressures or the distinct boundaries in the nodes. Due to the homogenous nature of the COMPARE code – that is, each node is assumed to be a perfectly mixed mixture of water, steam and air – an adequate nodalization is particularly important in ensuring the differential pressure developed across the nodes is accurate, as an overly relaxed nodalization can result in averaging regions of high and low pressure. SRP Section 6.2.1.2 directs applicants and reviewers to guidance in NUREG–0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981 (ML13255A427), for developing an appropriate nodalization scheme.

Because the COMPARE code, used by the applicant and further described below, has been reviewed and accepted by the staff for use in subcompartment pressurization analyses, sensitivity studies of the nodalization are not required if the analysis employs appropriate procedures and employs the guidance offered in NUREG–0609. The staff reviewed the nodalization and justifications behind them in detail in the aforementioned regulatory audit (ML17037A756).

The nodalization used for the pressurizer spray valve room, regenerative heat exchanger room, and letdown heat exchanger and valve room are relatively straightforward as small rooms with relatively few small blockages inside the compartments. As such, the nodalizations, which draw boundaries at flow restrictions, are in alignment with the guidance in NUREG–0609. The pressurizer and SG subcompartments are more complex due to the large structures within the compartment and, therefore, are subdivided within the subcompartment into multiple nodes. The nodal boundaries, in accordance with the guidance offered in NUREG–0609, occur at locations where reductions in area occur (i.e., where components are in close proximity to compartment walls), as well as being regularly distributed throughout the compartment so that pressure gradients are not developed between nodes. The staff reviewed the nodalization presented in the DCD, audited the calculations for the SG subcompartment, and found them acceptably conservative. Although staff would prefer a nodalization analysis be performed in cases where a subcompartment is divided into multiple nodes, the analysis produced by the applicant conforms to the guidance in NUREG–0609 and is, therefore, acceptable.

6.2.1.2.4.4 Subcompartment Pressure Analysis

The applicant used COMPARE-MOD1A (Reference 1) to perform the transient subcompartment pressure analysis. COMPARE-MOD1A was developed for the transient calculation of fluid and energy flow through a system of volumes connected by vents and has been used for subcompartment analyses in previous safety analyses accepted by the NRC. The code has also been used in the past by the NRC staff for performing confirmatory analysis.

In calculating the differential pressures experienced by the subcompartments, the applicant made a number of assumptions. The applicant conservatively uses the largest possible expansion loss coefficient of 1.0 and contraction loss coefficient of 0.5, and assumes a break discharge coefficient of 1.0, all of which serve to maximize the energy release. Initial conditions (maximum air temperature and minimum pressure and humidity) are selected such that the differential pressure is maximized as specified in SRP Section 6.2.1.2. Additionally, the applicant assumed 100 percent entrainment of droplets in the release steam in the vent flow. The vent critical flow model selected (thermal homogenous equilibrium) is consistent with the guidance in SRP Section 6.2.1.2 for air-steam-water mixtures. These assumptions all serve to conservatively bias the energy release from the break higher and are therefore acceptable.

During the staff's audit, a number of conservative assumptions not explicitly identified in the DCD were found:

- for calculating friction losses between nodes, the applicant assumed turbulent flow;
- the applicant's analysis conservatively reduced flow areas to account for supports, equipment, and further reduced where a wire mesh door is present; the volumes of some nodes were reduced from their expected value to conservatively account for equipment within the volume;
- M&E inputs into the subcompartment were calculated by the applicant using a Henry-Fauske model with a discharge coefficient of 1.0, and the duration of the M&E release was based on the size of the source (pipe, valve header, etc.).

In general, the staff found the applicant conservatively reduced flow areas and volumes compared with the expected design, resulting in conservatively higher differential pressures between subcompartments.

The staff documented the results of the audit of the APR1400 subcompartment analyses in an NRC staff memorandum (ML17037A756). Based on that audit and in conjunction with the confirmatory analysis conducted and discussed below, the staff concluded that the applicant adequately modeled the subcompartments, as the nodalization schemes adequately reflect the pressure response without distortions and the applicant conservatively modeled the mass and energy release and flow between nodes.

The DCD provided no information related to transient vent paths or blockages in vent paths. Therefore, on June 23, 2015, the staff issued RAI 52-7832, Question 06.02.01.02-3 to address this issue (ML15174A090). In its response to RAI 52-7832, Question 06.02.01.02-3, dated August 6, 2015 (ML15218A302), the applicant gave the dimensions associated with the mesh doors and stated they are accompanied by a 35 percent flow area reduction in the analysis. In addition, the applicant stated no blowout or flow area change is expected following a break and, therefore, no credit is taken for additional flow area.

The staff performed confirmatory analysis for the pressurizer spray valve subcompartment. This compartment was chosen as a representative volume due to the relatively low margins for the differential pressures when compared with the design values of the other subcompartments (the entrance labyrinth to the compartment has a calculated margin of 40 percent, which is the expected margin in SRP Section 6.2.1.2). The staff employed NRC-sponsored computer code MELCOR to perform the confirmatory analysis. MELCOR is an integrated thermal-hydraulics code developed for the NRC, which is capable of analyzing behavior in the reactor coolant system, containment, and various core parameters.

The confirmatory analysis yields a peak differential pressure margin in the spray valve room of 56 percent (as compared to the applicant's calculated value of 44 percent) and a peak differential pressure in the entrance labyrinth of 43 percent (as compared to the applicant's calculated value of 40 percent). This result indicates the applicant's analysis methodology is conservative and that sufficient margin exists between the calculated differential pressure and compartment design pressure.

The subcompartments meet the 40-percent pressure margin specified in SRP Section 6.2.1.2, for the CP stage of a review, which has been found acceptable by the staff for the DCD stage of design reviews. In all cases, the applicant's calculations show margins of at least 40 percent exist.

DCD Tier 1, Section 2.11.1, Revision 1, contains two ITAAC related to this section. Initially, no ITAAC were provided to demonstrate that the as-built facility parameters correspond to the values used in the containment subcompartment analyses. Therefore, the staff issued RAI 18-7900, Question 14.03.11-1 to address this issue. In its response to RAI 18-7900, Question 14.03.11-1, dated July 14, 2015 (ADAMS Accession No. ML15195A441), the applicant provided an additional ITAAC table item (DCD Tier 1, Table 2.11.1-2, item 6) to ensure the calculated peak subcompartment pressures do not exceed the design pressure. Additionally, as part of a subsequent RAI response dated January 12, 2016 (ADAMS Accession No. ML16012A559), the applicant further added an ITAAC (DCD Tier 1, Table 2.11.1-2, item 8) to account for the subcompartment accessway dimensions. In concert, these ITAAC ensure the as-built design parameters are acceptable in comparison to the design assumptions discussed in this section of this SE, and therefore the staff finds the response acceptable.

6.2.1.2.5 Combined License Information Items

There are no COL information items associated with Section 6.2.1.2 of the APR1400 DCD.

6.2.1.2.6 Conclusion

The analysis performed to calculate subcompartment pressures utilized an appropriate nodalization scheme, conservatively biased the initial conditions to maximize the resultant pressure following a postulated break, and conservatively defined junctions in the model so that flow areas into and out of subcompartments were accurately represented. Based on the review of the subcompartment analysis, as documented above, the staff finds that the applicant has met the requirements associated with GDC 4 and GDC 50 by demonstrating subcompartment pressures remain below the design pressure with adequate margin.

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)

6.2.1.3.1 Introduction

A LOCA is defined as a breach of the reactor pressure boundary. The methodology for the analysis of a LOCA includes the containment response for peak pressure, affecting the containment structural integrity. The containment response is analyzed in APR1400 DCD Tier 2, Chapter 6. The goal of the M&E release analyses for postulated LOCAs is to maximize the M&E release as conservative input to the containment design basis analyses that are discussed in Section 6.2.1.1 of this SER. The objective is to ensure sufficient margin to the containment design pressure even in the case of the worst LOCA, as documented in Technical Report (TR) APR1400-Z-A-NR-14007-P/NP, "LOCA Mass and Energy Release Methodology," (Reference 1). The M&E releases for containment subcompartment analyses are discussed in Section 6.2.1.2 of this SER. M&E releases from secondary system piping ruptures are discussed in Section 6.2.1.4 of this SER. The analyses of the M&E release are reviewed to assure that the methodology and data used to evaluate the containment and subcompartment functional design are acceptable for that purpose. A large break LOCA (LBLOCA) is a rupture of a large RCS pipe. The M&E release during such a postulated event causes a sudden increase in the containment pressure over a short period of time. Assumptions that are conservative for containment analyses lead to a rapid removal of heat from the reactor core, RCS piping and internals, and SGs and an ensuing transmittal of the reactor coolant to the containment. Assumptions that are conservative for containment analyses may not be the same as those used in Section 15.6, “Decrease in Reactor Coolant Inventory,” of the DCD to evaluate the performance of ECCS and peak cladding temperature following a LOCA.

6.2.1.3.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this evaluation is provided in DCD Tier 1, Section 2.11.1 and Tier 1, Section 2.4, “Reactor Systems,” which includes Tier 1, Section 2.4.1, “Reactor Coolant System,” and Tier 1, Section 2.4.3. A summary of the technical information is as follows.

The RCS is a safety-related system that removes the heat generated in the reactor core and transfers the heat to the SGs. The RCS forms the pressure and fission product boundary between the reactor coolant and the RCB atmosphere. The SIS is a safety-related system that provides core cooling and reactivity control following most DBAs. It consists of four SIPs, four SITs, and the associated valves and piping. The system delivers water to the RCS following DBA and, as such, makes up the largest contributor to the M&E release for a LOCA calculated in DCD Tier 2, Section 6.2.1.3.

DCD Tier 2: The Tier 2 information associated with this evaluation is provided in DCD Tier 2, Section 6.2.1.3. Additional details are provided in the TR (Reference 1). The report describes the methodology used and demonstrates its applicability to the M&E release analysis for the LOCA as well as secondary system pipe ruptures for the APR1400 design. The LOCA M&E analysis includes evaluation of the advanced design features of APR1400 that include four independent ECCS trains and the IRWST. Each train has one active SIP and one passive SIT equipped with a fluidic device (FD). The use of SIT-FD eliminates the need for low pressure SIPs. An additional feature of the APR1400 design is the delivery of emergency core coolant by direct vessel injection (DVI) nozzles as opposed to the traditional cold leg injection.

LOCA M&E analyses were performed for a spectrum of break locations and SI flows. Double-ended slot breaks were analyzed for suction/discharge legs with maximum/minimum SIP flows. The double-ended hot leg slot break was also analyzed. Failure of one SIP and the failure of one EDG are considered two potential SFs in the M&E release analysis. A SF is assumed for the minimum SI flow, while no failure is assumed for the maximum SI flow. The failure of one train of containment spray is considered the SF for containment analysis for both maximum and minimum SI flows. A simultaneous LOOP was also assumed in the determination of the most severe LOCA case. The limiting case of the M&E release that led to peak containment pressure was the DEDLSB LOCA with maximum SI flow. The report also describes the methodology to analyze the containment pressure and temperature response to the spectrum of high-energy pipe breaks in the RCB, which was used in DCD Tier 2, Section 6.2.1.1 to determine the maximum containment pressure and temperature following the postulated DBAs. The maximum calculated APR1400 containment pressure for the limiting transient is below the containment design pressure of 4.218 kg/cm^{2}G (60 psig or 515 kPa).

The application characterizes LOCA M&E release transients by the following five periods which occur in sequence: blowdown, refill, reflood, post-reflood, and long-term cooling or decay heat period, as briefly described below.

- Blowdown period starts from the accident initiation with a rapid depressurization of the primary system and continues until the RCS pressure equalizes with the containment pressure and there is no pressure differential to drive the break flow. During blowdown, most of the initial primary coolant is released to the containment as a two-phase critical flow established at the break location based on the upstream thermal-hydraulic conditions, and containment pressure increases, and at the end, the break flow goes to zero.

- During refill, the SIS adds water into the reactor pressure vessel (RPV) to refill the reactor vessel lower plenum up to the bottom of the active fuel core. The refill period ends when the water level reaches the bottom of the active core. The refill period is conservatively ignored in the LOCA M&E analysis.

- Reflood period begins when the water from the lower plenum enters the core. During reflood, SIS water floods the active fuel core from the bottom to the top and chugging may occur as the cold water comes in contact with the hot fuel cladding and the water level rises within the core producing a mixture of water and steam. The reflood phase ends when all core locations are no longer blanketed with steam and begin to cool.

- Post-reflood phase begins when SI has reached just below the top of core and ends when the affected SG has transferred heat to the primary side inventory through reverse heat transfer to reach thermal equilibrium with containment vapor and their temperatures equalize. At the beginning of the post-reflood period, the flooded core is being cooled and the decay heat and the sensible heat are removed from the RCS. As the core is being reflooded from the lower core region, more liquid entrainment occurs in the upper plenum, and the level of the two-phase mixture in the pool can reach the hot leg. When the entrained liquid reaches the U-tubes of the SGs, it is vaporized by reverse heat transfer from the secondary side to the primary side. The post-reflood ends when the core is quenched. This specific phase applies only to a cold leg (CL) LOCA as there is no viable flow path to the SGs in a hot leg LOCA.

- During the decay heat period, which begins at the end of post-reflood, the dominant mechanisms for release rates are the decay heat and the sensible heat of all nuclear steam supply system (NSSS) metal. The decay heat period ends when the containment pressure drops back to its initial value.

LOCA M&E release is analyzed using the computer codes CEFLASH-4A for the analysis of the blowdown period, and FLOOD3 for the analysis of the reflood and post-reflood periods. M&E release data for the double-ended (DE) slot breaks for the suction leg, discharge leg, and hot leg break LOCA cases that are listed in Table 6.2.1-1, are given in Part A of Tables 6.2.1-4,through 6.2.1-8, “Double-Ended Hot Leg Slot Break (1.7877m^2 (19.2423ft^2) Break Area).” The LOCA M&E release data in Tables 6.2.1-4 and 6.2.1-6, “Double-Ended Discharge Leg Slot Break – Maximum SIS Flow (0.9121m^2 (9.8175ft^2) Break Area),” are based on the maximum SIS flow and 0.9121 m^2 (9.8175 ft^2) total break area. The LOCA M&E release data in Tables 6.2.1-5, “Double-Ended Suction Leg Slot Break – Minimum SIS Flow (0.9121m^2 (9.8175ft^2) Break Area),” and 6.2.1-7, “Double-Ended Discharge Leg Slot Break – Minimum SIS Flow (0.9121m^2 (9.8175ft^2) Break Area),” are based on the minimum SIS flow and 0.9121 m^2 (9.8175

ft^2) total break area. The hot leg LOCA M&E release data in Table 6.2.1-8 are based on 1.7877 m^2 (19.2423 ft^2) total break area.

ITAAC: The ITAAC associated with this evaluation of APR1400 DCD Tier 2, Section 6.2.1.3 are provided in DCD Tier 1, Table 2.11.1-2, Table 2.4.1-4, "Reactor Coolant System ITAAC," and Table 2.4.3-4, "Safety Injection System ITAAC."

TS: The TS associated with this evaluation of DCD Tier 2, Section 6.2.1.3 are provided in DCD Tier 2, Chapter 16, Section 3.6 and Section 3.5, "Emergency Core Cooling System."

6.2.1.3.3   Regulatory Basis

The Commission regulations and the associated acceptance criteria for this area of review are given in NUREG–0800, Section 6.2.1.3. Review interfaces with other SRP sections can also be found in NUREG–0800, Section 6.2.1.3. The acceptance criteria are based on meeting the relevant regulatory requirements as summarized below.

- GDC 50, requires that the containment structure and its associated systems be designed to withstand the calculated pressure and temperature conditions resulting from any LOCA, without exceeding the design leakage rate and with sufficient margin.

- The regulation in 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," provides requirements to assure that all sources of energy have been considered in analyzing a LOCA event. These include but are not limited to: reactor power, decay heat, stored energy in the core, stored energy in the RCS metal, including the reactor vessel internals, metal-water reaction energy, and stored energy in the secondary system, including the SG tubing and secondary water. Calculations of energy release should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph 1.A. Additional conservatism should be included to maximize the energy release to containment during the blowdown and reflood phases of a LOCA event. The requirements of paragraph 1.B in 10 CFR Part 50, Appendix K, concerning prediction of clad swelling and rupture should not be considered in order to maximize the energy available for release from the core.

- The regulation in 10 CFR 52.47(b)(1) requires that the DC application contain proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations.

6.2.1.3.4   Technical Evaluation

The APR1400 M&E analysis is based on the Combustion Engineering (CE) methodology, which has been utilized by the CE plants since their inception. The TR (Reference 1) provides the details of the methodology used by the applicant for the APR1400 design. The report also describes the methodology to analyze the containment pressure and temperature response to the spectrum of high-energy pipe breaks in the RCB, discussed in Section 6.2.1.1 of this SER, to determine the maximum containment pressure and temperature following the postulated DBAs. The APR1400 methodology conforms to the regulatory guidelines for conservatism (SRP Section 6.2.1.3) and takes only partial credit for the RCS and containment active safety features. For instance, regardless of the assumed SF, no credit is taken for any containment fan coolers. APR1400 is equipped with four EDGs to power the four independent ECCS trains and two trains of CSs, with the number of CS trains credited in all events is reduced to one due to the assumed SF. There are no cross-tie lines among the injection paths. The methodology also includes two cases with respect to the assumed SF, max-SI and min-SI. It accounts for potential SFs in both the RCS and containment. The methodology conforms to ANSI/ANS 56.4-1983, “Pressure and Temperature Transient Analysis for Light Water Reactor Containments,” guideline by assuming a concurrent LOOP. In the max-SI case, the SF for the double-ended slot break on the discharge side of the RCP could be the loss of one train of service water, rendering one train of the containment fan coolers unavailable. However, the APR1400 methodology does not take credit for the heat removal of any containment fan coolers. Therefore, the SF for the max-SI case can be viewed as no failure. In the min-SI case, the SF for the RCS is the loss of one EDG, which renders a SIP inoperable. Regarding SF for containment, the loss of one train of EDG is assumed for both max-SI and min-SI cases. The staff finds that this assumption is conservative since it would render one train of CS inoperable, which conservatively reduces the heat removal capability of the active containment cooling system.

The staff determined that various phases of a DBA LOCA M&E release transients, as specified by SRP Section 6.2.1.3, are duly considered for the APR1400 containment design including the initial blowdown phase, core reflood phase, post-reflood phase, and decay heat phase for long term cooling. The duration of each phase depends on the assumed SF. For blowdown, reflood, and post-reflood (only CL LOCA) phase durations are on the order of 20, 200, and 360 seconds, respectively.

Appendix K to 10 CFR Part 50, “ECCS Evaluation Models,” requires that all sources of stored and generated energy, both in the RCS as well as in the secondary coolant system, shall be considered in performing the LOCA analyses. RG 1.206, “Combined License Applications for Nuclear Power Plants,” and Acceptance Criterion No. 1A of the SRP Section 6.2.1.3, also highlight this requirement. DCD Tier 2, Section 6.2.1.3.2, “Energy Sources,” identifies the following sources of stored and generated energy in the RCS and secondary coolant system considered in the LOCA analyses: primary coolant, secondary coolant (cold leg break LOCA case), primary side metals (RCS metal, reactor vessel, reactor vessel internals), secondary side metals (cold leg break LOCA case), SI water, reactor power, stored energy in the core, and decay heat. Table 6.2.1-37, “Stored Energy Sources,” of the DCD Tier 2 further elaborates that energy stored in the reactor core, the reactor pressure vessel walls, and the SG tubing are also accounted for in the LOCA M&E analysis, as explicitly suggested by the SRP Section 6.2.1.3. The staff review of the DCD information provides assurance that the applicant’s analysis accounts for all sources of energies in a DBA LOCA in accordance with 10 CFR Part 50, Appendix K.

The regulation in 10 CFR Part 50, Appendix K, also requires the energy from the metal-water reaction within the core to be considered in the LOCA M&E analysis. The APR1400 DCD does not account for the metal-water reaction energy in the LOCA containment analyses. According to Table 1.9-2, metal-water reaction energy is not included in the mass/energy release, as it has a small effect on the containment pressure. Therefore, the metal-water energy is not included in the mass/energy source terms in Tables 6.2.1-4, through 6.2.1-8, “Double-Ended Hot Leg Slot Break (1.7877m^2 (19.2423ft^2) Break Area.” DCD Tier 2, Section 6.2.1.3.8, “Metal-Water Reaction,” describes a bounding calculation assuming a maximum allowable 1 percent zirconium water reaction, which produces 0.351x 10^6 kcal (1.3933 x 10^6 Btu). This metal-water reaction energy is less than 0.6 percent of the total amount of energy released up to the time of peak containment pressure during the limiting LOCA and increases the containment peak pressure by 0.24 percent. The staff ascertained that that the maximum fuel clad temperature of

808.2°C (1,486.7°F) during the blowdown phase is below the range of the correlations that are developed for metal-water reaction energy release. The staff determined that the Baker-Just equation, which is recommended by Appendix K, is based on the data taken at 1000°C - 1850°C (1800°F - 3350°F). As the lower limit of the applicability of the correlation is 1000°C (1800°F), the maximum fuel clad temperature of 808.2°C (1,486.7°F) is too low for the release of any significant metal-water reaction energy during the DBA LOCA M&E analysis. Even if the limiting metal-water reaction is assumed to occur, the energy release will not be significant and would have little effect on the containment pressure. Therefore, the staff accepts neglecting the energy from the metal-water reaction.

The staff reviewed the APR1400 M&E release analysis methodology for the LOCA described in the TR (Reference 1) in accordance with the acceptance criteria described in SRP Section 6.2.1.3 to establish the limiting case of the M&E release for the APR1400 containment. The APR1400 methodology involves the M&E release models for the blowdown, reflood, and post-reflood phases and the modifications in the calculation process for reflecting the advanced design features of the APR1400. The review involves evaluating three computer codes used in the M&E release analysis: CEFLASH-4A (Reference 2), FLOOD3 (Reference 3), and GOTHIC (Reference 4) and their capability to conservatively model the bounding LOCA postulated accidents to demonstrate sufficient margin in the APR1400 containment design limits. The objective of the review is to establish the applicability and adequacy of the methodology and assess the supporting assumptions, model inputs, and analysis results. The M&E release data for the blowdown are computed using CEFLASH-4A and for reflood and post-reflood phases are computed using FLOOD3. The M&E corresponding to the decay heat phase is calculated by using the EPRI-sponsored GOTHIC computer code. The containment analysis is performed subsequent to the M&E analysis using GOTHIC, which calculates containment pressure and temperature in a LOCA.

A review of the computer codes (References 2-4) and their applicability to APR1400 LOCA M&E methodology was performed. Calculations of the energy available for release are done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A. The staff ascertained that additional conservatisms are included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA. The conservatisms built in the APR1400 LOCA M&E analysis are summarized below:

- Assuming a double-ended (DE) split versus DE guillotine break, per the approved CE methodology.

- Using a small time step especially for FLASH-4A, which would avoid obscuring the peak M&E during the blowdown phase.

- Ignoring the refill phase of LOCA, which reduces time to the higher reflood flow rate and, thus, maximizes the release rate to the containment.

- Assuming all RCS metal as carbon steel ($k_{CS} \approx 2k_{SS}$) in CEFLASH-4A.

- Assuming no thermal resistance with heat conductors ($h_{Wall} = \infty$) in CEFLASH-4A.

- Assuming no auxiliary feedwater (AFW) flow in CEFLASH-4A calculations.

- Ignoring SI until the end of the blowdown phase, which increases the enthalpy of the break flow rate.

- Not taking credit for condensation of steam in the downcomer after the SITs are emptied in FLOOD3 calculations.

- Assuming the initial core power to be 102 percent of its normal power which includes the instrumentation error.

LOCA M&E analyses were performed for a spectrum of postulated break sizes, break locations, SFs, and SI flows to determine the most severe design basis LOCA. In this regard both hot-leg and cold-leg breaks, located on the suction and discharge sides of the RCP, were analyzed with maximum/minimum SIP flows. All of the breaks are assumed to be DE slot type. By using the DE break, flow from both sides of the pipes enters the containment, and, by using the slot type, both sides of the pipe are allowed to communicate across the break. Table 3-1, "Containment P/T with 1 Percent Metal-Water Reaction," in the TR (Reference 1) identifies a DEDLSB, on the vessel side of a cold leg, with maximum SI flow as the limiting case of the M&E release that led to the most severe DBA LOCA with the peak containment pressure. The total break area for the DE slot break is 0.9121 m^2 (9.8175 ft^2). Per APR1400 DCD Tier 2, Table 6.2.1-2, this would lead to a peak containment pressure of 3.592 kg/cm^2G (51.09 psig or 453.58 kPa) and a peak containment atmosphere temperature of 134.59 °C (274.25 °F), as discussed in Section 6.2.1.1 of this SER. The corresponding M&E release data are given in DCD Tier 2, Table 6.2.1-6. The peak pressure and temperature analysis of the limiting DEDLSB LOCA case is based on the DCD Tier 2, Table 6.2.1-6 M&E release data, and is evaluated in Section 6.2.1.1 of this SER.

GDC 50 and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," require that the selected combination of power distribution shape and peaking factor shall be the one that results in the most severe calculated consequences for the spectrum of postulated breaks and SFs that are analyzed. NUREG–0800, SRP Section 6.2.1.3, on "Break Size and Location," suggests that "[c]ontainment design basis calculations should be performed for a spectrum of possible pipe breaks, sizes, and locations to assure that the worst case has been identified." As discussed in Section 6.2.1.1 of this report, the APR1400 DCD identified a double-ended discharge leg slot break (DEDLSB) on the pump discharge side of a cold leg with maximum SI flow as the limiting case for M&E release that leads to the most challenging DBA LOCA with the peak calculated containment pressure. The maximum SI flow rate would result in a higher M&E release and subsequently, a higher containment pressure. However, the staff noted that a double ended guillotine break was not analyzed. The APR1400 methodology followed the traditional assumption of a hot leg piping slot break of the same flow area as that of a DE guillotine piping rupture and does not report any DE guillotine break analysis results. Table 3-1, "Containment P/T with 1 Percent Metal-Water Reaction in LOCA," in the TR (Reference 1) shows that the double-ended hot leg slot break (DEHLSB) results in the lowest peak pressure compared to all four cold leg breaks. It is not documented in the DCD or the TR whether the DEHLSB was assumed to be the limiting break size for a hot leg break LOCA or it was obtained from a break spectrum analysis. The APR1400 methodology does not satisfy the required break spectrum analysis (small, medium, and large breaks) to identify the most limiting LOCA. This requirement appears to have been interpreted only in terms of hot leg and cold leg breaks as opposed to the break flow areas ranging from small slot to DE guillotine break. The DE break should have been analyzed for the most limiting hot leg break through a series of trial runs to rule out smaller break sizes as being more limiting.

The staff determined that the double ended hot leg slot break (DEHLSB) was assumed to be the limiting break size for a hot leg break LOCA, while no hot leg slot break spectrum analysis was performed. Therefore, on November 3, 2015, the staff issued RAI 290-8336, Question

06.02.01.03-1, to address the gaps in the applicant's break spectrum analysis as double ended guillotine break and hot leg slot break spectrum were not analyzed for the mass and energy release and the subsequent containment response analyses. The RAI asked the applicant to demonstrate that the M&E release and subsequent containment thermal-hydraulic response analyses for DEHLSB are most conservative across the possible hot break spectrum including smaller slot break sizes. In order to meet the break spectrum analysis requirement to identify the most limiting LOCA, the methodology used by the applicant for APR1400 needs to demonstrate that a limiting DE guillotine break would result in less severe thermal-hydraulic conditions in the containment than resulted from the limiting DBA, i.e., DEDLSB with maximum SI flow. In its response to RAI 290-8336, Question 06.02.01.03-1, dated November 4, 2016 (ML15356A226), the applicant presented tables and figures of analyses supporting their conclusions. The provided calculated double-ended hot leg guillotine break (DEHLGB) blowdown M&E release data are compared with those of the DEHLSB case of APR1400 DCD Section 6.2.1.3, which shows that the DEHLGB case has slightly more severe results than the DEHLSB case. The response also compared the calculated blowdown M&E data of the DEDLGB with the DEDLSB case of APR1400 DCD Section 6.2.1.3.

Even though the DEHLGB case is more severe than the DEHLSB case, the DEHLGB peak pressure is less than that of the limiting LOCA case (i.e., DEDLSB), as reported in the APR1400 DCD Tier 2, Table 6.2.1-19, "Summary Results of Postulated Pipe Rupture Analysis." The staff accepts that the DEDLSB case is more severe than the DEDLGB case. Thus, the DEDLSB case in the APR1400 DCD is still the limiting LOCA case.

The staff noted that the applicant analyzed only two smaller break sizes that are 60 percent and 80 percent of the DEHLSB. Small break LOCAs allow time for the primary coolant to absorb energy from the secondary side of the SGs as opposed to the DE breaks, which result in rapid blow down and an insufficient time period for any reverse heat transfer from the SGs. During the July 7, 2016, public teleconference, the staff emphasized the need to further analyze smaller hot leg slot breaks below 60 percent of the DE area, per SRP Section 6.2.1.3, to ensure that no limiting LOCA exists for a smaller break.

The applicant submitted a supplemental response to RAI 290-8336, Question 06.02.01.03-1, dated November 4, 2016 (ML16309A247) that provided tables and figures of additional analyses to support their conclusions. The RAI compared the newly calculated DEHLGB blowdown M&E release data and the resulting peak pressure with those of the DEHLSB case of APR1400 DCD. The staff concluded that for the hot leg even though the guillotine break turns out to be more severe than the double ended slot break, the resulting peak pressure (64.50 psia) is still less than that of the limiting LOCA case documented in the DCD, i.e., 65.786 psia for DEDLSB with maximum SI flow. The RAI response also compared the newly calculated blowdown M&E data of the DEDLGB with the limiting DEDLSB case documented in the DCD, which showed that the DEDLSB case is more severe than the DEDLGB case and is still limiting. The applicant also submitted additional analyses of smaller hot leg slot breaks to ensure that no limiting LOCA exists for a smaller break. The submitted M&E release and the subsequent containment response analysis results for six hot leg slot break cases from 80 percent through 5 percent of the double ended area showed decrease in the predicted containment peak pressure with the reduction in the break area. As such, the staff accepts that the double ended hot leg slot break (DEHLSB) case in the APR1400 DCD is most conservative across the possible hot leg slot break spectrum including smaller slot break sizes. The staff determined that sufficient details on the mass and energy release and containment peak pressure analyses spectrum have been provided to establish the DCD's conclusion that the DEDLSB is the limiting LOCA case for the APR1400 design. The applicant also submitted the mark-ups to appropriately revise the DCD Tier 2, Section 6.2.1.3.1, “Mass and Energy Release Data” and Section 3 of the TR (Reference 1) for the updated spectrum analysis as a part of the RAI response. The staff finds the applicant’s response acceptable, and RAI 290-8336, Question 06.02.01.03-1 is resolved, and the verification of the proposed DCD and TR revisions is being tracked under **Confirmatory Item 06.02.01.03-1.**

The applicant’s APR1400 methodology identifies the DEDLSB with maximum SI flow to be the most limiting LOCA, as documented in Table 3-1 in the TR (Reference 1). It can be postulated that a higher SI rate and flow enthalpy would result in a higher M&E release and subsequently higher peak containment pressure and temperature. Therefore on November 30, 2015, the staff issued RAI 322-8393, Questions 06.02.01.03-2, (ML15334A334), to seek information to address safety concerns about the CEFLASH-4A computer code’s modeling of the APR1400 new features such as FD and DVI line injection for the limiting LOCA analysis for the containment.

Table 3-1 in the TR (Reference 1) shows that, for the limiting cold leg LOCA, it takes 324.1 seconds from the start of the accident until the containment peak pressure is reached. Table 4-2, of the TR (Reference 1) shows that it takes 49.35 seconds for the SIT flow to be turned down to the low-flow rate by the FD. This leaves 274.75 seconds for the remaining SIT flow to contribute to the containment peak pressure. However, it is stated in the TR Section 3.11.1, “Effect of Fluidic Device Controlled by K-Factor,” that the effect of the SIT flow from the FD on steam condensation in the downcomer of the intact loop is conservatively ignored considering the FD low-flow rate to be small. The TR did not provide sufficient clarity with respect to the treatment of the SIT water below the top of the stand pipe of the FD. Therefore, on November 30, 2015, the staff issued RAI 322-8393, Question 06.02.01.03-2 to inquire about whether or not the SIT water inventory is credited in the analysis to enter the RCS after the SIT water level drops below the top of the FD stand pipe. The staff concern is that if no SIT water inventory is credited in the analysis, the extent of conservatism by ignoring steam condensation would be reduced, as the SIT water below the top of stand pipe does not enter the core to pick up heat and transfer into containment for pressurization. In that case, the applicant would need to justify using a less than 100 percent SIT flow as the limiting cold leg break LOCA uses maximum SI. Otherwise, if the entire SIT water inventory is credited in the analysis, the applicant would need to justify the capability of the CEFLASH-4A code for modeling the SIT flow from the FD to account for its contribution to containment peak pressure.

The applicant responded to RAI 322-8393, Question 06.02.01.03-2 (ML16048A406), by letter dated February 17, 2016, and the NRC staff’s evaluation of the response is as follows:

a) The applicant explained that the SIT flow is reduced to low flow only during the post-blowdown phase but not during the blowdown phase. After the FD is turned down, the SIT water inventory below the top of the FD stand pipe enters the RCS in the LOCA analysis. The applicant further explained that the limiting cold leg break LOCA case (i.e., DEDLSB) assumes maximum SIP flow during the post-blowdown phase and the flow rate does not depend on the SIT water inventory below the top of the FD stand pipe. The staff determined that the applicant’s explanation is supported by Table 1 of the response that shows that the total water volume of the four SITs at the end of blowdown is greater than the total SIT water volume below the top of their stand pipes, for the limiting DEDLSB case. This information shows that the FD turndown has not occurred during the blowdown phase.

b) Regarding part 2 of the RAI, the applicant explained that the SIT water inventory below the top of the FD stand pipe is credited to enter the RCS after the FD turndown in the analysis even though the SIT-FD low flow is not credited to condense steam flows in the downcomer. The staff agrees that it is a conservative assumption since the steam flow in the RCS is not decreased by the SIT-FD low flow and the M&E release rate is maximized. The response to RAI 322-8393, Question 06.02.01.03-2, further explained that the modeling of the FD by the CEFLASH-4A code is performed by adding the frictional loss coefficient (k-factor) value of the FD to that of the SI line. Since the FD turndown does not occur during the blowdown phase, the SIT-FD high flow is maintained throughout the blowdown phase. Therefore, CEFLASH-4A code calculates the SIT-FD high flow using the input k-factor of the FD and the SI line.

The applicant subsequently revised DCD Tier 2, Sections 6.2.1.3.4, "Description of Core Reflood Model," and Sections 3.6, "Description of Core Reflood Model," and 3.11.1 of the TR (Reference 1) by including necessary clarifications and submitted the mark-ups for the revisions as a part of the RAI response. The staff finds the applicant's response acceptable, and RAI 322-8393, Question 06.02.01.03-2 is resolved, and the DCD and TR revisions are tracked under **Confirmatory Item 06.02.01.03-2.**

Figure 2-2, "Reactor Vessel and Internals," in the TR (Reference 1) shows the unique APR1400 feature of DVI, delivering SI flow to the downcomer during a LOCA. According to Figure 2-2 of the TR and Figure 5.3-8, "Reactor Vessel Assembly," of the DCD Tier 2, the DVI nozzle elevation is well above the cold leg. The spillage flow rates for APR1400 design are given in Table 4-2, of the TR. However, no information was available on the fraction of the coolant flow that spills out of the break into the containment, so that the spillage flow as a percentage of total SI could be compared with the 25 percent for the traditional PWR cold leg injection. Comparatively, a traditional PWR cold leg SI credits the ECCS injection only into three out of the four cold legs, with the broken leg coolant added to the containment as spillage flow. Therefore, the staff issued RAI 8393, Question 06.02.01.03-3 asking the applicant to provide the fraction of the coolant flow that spills out of the break into the containment. The RAI question also inquired whether the computer codes, CEFLASH-4A and FLOOD3, used for modeling the SI into the RCS by the SITs and the SIPs during the blowdown, refill, and the reflood phases of a LOCA, have been validated and approved to model DVI type injection.

In its response to RAI 322-8393, Question 06.02.01.03-3 (ML16048A406), dated February 17, 2016, the applicant provided the following information for staff's evaluation.

a) The applicant stated that the direct spillage of one out of four SI lines for the traditional PWR cold leg injection is not included in the spillage data since all the SIT water of the APR1400 is injected directly into the reactor vessel through the DVI nozzles instead of the traditional cold leg injection. Table 1 of the RAI response shows the spillage flow as 0 percent, 41.4 percent, and 51.5 percent of the integrated SI flow by the end of blowdown, reflood, and post-reflood, respectively. The response also describes the various spillage components. The staff concludes the APR1400 spillage to be conservative compared to the typical 25 percent spillage for the traditional PWR due to ECCS injection only into three out of the four cold legs.

b) Regarding the modeling of the DVI by the traditional codes CEFLASH-4A and FLOOD3, the applicant stated that the DVI type injection in the LOCA blowdown is modeled in CEFLASH-4A by using nodalization and flow paths similar to other regions in the vessel. The effect of the DVI type injection during the LOCA blowdown phase is similar to that of the cold leg injection (CLI) type with the exception that there is no loss of one train of SI for DVI type injection. Regardless of the SI type, most of the injected SIT water during the blowdown period bypasses the core support barrel and is released to the containment through the break. Based on the information provided in the response, the staff determined that both the LOCA blowdown transients, DVI and CLI, are analyzed using the CEFLASH-4A code without any code model change. In the FLOOD3 code (Reference 3) for the reflood phase, the SI flow is modeled as a boundary condition for the reactor vessel annulus. Since this approach is similar to that used in the FLOOD-MOD2 code (Reference 5), which is the NRC approved version, no additional code validation and approval to model DVI type injection is needed, and the staff questions about the applicability of the computer codes, CEFLASH-4A and FLOOD3, to model the mass and energy release during the blowdown and the reflood phases of a APR1400 LOCA have been addressed.

As a follow-up to RAI 322-8393, Question 06.02.01.03-3, the staff requested the applicant in the July 9, 2016, public teleconference to clarify how spillage in the licensing basis containment M&E release inputs was modeled in order to align it with the values used in the staff's confirmatory calculations. Transient data were provided in the DCD for many of the M&E release components, but spillage data were listed as a single integral line item over a period at the beginning of the transient, rather than on a time dependent basis. During the teleconference, the staff requested the transient spillage data (liquid from the break) in order to determine if there was any impact on conservatism resulting from adding the mass and energy resulting from spillage broken over a similar discrete timescale. Further, the staff noted that the DCD should include the spillage data in transient form. The applicant submitted a supplemental response to RAI 322-8393, Question 06.02.01.03-3, dated November 14, 2016 (ML16319A305) that provided transient M&E release data for spillage. By integrating the transient spillage data, the staff confirmed that it was well within 1 percent of the aggregate spillage, which shows that the spillage has already been accounted for. The staff confirmed that the correct M&E release input is being used for the licensing basis calculations. As a part of the RAI response, the applicant also submitted the mark-ups to appropriately revise the DCD Tier 2, Tables 6.2.1-4 through 6.2.1-7, and Section 3.10 and Table 4-2 of the TR (Reference 1) to include the transient spillage data for the limiting LOCA. The staff concludes that the applicant provided sufficient information about the applicability of the computer codes to the APR1400 design mass and energy release, as well as the spillage data. The staff finds the applicant's response acceptable and RAI 322-8393, Question 06.02.01.03-3, is resolved. The verification of the proposed DCD and TR revisions is being tracked under **Confirmatory Item 06.02.01.03-3.**

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(ii) states that calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. The DCD does not mention using a nucleate boiling correlation for the blowdown phase. However, the TR (Reference 1) mentions that the wall surface HTCs are assumed to be infinite, which the staff finds conservative. SRP Section 6.2.1.3 Acceptance Criterion No. 1C(ii) also states that for surfaces exposed to steam, heat transfer calculations should be based on forced convection. The staff determined that assuming an infinite wall surface HTCs meets this guidance.

Per GDC 50 and Appendix K to 10 CFR Part 50 requirements, to analyze the most severe consequences for the spectrum of postulated pipe breaks sizes, locations, and SFs, SRP Section 6.2.1.3 lays out several acceptance criteria to ensure that containment M&E release calculations are performed for the worst DBA. Therefore, on February 3, 2016, the staff issued RAI 394-8460, Question 06.02.01.03-5, to seek information on the conservative treatment of M&E release calculations for the limiting LOCA analysis from the containment perspective, as discussed below.

SRP Section 6.2.1.3, Acceptance Criterion No. 1, C(ii), states that mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data. Even though the DCD and the TR (Reference 1) mention that the LOCA M&E release is analyzed using the computer codes CEFLASH-4A and FLOOD3 for the blowdown and reflood/post-reflood periods respectively, they do not state whether these codes have been validated against experimental data. The staff issued RAI 394-8460, Question 06.02.01.03-5 to request the applicant to document this information in the DCD. In its response to RAI 394-8460, Question 06.02.01.03-5, dated June 8, 2106 (ADAMS Accession No. ML16160A034), the applicant stated that the mass release calculation model in the CEFLASH-4A code during blowdown period has been demonstrated to be conservative by comparison to experimental data in Section III.C.1.b.(4) of Reference 6. The staff confirmed that a comparison was performed between the CEFLASH-4A analytical results and the experimental results of LOFT semiscale test 850. The reference describes three critical flow formulations used in the CEFLASH-4A for the mass release calculation: Moody correlation, the modified Henry--Fauske correlation, and a combination of the modified Henry--Fauske and Moody correlations. During reflood/post-reflood periods, no critical break flow is predicted and the mass release is calculated in the FLOOD3 code using the flow resistances in primary coolant flow network. The staff accepts that, even though no experimental data are available for the reflood break flow, the FLOOD3 mass release calculation is conservative because the flow resistances were minimized in the calculation of the break flow. The response also submitted the associated markups to revise DCD Tier 2, Section 6.2.1.3.3, and TR, Section 3.5, "Description of Blowdown Model," accordingly. Therefore, RAI 394-8460, Question 06.02.01.03-5 is resolved and the DCD and TR revisions are tracked under **Confirmatory Item 06.02.01.03-5.**

DCD Tier 2, Section 6.2.1.3 makes the following two statements regarding the nature of two-phase flow from the LOCA break: (1) "During blowdown, most of the initial primary coolant is released to the containment as a two-phase mixture[,]" and (2) "The onset of the two-phase release to the containment may or may not occur before the end of reflood; typically, this occurs close to the end of the reflood." On February 3, 2016, the staff issued RAI 394-8460, Question 06.02.01.03-6, to request the applicant to reconcile the two statements and update the DCD accordingly. In its response to RAI 394-8460, Question 06.02.01.03-6, dated May 19, 2016 (ML16142A010), the applicant provided the necessary clarifications. It stated that during the blowdown phase of the discharge leg break or the suction leg break, most of the initial primary coolant in the reactor vessel directly goes to the cold leg break, bypassing the core and the SG, and is released to the containment as a two-phase mixture with relatively low enthalpy. During the blowdown of the hot leg break, the flow from the reactor vessel passes through the core and has relatively slowly increasing enthalpy due to the core decay heat transfer and SIT injection. Meanwhile, the enthalpy of the SG inventory flowing backward to the hot leg break is rapidly increased due to the reverse heat transfer in the SG. The total hot leg break flow is the sum of these two flows that is released to the containment as a two-phase mixture with a higher enthalpy than that of the discharge leg break or suction leg break. During the reflood phase of the discharge leg break or the suction leg break, a significant amount of the SIS water entering the core is carried out of the core by the steaming action of the core-to-coolant heat transfer. This fluid then passes through a SG where reverse heat transfer heats it before it is released to the containment. During the initial part of the reflood period, the residual SG secondary energy is sufficient to convert all of this fluid to superheated steam. Subsequently, as the SGs are cooled by this process, there is not enough heat transfer to boil the fluid off completely, which causes the break flow to change from pure steam to two-phase. Thus, the onset of the two-phase release to the containment due to the cooled down SG may or may not occur before the end of reflood; typically, this occurs close to the end of the reflood. The response also submitted the associated markups to revise DCD Tier 2, Section 6.2.1.3, and TR Section 3.2, “Accident Description for LOCA Mass and Energy Release,” accordingly. The mark-ups stated that for the discharge leg break or the suction leg break, the major portion of the primary coolant in the reactor vessel directly goes to the cold leg break bypassing the core and the steam generator, thus has relatively low enthalpy. For hot leg break, the break flow from the reactor vessel side has relatively slow increasing enthalpy due to the core decay heat transfer and SIT injection. The two-phase mixture break flow is mainly due to the low enthalpy flow from the reactor vessel side during the blowdown phase. The staff found the applicant’s description to be sufficient because it provided additional information and addressed inconsistencies. Therefore, RAI 394-8460, Question 06.02.01.03-6, is resolved and the DCD and TR revisions are tracked under **Confirmatory Item 06.02.01.03-6.**

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(ii) states that calculations of heat transfer from the secondary coolant to the SG tubes for PWRs should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam. It was not clear whether the application has followed the suggested approach. On February 3, 2016, the staff issued RAI 394-8460, Question 06.02.01.03-7 to request the applicant to add an appropriate description in the DCD and justify the approach used. In its response to RAI 394-8460, Question 06.02.01.03-7 (ML16142A010), the applicant stated that the heat transfer across the SG tubes is modeled with the same heat transfer coefficient in both the forward and reverse directions. The staff agrees that using the same nucleate boiling heat transfer coefficient on both the primary and secondary sides of the SG tubes would be conservative since it would maximize the reverse heat transfer from the secondary coolant to primary coolant during the LOCA blowdown. The staff agrees that due to the closure of the turbine stop valves following the LOCA, the secondary side heat transfer would be through natural convection that would lead to a smaller overall heat transfer than realized by the present methodology. The response also submitted the associated markup to revise DCD Tier 2, Section 6.2.1.3.3, accordingly. The staff found the applicant’s description acceptable and, therefore, RAI 394-8460, Question 06.02.01.03-7 is resolved and the DCD revision is tracked under **Confirmatory Item 06.02.01.03-7.**

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(iii) specifies that the calculations of liquid entrainment, i.e., carryout rate fraction (CRF), which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the PWR full length emergency cooling heat transfer experiments. The DCD and TR (Reference 1) do not document the technical basis of the CRF values used in the M&E calculations. On February 3, 2016, the staff issued RAI 394-8460, Question 06.02.01.03-8 to request the applicant to justify its CRF selection. In its response to RAI 394-8460, Question 06.02.01.03-8 (ML16153A471), the applicant clarified that the CRF values used in the reflood M&E analysis are provided in Section 3.6 of the TR (Reference 1). The staff determined that the CRF values used by the applicant were consistent with the acceptable approach defined in Acceptance Criterion No. 1C(iii) of the NUREG–0800, Section 6.2.1.3. The response also submitted the associated markup to revise the TR (Reference 1) to include a statement referencing the guideline set forth in NUREG–0800 under Section 6.2.1.3 as a technical basis for the CRF values used in the report. The staff found the applicant’s response to be acceptable and, therefore, RAI 394-8460, Question 06.02.01.03-8 is resolved and the TR revision is tracked under **Confirmatory Item 06.02.01.03-8.**

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(iii) asks for a justification for the assumption of steam quenching by comparison with applicable experimental data. No information was provided either to ascertain if the liquid entrainment calculations considered the effect on the CRF of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water. The DCD or TR (Reference 1) do not mention whether or not steam quenching was assumed. Criterion No. 1C(iii) also suggests to assume the steam leaving the SGs to be superheated to the temperature of the secondary coolant. DCD does mention the discharge fluid to be superheated steam but does not mention what assumption was made about its temperature. On February 3, 2016, the staff issued RAI 394-8460, Question 06.02.01.03-9 to request the applicant to include these descriptions into the DCD. In its response (ML16153A471), dated June 1, 2016, the applicant clarified that the liquid entrainment, i.e., carryout rate fraction (CRF), used in the reflood analysis is not based on the calculated results from the computer code, FLOOD3, but given as the input based on the acceptable approach defined in Acceptance Criterion No. 1C(iii) of the NUREG–0800, Section 6.2.1.3. Thus, the CRF is not affected by the increase of the core inlet water temperature. The response also clarified that during reflood, credit is taken for the condensation of steam in the annulus by the cold SI water. For conservatism, no credit is taken when the reactor vessel annulus is not full or when the SI flow rate is too low to thermodynamically condense all of the steam in the annulus, i.e., after the SITs empty or the turndown to low SIT flow by the FD. The staff agrees that the treatment of steam quenching is conservatively assumed to maximize the steam release to the containment. The response also clarified that the steam leaving the SGs during the initial time of the reflood phase is predicted to be superheated to the temperature as high as that of the SG secondary coolant temperature by the FLOOD3 code. The staff determined that the assumptions of the initial condition of the SGs secondary side and the SG secondary-to-primary heat transfer described in DCD Section 6.2.1.3.4 and TR (Reference 1) Section 3.6 are conservative for the prediction of the temperature of the superheated steam. The response also submitted the associated markups to revise DCD Tier 2, Section 6.2.1.3 and Table 6.2.1-20 and TR Section 3.2 and Table 4-3, “Initial Conditions for Containment Peak Pressure Analysis,” accordingly. RAI 394-8460, Question 06.02.01.03-9 is resolved and the DCD and TR revisions are tracked under **Confirmatory Item 06.02.01.03-9.**

SRP Section 6.2.1.3 Acceptance Criterion No. 1C(iv) asks for a description of the long-term cooling (or post-reflood) model. However, the DCD or TR (Reference 1) do not provide any discussion or justification of the methods used to calculate the core inlet and exit flow rates and removal of the sensible heat from primary system metal surfaces and the SGs. Liquid entrainment correlations for fluid leaving the core and entering the SGs are neither described nor justified by comparison with experimental data. No statements are made about steam quenching by ECCS water or the applicable experimental data, or whether and how all the remaining stored energy in the primary and secondary systems would be removed during the post-reflood phase. No references are made to compare the results of post-reflood analytical models with the applicable experimental data. On February 3, 2016, the staff issued RAI 394-8460, Question 06.02.01.03-10 to request the applicant add these descriptions in the DCD, or appropriately reference them. In its response to RAI 394-8460, Question 06.02.01.03-10, dated June 8, 2016 (ML16160A034), the applicant stated the post-reflood transient is treated as a continuation of the reflood transient, all modeling assumptions and method are identical, except that the carryout rate fraction (CRF) for liquid entrainment calculations is changed from 0.8 for reflood to 1.0 for post-reflood. The staff agrees that the CRF value of 1.0 is conservative as it would increase the system flow rate and maximize the break flow during the post-reflood period. The response also stated that the assumption of steam quenching and the description of the removal of the remaining stored energy in the primary and secondary systems are identical to that of the reflood period. The staff agrees that not taking credit for condensation after the turndown to low SIT flow during post-reflood period was conservative. The response also provided details of the lumped-parameter GOTHIC model for the RCS used to calculate the M&E release through the break during the decay heat phase. The RCS model has three lumped-parameter volumes that represent a RCS core, a downcomer and a hot/cold leg piping segment. The response also submitted the associated markups to appropriately revise DCD Tier 2, Section 6.2.1.3.5, “Description of Post Reflood Model,” and TR (Reference 1) Section 3.7, “Description of Post-Reflood Model,” accordingly. However, in the July 7, 2016 public teleconference, the staff inquired whether the use of the GOTHIC code for the decay heat phase M&E release analysis can be considered appropriate for this application, as neither the DCD nor the TR discussed it. The applicant submitted a supplemental response to RAI 394-8460, Question 06.02.01.03-10, dated November 18, 2016 (ML16323A494) that characterized the decay heat phase to be a relatively stable period due to the release of core decay heat and the sensible energy of the RCS and SGs metal under thermal equilibrium, to the containment and, thus, does not require specific conservative models to calculate the break flow. The response also cited that the use of GOTHIC for the M&E release calculations for the Dominion’s power plant during the post-reflood decay heat phase in their containment analysis methodology, was previously approved by the NRC in their containment analysis methodology (Reference 7). The staff found the previous NRC approval of the use of the GOTHIC code for the decay heat phase M&E release analysis to be an appropriate justification. The staff finds the applicant’s response acceptable, and RAI 394-8460, Question 06.02.01.03-10, is resolved, and the verification of the proposed DCD and TR revisions is being tracked under **Confirmatory Item 06.02.01.03-10.**

According to SRP Section 6.2.1.3 Acceptance Criterion No. 1C(v), the fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in SRP Section 9.2.5. However, the DCD or TR (Reference 1) provides no information to ascertain how conservative the decay energy model is compared to the one given in SRP Section 9.2.5. SRP Section 6.2.1.3 Acceptance Criterion No. 1C(v) also states that steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water. No such description is found in the DCD or the TR. On February 3, 2016, the staff issued RAI 394-8460, Question 06.02.01.03-11, to request the applicant to clarify these two aspects in the DCD. In its response to RAI 394-8460, Question 06.02.01.03-11 (ML16160A034), dated June 8, 2016, the applicant explained that the decay energy model in the LOCA M&E calculation uses two different decay heat curves both of which are based on the ANSI/ANS 5.1-1994, “Decay Heat Power in Light Water Reactors,” standard decay heat curve. One is ANS 5-1971, “Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors,” decay heat curve, which is chosen for decay energy releases during the earlier phases of a LOCA (blowdown, reflood and post-reflood), and the other is ANSI/ANS 5.1-1979 decay heat curve, which is used for the decay heat phase. The staff determined that, while the ANS 5-1971 is the required decay heat model for Appendix K analysis, the ANS 5.1-1979 model is appropriate for the containment response analysis for the decay heat phase. For the earlier phases of a LOCA until the end of post-reflood, the ANS standard decay curve corrected for decay of the heavy elements U_{239} and Np_{239} has been incorporated into the CEFLASH-4A. The decay heat contribution from actinides other than U_{239} and Np_{239} is additionally taken into account to the decay heat curve for conservatism. The staff agrees that the decay heat model used in the analysis is more conservative compared to that provided in the SRP Section 9.2.5, “Ultimate Heat Sink.”

The response also stated that in the RCS model for the LOCA M&E release during decay heat phase, energy release from coolant and metal of the SGs’ secondary side are modeled using the GOTHIC heater components, which are submerged in the RCS core volume. In the RCS model, the downcomer receives the IRWST water through the SIP, and then feeds it to the core as needed to make up for steaming and returns the remaining water to the IRWST as spillage without temperature increase. This modeling approach basically excludes mixing of the steam with the SI injection water and maximizes steaming in the core. The staff accepts that APR1400 methodology conservatively assumes that there is no mixing of steam and SI water in the RCS during the decay heat phase. The response also submitted the associated markups to revise DCD Tier 2, Section 6.2.1.3.2, “Energy Sources,” and TR Section 3.4, accordingly. The staff accepts the applicant’s response as it has provided the required information. Therefore, RAI 394-8460, Question 06.02.01.03-11 is resolved and the DCD and TR revisions are tracked under **Confirmatory Item 06.02.01.03-11.**

Initial testing for the SIS (specifically, DCD Tier 2, Sections 14.2.12.1.21, “Safety Injection System Test,” and 14.2.12.1.22, “Safety Injection Tank Subsystem Test”) are addressed in Section 6.3 of this SER.

ITAAC: The regulations in 10 CFR 52.47(b)(1) requires, in part, that a DC application contain the ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the as-built plant incorporating the ITAAC will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC’s regulations. ITAAC for the SIS is evaluated in Section 6.3 of this SER. The discharge values for the SITs and SIPs are such that they meet the values assumed in the safety analysis related to this section, and are verified by ITAAC items 9.a.i and 9.a.iii in Table 2.4.3-4. The staff finds the ITAAC acceptable.

The staff determined DCD Tier 1, Table 2.11.1-2, provides ITAAC No. 4 and ITAAC No. 5 to demonstrate two key containment functional design commitments, structural integrity and acting as a barrier against uncontrolled fission products release, required by GDC 16, GDC 38, and GDC 50. ITAAC No.4 ensures that an analysis will be performed that concludes that the as-built containment peak pressure following a high energy line break remains below its design pressure of 4.218 kg/cm^{2}G (60 psig or 515 kPa) with more than 10 percent margin above the maximum calculated peak pressure. ITAAC No.5 ensures that an analysis will be performed that concludes that the as-built containment pressure is reduced to less than 50 percent of its maximum calculated peak pressure for the design basis LOCA within 24 hours after the postulated accident.

TS: TS for initial containment pressure and temperature are specified in APR1400 DCD, Tier 2, Chapter 16, Sections 3.6.4 and 3.6.5 respectively. TS 3.6.4 specifies the LCO, “Containment pressure shall be ≥ -0.007 kg/cm^{2}G (−0.1 psig) and ≤ +0.07 kg/cm^{2}G (+1.0 psig).” Similarly, for temperature the LCO requires that “Containment average air temperature shall be ≤ 49 −C (120 °F).” The safety analyses in DCD Tier 2, Section 6.2.1.3, assume a scenario-dependent SI flow. TS Surveillances 3.5.1.2 and 3.5.2.5 ensure that during normal operation, the SIT volume and SIP flow, respectively, are greater than the minimum values assumed in the safety analysis. As part of the TS for the APR1400, SRs provided for the differential pressure developed by the pump at minimum flow rate (3.5.2.4) and for the design flow rate at design pressure (3.5.2.5). However, no provisions are included to ensure that the SIPs are capable of maintaining the flow rate assumed during the long term recirculation phase of the accident. No SR is provided for the long term SI flow by the SIPs, which is higher than the flow at the differential pressure specified in SR 3.5.2.5, per DCD Tier 1, Table 6.3.2-1, “SIS Component Parameters.” This parameter was included as part of the development of the pump curve in the CE standard TSs, which most closely resemble those for the APR1400. Therefore, on December 11, 2015, the staff issued RAI 331-8419, Question 06.02.01.03-4 to address this issue. In its response to RAI 331-8419, Question 06.02.01.03-4 (ML16007A061), dated January 7, 2016, the applicant stated that the two existing SRs provide sufficient information to detect degradation of the SIP in accordance with the requirements for ASME group B pumps (of which the SIP is one), which states tests shall be conducted at a minimum of one reference point. The staff determined a full pump curve is already required by ITAAC 9.b in DCD Tier 1, Table 2.4.3-4, “Safety Injection System ITAAC,” so there is reasonable assurance the pump will produce its required flow rate and differential pressure over a range of flow conditions. The response is acceptable because the SRs conduct tests at two reference points (for both minimum required flow rate and pump rated condition), which is sufficient to demonstrate pump performance is not degraded, and therefore no change in the TSs is required. Therefore, RAI 331-8419, Question 06.02.01.03-4 is resolved and closed.

6.2.1.3.5   Combined License Information Items

There are no COL information items associated with Section 6.2.1.3 of the APR1400 DCD.

6.2.1.3.6   Conclusion

The regulations and the associated acceptance criteria for this area of review are given in NUREG–0800, Section 6.2.1.3. Review interfaces with other SRP Sections can also be found in NUREG–0800, Section 6.2.1.3. The acceptance criteria are based on meeting the relevant regulatory requirements that include GDC 50; 10 CFR Part 50, Appendix K; and 10 CFR 52.47(b)(1). The technical report (TR), “LOCA Mass and Energy Release Methodology” (Reference 1), describes the CE methodology used by the applicant for the APR1400 design M&E release as well as the analysis of the containment pressure and temperature response to the spectrum of LOCA pipe breaks in the RCB discussed in Section 6.2.1.1 of this SER. The staff’s review of the compliance of the APR1400 M&E release methodology with the associated GDC and other regulatory requirements led the staff to issue several RAIs to the applicant. All RAI responses were received. However, as discussed in Section 6.2.1.3.4, several of them are being tracked as open or confirmatory items.

The staff’s review of the compliance of the APR1400 LOCA M&E release methodology with GDC 50 and other regulatory requirements led to issuing several RAIs to the applicant. All RAI responses and any subsequent supplemental responses were received and all open items were resolved. However, as discussed in Section 6.2.1.3.4 of this SER, several of the RAI responses are being tracked as confirmatory items. The staff will update Section 6.2.1.3 of this report to reflect the final disposition of the confirmatory items in the DCD application.

GDC 50 requires that the containment structure and its internal compartments be designed to accommodate the calculated pressure and temperature conditions resulting from any LOCA accident including sufficient margin. In order to satisfy this requirement, the LOCA analysis should be based on the assumption of LOOP and the most severe single failure in the emergency power system. In its analysis, the applicant assumed a LOOP, and performed parametric evaluations on single failures in the emergency power system to determine the most limiting LOCA scenarios. The applicant satisfies the requirements established in GDC 50 for using the most severe single failure in its analysis. The calculations of mass and energy released during the postulated LOCAs also follow the requirements of 10 CFR 50 Appendix K in accordance with Section 6.2.1.3 of NUREG-0800. The calculations show that the mass and energy release rates for LOCA were calculated to maximize the post-accident containment pressure and the containment subcompartment response.

6.2.1.3.7   References

1.   KHNP Technical Report (TR) APR1400-Z-A-NR-14007-P/NP, “LOCA Mass and Energy Release Methodology,” Revision 0, KHNP, November 2014 (ML15009A322).

2. Combustion Engineering Inc., Topical ReportCENPD-133P (Proprietary), “CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis,” Combustion Engineering, Inc., August 1974.

3. “Computer Code Description and Verification Report for FLOOD3 (Proprietary),” Nuclear Power Systems, Combustion Engineering, Inc., FLOOD3-MOD1, February 4, 1988.

4. GOTHIC Thermal Hydraulic Analysis Package User Manual, Version 8.0 (QA), NAI 8907-02, Revision 20, Numerical Applications Inc., January 2012.

5. “FLOOD-MOD2 - A Code to Determine the Core Reflood Rate for a PWR Plant with Two Core Vessel Outlet Legs and Four Core Vessel Inlet Legs,” Interim Report, Aerojet Nuclear Company, November 2, 1972.

6. Combustion Engineering Inc., Topical Report CENPD-132P (Proprietary), “Calculative Methods for the C-E Large Break LOCA Evaluation Model,” Volume 1&2, August 1974.

7. Transmittal of Approved Topical Report DOM-NAF-3 NP-A, “Gothic Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment.” (ML063190467).

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures

6.2.1.4.1 Introduction

Secondary system piping ruptures inside a reactor containment may result in significant releases of high-energy steam and water into the containment environment that can generate high pressures and temperatures inside the containment. Containment pressurization following a secondary side pipe rupture depends on the quantity and enthalpy of the fluid that enters the containment atmosphere through the break. The M&E release inside the containment following a main steam line break (MSLB) or main feedwater line break (MFLB) depends upon the configuration of the plant’s main steam and feedwater systems, plant operating conditions, and size of the pipe break. The main objective of the M&E release analysis for postulated secondary pipe ruptures inside the containment, as discussed in this Section, is to provide a conservative M&E release input to the resulting containment pressure and temperature calculations performed for containment functional design considerations, as discussed in Section 6.2.1.1 of this SER.

6.2.1.4.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this evaluation is provided in DCD Tier 1, Section 2.11., “Containment System,” Tier 1, Section 2.7.1.2, “Main Steam System,” and Tier 1, Section 2.7.1.4, “Condensate and Feedwater System.” The main steam system is safety-related up to the main steam isolation valves (MSIVs), and is evaluated in Section 10.3 of this SER. The feedwater system has a safety-related function to isolate on the receipt of a main steam isolation signal and is evaluated in Section 10.4.7 of this SER.

DCD Tier 2: The Tier 2 information associated with this evaluation is provided in DCD Tier 2, Section 6.2.1.4 and is summarized in part as follows:

This section describes the sequence of events and effects of transient phenomena, the evaluation model, the input parameters and accident conditions, and the evaluation results for the postulated secondary system pipe ruptures inside the containment. Following a postulated MSLB or MFLB inside the containment, the bulk of the affected SG inventory is released to the containment. Most of the inventory of the unaffected SG are isolated by the closure of the main steam isolation valve (MSIV) and the main feedwater isolation valve (MFIV). Containment pressurization following a secondary side rupture depends on the quantity of the fluid that enters the containment atmosphere as steam through the break. MSLB flow is either pure steam or two-phase water, while MFLB flow is two-phase water. With a pure steam blowdown, all of the break flow enters the containment atmosphere. With a two-phase blowdown, a portion of the liquid in the break flow flashes upon entering the containment and is also added to the atmosphere, while the liquid falls to the sump and does not contribute to containment pressurization. Therefore, a spectrum of reactor power levels with an assumed SF is analyzed to determine the most severe MSLB case that results in the peak MSLB containment pressure or temperature. Non-class 1E electric power is conservatively assumed to be available because it allows for the continuation of RCP operation, which maximizes the rate of heat transfer to the affected SG that, in turn, maximizes the rate of the M&E release. The feedwater distribution box is below the SG water level and is filled with liquid water. Therefore, MFLB cases result in a two-phase blowdown at a lower enthalpy and do not produce peak containment pressures as severe as MSLBs.

The M&E release analyses for the APR1400 secondary system pipe breaks are performed using the SGN-III computer code (Reference 1). The modeled phenomena are described in Appendix 6B, "Description of the SGNIII Computer Code Used in Developing Main Steam Line Break Mass/Energy Release Data for Containment Analysis," of the Combustion Engineering Standard Safety Analysis Report (CESSAR) (Reference 2). The sources of energy considered in the MSLB analysis include the stored energy in the (1) affected SG's metal including the SG tubing, (2) water in the affected SG, (3) feedwater transferred to the affected SG before the closure of the MFIV, and (4) steam from the unaffected SG before the closure of the MSIV. The energy sources that are considered also include the energy transferred from the primary coolant to the water in the affected SG during blowdown. All cases analyzed consider the flashing of the liquid in the lines from the upstream MFIVs to the affected SG.

The flow area of the main steam lines is 0.418 m^2 (4.493 ft^2). For the MSLB analysis, the postulated rupture is conservatively assumed to occur at the nozzle of one of the SGs. The flow restrictor area is 0.119 m^2 (1.28 ft^2). The largest breaks at which a pure steam blowdown can occur are determined. The break flow rate is obtained from the Moody critical flow model (Reference 3). The contribution to the containment pressure of the feedwater flow is handled by a feedwater flow addition to the affected SG, while no feedwater is added to the unaffected SG for conservatism. There is a total of ten MSLB calculations completed by the applicant that cover a spectrum of power levels, break sizes, and failure conditions. M&E release data for the ten MSLB cases that are listed in DCD Tier 2, Table 6.2.1-1, "Spectrum of Postulated Accidents," are given in Part A of Tables 6.2.1-9, "Main Steam Line Break, 102% Power – Loss of One CSS Train (0.849 m^2 (9.134ft^2) Total Break Area) through 6.2.1-18, "Main Steam Line Break, 0% Power – MSIV Failure (0.381 m^2 (4.1 ft^2) Total Break Area." The data represent the total M&E release to the containment at 102, 75, 50, 20, and 0 percent reactor power levels. The MSLB M&E release data in Tables 6.2.1-9, 6.2.1-11, "Main Steam Line Break, 75% Power – Loss of One CSS Train (0.849 m^2 (9.134 ft^2) Total Break Area)," 6.2.1-13, "Main Steam Line Break, 50% Power – Loss of One CSS Train (0.849 m^2 (9.134 ft^2) Total Break Area)," 6.2.1-15, "Main Steam Line Break, 20% Power – Loss of One CSS Train (0.849 m^2 (9.134 ft^2) Total Break Area),"and 6.2.1-17, "Main Steam Line Break, 0% Power – Loss of One CSS Train (0.849 m^2 (9.134 ft^2) Total Break Area)," are based on a loss of one CSS train. The MSLB M&E release data in Tables 6.2.1-10, "Main Steam Line Break, 102% Power – MSIV Failure (0.849 m^2 (9.134

ft^2) Total Break Area)," 6.2.1-12, "Main Steam Line Break, 75% Power – MSIV Failure (0.849 m^2 (9.134 ft^2) Total Break Area)," 6.2.1-14, "Main Steam Line Break, 50% Power – MSIV Failure (0.849 m^2 (9.134 ft^2) Total Break Area)," 6.2.1-16, "Main Steam Line Break, 20% Power – MSIV Failure (0.849 m^2 (9.134 ft^2) Total Break Area)," and 6.2.1-18, "Main Steam Line Break, 0% Power – MSIV Failure (0.849 m^2 (9.134 ft^2) Total Break Area)," are based on one MSIV failure to close. A total break area of 0.849 m^2 (9.134 ft^2) is used for Tables 6.2.1-9 through 6.2.1-14 to cover 102, 75, and 50 percent power. A total break area of 0.818 m^2 (8.8 ft^2) is used for Tables 6.2.1-15 and 6.2.1-16 to cover 20 percent power. A total break area of 0.381 m^2 (4.1 ft^2) is used for Tables 6.2.1-17 and 6.2.1-18 to cover 0 percent power.

ITAAC: ITAAC associated with this evaluation of APR1400 DCD Tier 2, Section 6.2.1.4 are provided in DCD Tier 1, Table 2.11.1-2, Table 2.7.1.2-4, "Main Steam System ITAAC," and Table 2.7.1.4-4, "Condensate and Feedwater System ITAAC."

TS: The TS associated with this evaluation of DCD Tier 2, Section 6.2.1.4 are provided in DCD Tier 2, Chapter 16, Section 3.6 and Section 3.7.2, "Main Steam Isolation Valves (MSIVs)" and 3.7.3, "Main Feedwater Isolation Valves (MFIVs)."

6.2.1.4.3 Regulatory Basis

The Commission regulations and the associated acceptance criteria for this area of review are given in NUREG–0800, Section 6.2.1.4. Review interfaces with other SRP Sections can also be found in NUREG–0800, Section 6.2.1.4. The acceptance criteria are based on meeting the relevant regulatory requirements as summarized below.

- GDC 50 as it relates to providing sufficient conservatism in the M&E release analysis for postulated PWR secondary system pipe ruptures to ensure the reactor containment structure and its associated heat removal system shall be designed so that the containment structure and its internal compartments can withstand, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

- The regulations in 10 CFR 52.47(b)(1) requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations.

6.2.1.4.4 Technical Evaluation

The APR1400 M&E analysis is based on the CE methodology, which has been utilized by the CE plants since their inception. The KHNP Proprietary Technical Report (TR), "LOCA Mass and Energy Release Methodology" (Reference 4), provides the details of the methodology used by the applicant for the APR1400 design for both LOCA and secondary system pipe ruptures inside the containment. The report also describes the methodology to analyze the containment pressure and temperature response to the spectrum of secondary system pipe breaks in the RCB, as discussed in Section 6.2.1.1 of this SER, to determine the maximum containment pressure and temperature following the postulated DBA. The APR1400 methodology also conforms to the regulatory guidelines for conservatism per SRP Section 6.2.1.4 and takes only partial credit for the RCS and containment active safety features. For instance, regardless of the assumed SF, no credit is taken for any containment fan coolers.

GDC 50 of Appendix A to 10 CFR Part 50 requires, in part, that the reactor containment structure and associated heat removal system shall be designed with sufficient margin to accommodate the calculated pressure and temperature conditions resulting from the DBA. NUREG–0800, SRP Section 6.2.1.1.A specifies that the containment design pressure should provide at least a 10 percent margin above the accepted peak calculated containment pressure following a LOCA, an MSLB, or an MFLB, to satisfy the GDC 16 and GDC 50 requirements for sufficient design margin. In addition, ANSI/ANS 56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments," which has established detailed guidelines for containment response to DBAs, specifies that initial conditions shall be chosen to yield a conservatively high peak containment atmosphere pressure and temperature. These guidance documents ensure sufficient conservatism in the M&E release analysis for the postulated primary and secondary system pipe ruptures during the DBA such that the reactor containment structure and heat removal system design can accommodate the calculated peak pressure and temperature conditions. On December 2, 2015 (ML15336A905), the staff issued RAI 324-8362, to seek additional information to gain safety insights into the initial and boundary conditions that the applicant used for the MSLB analyses for the containment, as discussed below.

Following the SRP acceptance criteria, the limiting SF MSLB analysis is based on two assumptions: (1) maximizing the flow of saturated and superheated steam out of the break; and (2) minimizing the rate of heat removal from the containment atmosphere. Since the APR1400 containment response analysis does not credit any fan coolers, the latter assumption is accomplished by the single-failure due to the loss of one train of the CSS. The former assumption can be based on any one of the several possible SFs including the failure of condensate booster pump to trip, feedwater regulating valve (FRV) to close, or MSIV to shut. The APR1400 MSLB analysis has considered the CSS and MSIV SFs to see which one is more conservative. The applicant was requested to clarify whether the SF of the feedwater regulating valve to close was also examined. The staff is concerned that during the time the feedwater bypass valve takes to shut the flow, a considerable amount of feedwater may enter the SG and gain heat from the hot primary-side. Any resulting additional steam would enter the containment to further increase the containment peak pressure and, especially, the peak temperature. On December 2, 2015, the staff issued RAI 324-8362, Question 06.02.01.04-1 to ask the applicant to demonstrate that the current limiting MSLB analysis is bounding for all possible SFs. In its response to RAI 324-8362, Question 06.02.01.04-1, dated February 17, 2016 (ML16048A410), the applicant stated that the current limiting MSLB analysis is bounding for all possible SFs, that the maximum feedwater enthalpy is assumed, and submitted mark-ups to revise DCD Tier 2, Section 6.2.1.4.4, "Description of Blowdown Model," and Table 6.2.1-20 for added clarity regarding the initial conditions of the limiting MSLB. However, during the July 7, 2016, public telephone conference, the staff noted that neither the DCD nor the RAI response presents the SF analysis of other components and requested the applicant justify that the current limiting MSLB analysis is bounding for all possible single failures, even though other possible SFs were apparently not considered for the MSLB analysis. The applicant submitted a supplemental response to RAI 324-8362, Question 06.02.01.04-1, dated November 3, 2016 (ML16308A338) that provided a table with qualitative evaluation of several other single failures of various safety and electrical system components that were considered in the APR1400 MSLB analysis. The staff reviewed the applicant's reasoning for why the single failures are not applicable to the APR1400 MSLB mass and energy release analysis. The staff accepted the exclusion of the control system single failures from the table, as they are typically actuated to mitigate the MSLB accident. The staff determined that the submitted additional information established the DCD's conclusion that the MSIV failure to close is the limiting single failure, as the other credible single failure events would not have conservative impact on the peak containment pressure and temperature and, thus, were not applicable to the APR1400 design. The applicant also submitted the mark-ups to appropriately revise the DCD Tier 2, Section 6.2.1.4.4, "Description of Blowdown Model," Table 6.2.1-20, and the newly added TR Table 3-3 (Reference 4) as a part of the RAI response. The staff determined that the proposed revisions are acceptable because they provide the details and clarity needed to conclude that the current limiting MSLB analysis with the MSIV failure is bounding for all possible SFs. The staff finds the applicant's response acceptable, and RAI 324-8362, Question 06.02.01.04-1 is resolved, and the verification of the proposed DCD and TR revisions is being tracked under **Confirmatory Item 06.02.01.04-1.**

DCD Tier 2, Section 6.2.1.4, assumes no liquid entrainment from the secondary pipe breaks and has analyzed a spectrum of the MSLBs beginning with the double-ended break and decreasing in area until it has been demonstrated that the maximum release rate has been considered. The staff finds this an acceptable approach in accordance with SRP Section 6.2.1.4 Acceptance Criterion No. 2D guidance.

Double-ended pipe breaks could be of the guillotine type or the slot type breaks. Typically, double-ended slot break (DESB) is most severe for LOCA analysis and double-ended guillotine break (DEGB) is most severe for MSLB analysis. On December 2, 2015, the staff issued RAI 324-8362, Question 06.02.01.04-2 and requested that the applicant clarify the type of break (guillotine versus slot) used for the APR1400 MSLB M&E release analysis for the containment and to explain how it was concluded to be the most conservative secondary pipe rupture. In its response to RAI 324-8362, Question 06.02.01.04-2, dated February 17, 2016 (ML16048A410), the applicant responded by providing a comparison of the containment peak pressures for slot and guillotine type breaks at different power levels (102 percent, 75 percent, 50 percent, 20 percent, 0 percent) with CSS failure. Based on the APR1400 sensitivity analysis provided, the staff determined that even though the pressure differences between guillotine and slot break are small and cross over when the power level is increased from 0 percent to 102 percent, the DEGB is more conservative than the DESB for MSLB at the 75 percent power level associated with the peak calculated pressure. As documented in Section 6.2.1.3 of this SER, the applicant has demonstrated that a DEDLSB LOCA is the most conservative DBA for the containment functional design. Therefore, the slot type break is chosen to be used in both the MSLB and LOCA analyses. The peak pressure of the MSLB for DEGB (61.72 psia) is still considerably below the worst peak pressure (65.79 psia) resulting from the limiting DEDLSB LOCA. RAI 324-8362, Question 06.02.01.04-2 is resolved and closed.

Active safety systems are always associated with time delay for actuation. Longer time delays reduce the effectiveness of such systems and are thus more conservative. For the CSS, the time delay consists of the time for the EDGs to start, the spray pump to reach the nominal speed, the spray regulating valve to reach the full stroke, and the drained spray piping and spray header to fill. Section A.3.1, "Assumptions," of the TR (reference 4) specifies that due to the availability of offsite power "unlike the LOCA analysis that assumes a 20-second delay on CS initiation for EDG startup, the MSLB analysis assumes no CS initiation time delay." In order to ensure that the CS actuation delay used in MSLB analysis is conservative, the staff issued RAI 324-8362, Question 06.02.01.04-3, on December 2, 2015, to ask the applicant to clarify whether the delay for CS flow entering the vapor space during an MSLB is 110 – 20 = 90 seconds as the above statement may be interpreted as no CS time delay. The 110 seconds is obtained from Section A.2.3.4, "Components," that states "[t]he CS pump starts automatically from a high-high containment pressure setpoint with a time delay of 110 seconds based on EDG startup (20 seconds) and CS piping fill-up (90 seconds). The delay time for CS piping fill-up includes EDG pump loading (20 seconds), pump startup (3 seconds), system pipe filling (58 seconds), signal delay (2 seconds), and 7 seconds for contingency." In its response to RAI 324-

8362, Question 06.02.01.04-3, dated December 30, 2015 (ML15364A578), the applicant stated that the assumption for the CS delay time for MSLB described in Section A.3.1 of the TR will be revised to clarify delay times required for operation of each component of the CS system and correct several errors. The applicant has presented the revision to Section A.3.1 of Appendix A, “Methodology Description for Containment Response Analysis,” to TR (Reference 4) that clarifies that CS will be initiated after a time delay of 90 seconds. The staff finds the CS initiation time delay of 90 seconds is consistent with the other operations leading up to CS initiation. Therefore, the response is acceptable, and RAI 324-8362, Question 06.02.01.04-3 is resolved and the TR revision is tracked under **Confirmatory Item 06.02.01.04-3.**

SRP Section 6.2.1.4, Acceptance Criterion No. 1 provides a listing of the potential sources of energy that should be considered in the analyses of steam and feedwater line break accidents. DCD Tier 2, Section 6.2.1.4, identifies the sources of energy considered in the MSLB analysis that include the stored energy in the (1) affected SG’s metal, including the SG tubing, (2) water contained in the affected SG, (3) feedwater transferred to the affected SG before the closure of the MFIVs, and (4) steam from the unaffected SGs before the closure of the MSIVs. The energy sources considered also include the energy transferred from the primary coolant to the water in the affected SG during blowdown. The staff determined these sources of energy to be in accordance with SRP Section 6.2.1.4. SRP Section 6.2.1.4 Acceptance Criterion No. 1 specifies that the MSLB accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102 percent of full power. DCD Tier 2, Section 6.2.1.4 reports that MSLB accidents are analyzed to determine peak temperature and pressure using SGN-III computer code at 102, 75, 50, 20, and 0 percent reactor power levels. The breaks are conservatively assumed to be at the nozzle of one of the SGs. DCD Tier 2, Section 6.2.1.1.3.3, “Analysis of Containment Response to Secondary System Piping Ruptures,” reports that the limiting MSLB for peak temperature corresponds to a double-ended rupture of the main steam line (break area 0.849 m^2 (9.134 ft^2)) at 102 percent power level concurrent with an MSIV SF. The staff finds the MSLB break spectrum analysis meets SRP Section 6.2.1.4 Acceptance Criterion No. 1 because it duly accounts for the specified potential sources of energy and analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102 percent of full power.

Per GDC 50 requirements to analyze the most severe consequences for the spectrum of postulated secondary pipe break sizes, locations, and SFs, SRP Section 6.2.1.4, lays out several acceptance criteria to ensure that the containment M&E release calculations are performed for the worst DBA. In this regard, the staff reviewed the compliance of the available details of the APR1400 M&E methodology for the secondary system pipe rupture and issued RAI 8465 to seek information on the conservative treatment of M&E release calculations for the limiting MSLB/MFLB analyses from the containment response standpoint, such that the post-accident containment pressure and temperature are maximized. The applicant was also requested to update the DCD or the TR (Reference 4) to document the explanations.

SRP Section 6.2.1.4 suggests review of single-failure analysis of both main steamline breaks (MSLBs) and main feedwater line breaks (MFLBs). DCD Tier 2, Table 6.2.1-1 “Spectrum of Postulated Accidents,” describes five LOCA and ten MSLB cases analyzed to identify the most severe DBA to meet the requirements of GDC 16, GDC 38, and GDC 50. However, no information is provided either in the DCD Tier 2, Chapter 6 or in the TR (Reference 4) about the main feedwater line break (MFLB) analysis or results. Table 6.2.1-37, “Stored Energy Sources,” reports data only for LOCAs and MSLBs, but does not include any MFLB data. On February 1, 2016, the staff issued RAI 385-8465, Question 06.02.01.04-4, to request the applicant to describe their MFLB analyses in the DCD or TR or justify why they were not considered in the break spectrum analysis and that it did not affect the conservatism in the limiting secondary pipe rupture DBA. In its response to RAI 385-8465, Question 06.02.01.04-4, dated April 25, 2016 (ML16116A455), the applicant made a qualitative comparison between the MSLB and MFLB cases and stated that containment pressurization following a secondary pipe rupture depends almost entirely on the amount of break fluid entering the containment atmosphere as steam with high enthalpy. Even when the MSLB flow is not pure steam, part of the liquid in the break flow flashes in the containment and the steam is added to the atmosphere while the remaining liquid falls to the sump without contributing significantly to containment pressurization. DCD Tier 2, Section 6.2.1.4 also states the feedwater distribution box is below the SG water level. Therefore, the staff agrees that the MFLB blowdown would be predominately liquid with relatively low enthalpy, as compared to the MSLB and LOCA cases (which generally have higher steam fractions), that would produce less severe peak containment pressure and temperature than MSLBs. The staff accepts the applicant's rationale for not analyzing MFLBs, and that the conservative M&E release calculations for the containment peak pressure and temperature are for the LOCA (reported in DCD Tier 2, Section 6.2.1.3) and the MSLB (reported in DCD Tier 2, Section 6.2.1.4). Therefore, RAI 385-8465, Question 06.02.01.04-4 is resolved and closed.

SRP Section 6.2.1.4 Acceptance Criterion No. 2A states that mass release rates for the secondary-system pipe rupture should be calculated using the Moody model for saturated conditions or a model that is demonstrated to be equally conservative(ML15118A293). DCD Tier 2, Section 6.2.1.4.4, "Description of Blowdown Model," briefly mentions that the break flow rate for secondary pipe ruptures is calculated using the Moody critical flow model for zero flow resistance. No further information is provided either in the DCD or in the TR (Reference 4) about the application of the Moody model. On February 1, 2016, the staff issued RAI 385-8465, Question 06.02.01.04-5, to request the applicant to justify how Moody's critical flow model was used conservatively for secondary system pipe break analysis and provide more details on the assumptions, regarding the fluid phase, and about the empiricism used in the model (e.g., Moody flowrate multiplier). In its response to RAI 385-8465, Question 06.02.01.04-5, dated April 25, 2016 (ML16116A455), the applicant stated that the MSLB analysis assumed the postulated rupture at the SG nozzle, with a zero flow friction coefficient from the affected SG to the break. The response also provided a tabulation of Moody's critical flow rates for zero flow resistance as a function of pressure and quality, as used by the SGN-III code to calculate the MSLB break flows. The response did not report any additional empiricism used in the model, such as, a flowrate multiplier, etc. The staff accepts the Moody critical flow model with zero flow resistance to be conservative as it has been shown to maximize the blowdown to the containment. Therefore, RAI 385-8465, Question 06.02.01.04-5 is resolved and closed.

SRP Section 6.2.1.4 Acceptance Criterion No. 2B states that the calculations of heat transfer to the water in the affected SG should be based on nucleate boiling heat transfer. DCD Tier 2, Section 6.2.1.3 (Mass and Energy Release Analyses for Postulated Loss-of Coolant Accidents) does mention that a nucleate boiling heat transfer coefficient is used to model the heat transfer from the SG tubes to the primary coolant, for the M&E release analysis for postulated LOCAs. However, no such information is provided in DCD Tier 2, Section 6.2.1.4 for the secondary system pipe rupture or in the TR, so it is not clear whether the statement made in DCD Tier 2, Section 6.2.1.3 applied to LOCA only or would also apply to MSLB. On February 1, 2016, the staff issued RAI 385-8465, Question 06.02.01.04-6, to request the applicant provide information about the heat transfer correlation used and justify it to be conservative. In its response to RAI 385-8465, Question 06.02.01.04-6, dated April 25, 2016 (No. ML16116A455), the applicant clarified that the M&E release in the MSLB analysis is calculated by considering the nucleate boiling heat transfer coefficient, as in the LOCA analysis. The SGN-III computer code used for the MSLB analysis models the secondary heat transfer coefficient as a boiling heat transfer co-efficient. The response also specified that both boiling and fouling resistances were assumed to be zero in the SG overall heat transfer calculation. The staff finds this approach amounts to using an infinite boiling heat transfer coefficient, which is the inverse of heat transfer resistance, to be conservative, as it maximizes the heat transfer in the SG. The response also submitted the associated markup to revise DCD Tier 2, Section 6.2.1.4.4, accordingly. The staff accepts the applicant's response as it has provided the required information. Therefore, RAI 385-8465, Question 06.02.01.04-6 is resolved and the DCD revision is tracked under **Confirmatory Item 06.02.01.04-6.**

The SGN-III computer code is used for the APR1400 secondary system pipe break M&E release analysis. The staff performed a review the SGN-III computer code (References 1 and 2) as used in the APR1400 secondary system pipe break M&E release methodology for the MSLB. The review showed that the APR1400 DCD and M&E Release TR do not comment on the acceptability of the SGN-III code for the APR1400 application, which needs to be established. On February 1, 2016, the staff issued RAI 385-8465, Question 06.02.01.04-7, to request the applicant to document whether the SGN-III computer code has been validated against pertinent experimental data. In its response to RAI 385-8465, Question 06.02.01.04-7, dated June 8, 2016 (ML16160A049), the applicant clarified that the SGN-III computer code was validated against pertinent experimental data, as documented in SYS80-CESSAR (Combustion Engineering Standard Analysis Report), Appendix 6B (Reference 2). The response also provided the validation details of the SGN-III code predictions against the test data from four different test facilities (Kreisinger Development Laboratory (KDL) tests, Battelle tests, General Electric tests and Vallecitos tests). The staff found the use of a steam separation rate multiplier of 2.5 in the methodology to be conservative in predicting the extent of the two-phase swell reaching the SG nozzles following the break for all four test datasets. The staff determined that the response addresses the key question whether or not the swell is sufficient for the two-phase level to reach and stay at the SG nozzles throughout the transient due to depressurization. The response made a commitment to update the TR (Reference 4) to cover the M&E release methodology for both LOCA and MSLB. However, during the July 7, 2016, public teleconference the staff noted that the RAI response did not comment on the acceptability of the SGN-III computer code for the APR1400 for secondary system pipe break analysis. The applicant submitted a supplemental response to RAI 385-8465, Question 06.02.01.04-7, dated November 3, 2016 (ML16308A392) to demonstrate the acceptability of the SGN-III computer code for the APR1400 design. The response stated that the use of SGN-III code was approved by NRC for analyzing the MSLB accident for the CE-type nuclear power plants, and the acceptability of the SGN-III code for the APR1400 application is explained and documented in the SYS80-CESSAR (Reference 2). The response made a commitment to revise the title of the TR (Reference 4) to include both LOCA and MSLB mass and energy release. The RAI response also submitted mark-ups to appropriately update the TR to explain the acceptability and validation of the SGN-III code against test results. The staff determined that the proposed revisions are acceptable because they provide the related details and clarity. The staff finds the applicant's response acceptable, and RAI 385-8465, Question 06.02.01.04-7, and the verification of the proposed TR revisions is being tracked under **Confirmatory Item 06.02.01.04-7.**

There is one MSIV installed in each main steam line and two MFIVs in each feedwater line. The closure of the MSIVs and the MFIVs by the engineered safety feature actuation system (ESFAS) is considered in the M&E analysis. Following closure of the MFIVs, there is an inventory of feedwater between the MFIVs and the affected SG. In addition to the energy sources identified and met in SRP Section 6.2.1.4 Acceptance Criterion No. 1, SRP Section

6.2.1.4 Acceptance Criterion No. 2C, states that an applicant should also account for the water contained in the affected SG's feedwater line, and steam in the affected SG, for conservatism. As the affected SG depressurizes, this inventory starts to boil. As steam in the line expands, the feedwater inventory is pushed into the SG and is boiled off by primary-to-secondary heat transfer. As described in DCD Tier 2, Section 6.2.1.4, the total volume of fluid between the upstream MFIV and each SG is assumed to be the maximum. The staff concludes that the applicant has accounted for the water contained in the affected SG's feedwater line. However, the DCD does not mention whether the energy stored in the steam in the affected SG is also accounted for in the M&E release calculations for the secondary system pipe ruptures. On February 1, 2016, the staff issued RAI 385-8465, Question 06.02.01.04-8 to request the applicant to clarify this in the DCD. In its response to RAI 385-8465, Question 06.02.01.04-8 (ML16126A527), dated May 5, 2016, the applicant clarified that the initial energy stored in the steam and liquid in the affected SG is taken into account for the M&E release calculations for the MSLB analysis. The response also provided the SGN-III code formulations of the initial steam and liquid masses for the affected SG as a function of various SG component volumes. The staff determined that the applicant's use of maximum SG component volumes to maximize the initial steam and liquid masses for the affected SG was conservative. The response made a commitment to revise DCD Tier 2, Section 6.2.1.4, as indicated in the mark-up associated with this response. The staff finds the applicant's response acceptable, and RAI 385-8465, Question 06.02.01.04-8 is resolved. The verification of the proposed DCD Section 6.2.1.4 revision is being tracked under **Confirmatory Item 06.02.01.04-8.**

SRP Section 6.2.1.4, Acceptance Criterion No. 2E, specifies that feedwater flow to the affected SG should be calculated considering the diversion of flow from the other SGs, feedwater flashing, and increased feedwater pump flow caused by the reduction in SG pressure. The MSLB M&E data given in DCD Tier 2, Tables 6.2.1-9 through 6.2.1-18 represent the total release to the containment including the contributions from the steam lines and feedwater lines. In addition, up to 30 minutes before auxiliary feedwater isolation to the affected SG following the MSLB, a constant 3,028 L/min (800 gpm) auxiliary flow to the affected SG is assumed and included in Tables 6.2.1-9, "Main Steam Line Break, 102% Power – Loss of One CSS Train (0.849 m^2 (9.134 ft^2) Total Break Area)," through 6.2.1-18. As described in DCD Tier 2, Section 6.2.1.4, the flashing of feedwater fluid into the affected SG and then into the containment is also considered in the analysis. The staff finds the isentropic expansion of the feedwater inventory into the affected SG assumed for the analysis to be conservative to M&E release due to maximum expansion and subsequent flashing of feedwater into two phase. The staff finds that the analysis accounts for the addition of unisolated feedwater mass to the affected SG after isolation and it further conservatively maximized the feedwater flow by assuming that all feedwater travels to the affected SG at the pump runout rate before isolation. Therefore, SRP Section 6.2.1.4, Acceptance Criterion No. 2E, has been met. The MSLB M&E release continues until the operator manually terminates auxiliary feedwater flow to the affected SG. The operator actions to close the manual valves to terminate auxiliary feedwater flow will be reviewed under SRP Section 10.4.9, "Auxiliary Feedwater System (PWR)."

ITAAC: The regulations in 10 CFR 52.47(b)(1) requires, in part, that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the as-built plant incorporating the ITAAC will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations. The ITAAC related to the main steam system and the main feedwater system are evaluated in Section 10.3, "Main Steam System," and Section 10.4.7, "Condensate and Feedwater System," respectively, of this SER. Tests from the initial test program listed in DCD, Tier 2, Section

14.2.12.1, “Preoperational Tests,” specifically, the tests listed in Sections 14.2.12.1.63, “Main Steam Safety Valve Test,” 14.2.12.1.64, “Main Steam Isolation Valves and MSIV Bypass Valves Test,” 14.2.12.1.65, “Main Steam System Test,” and 14.2.12.1.68, “Feedwater System Test,” are also addressed in their respective sections of this SER.

TSs: The staff determined that the M&E releases reported in DCD Tier 2, Section 6.2.1.4, rely on the MSIVs to close in 5 seconds and the MFIVs to close in 10 seconds. It was also found that these time durations, as inputs to safety-related analyses, are confirmed periodically by TS surveillances 3.7.2.1 and 3.7.3.1, respectively, in accordance with the assumptions in the safety analyses.

6.2.1.4.5 Combined License Information Items

There are no COL information items associated with Section 6.2.1.4 of the APR1400 DCD.

6.2.1.4.6 Conclusion

The NRC regulations and the associated acceptance criteria for this area of review are given in NUREG–0800, Section 6.2.1.4. Review interfaces with other SRP Sections can also be found in NUREG–0800, Section 6.2.1.4. The acceptance criteria are based on meeting the relevant regulatory requirements that include GDC 50 and 10 CFR 52.47(b)(1).

The KHNP Proprietary Technical Report (TR), “LOCA Mass and Energy Release Methodology” (Reference 4), describes the CE methodology used by the applicant for the APR1400 design M&E release as well as the analysis of the containment pressure and temperature response to the spectrum of high-energy secondary pipe ruptures in the RCB discussed in Section 6.2.1.1 of this SER. The staff reviewed the APR1400 M&E release methodology for the design basis secondary pipe rupture per the NRC regulations and associated acceptance criteria provided in NUREG–0800, Section 6.2.1.4.

The staff’s review of the compliance of the APR1400 MSLB/MFLB M&E release methodology with GDC 50 and other regulatory requirements led to issuing several RAIs to the applicant. All RAI responses and any subsequent supplemental responses were received and all open items were resolved. However, as discussed in Section 6.2.1.4.4 of this SER, several of the RAIs are being tracked as confirmatory items. The staff will update Section 6.2.1.4 of this SER to reflect the final disposition of the confirmatory items in the DCD application.

The relevant requirements of the Commission’s regulations for reviewing mass and energy release analysis for postulated secondary system pipe ruptures are GDC 50, 10 CFR 52.47(b)(1), and 10 CFR 52.80(a), as discussed in SRP Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures.”

The staff verified that the applicant followed the SRP Section 6.2.1.4 acceptance criteria and provided sufficient information about the mass and energy release analysis for the postulated APR1400 secondary system pipe ruptures. This also involved reviewing the sources of energy used in the analyses of steam and feedwater line break accidents including the energy stored in metal and fluid inventories. The single-failure accident analyses were performed with conservative assumptions to maximize the post-accident containment pressure and temperature. The review ensured that the reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed with sufficient margin to accommodate the pressure and temperature conditions resulting from any steam and feedwater line break, without exceeding the design leakage rate.

Based on its review of the docketed information, the staff concludes that the APR1400 mass and energy release analysis for postulated secondary system pipe ruptures is acceptable and meets the requirements of GDC 50 for providing sufficient conservatism in the containment design basis.

6.2.1.4.7 References

1. Description of the SGNPV Digital Computer Code Used in Developing Main Steam Line Break Mass/Energy Release Data for Containment Analysis (non-proprietary), SGNIII-MOD1, Combustion Engineering Inc., February 9, 1988.

2. SYS80-CESSAR (Combustion Engineering Standard Safety Analysis Report), Appendix 6B, "Description of the SGNIII Computer Code Used in Developing Main Steam Line Break Mass/Energy Release Data for Containment Analysis," (non-proprietary), 1974.

3. J. Moody, "Maximum Two-Phase Vessel Blowdown from Pipes," Journal of Heat Transfer, Volume 88, August 1966.

4. KHNP Technical Report (TR) APR1400-Z-A-NR-14007-P/NP, "LOCA Mass and Energy Release Methodology," Revision 0, KHNP, November 2014 (ML15009A322).

5. NUREG -1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," Volume 1, Page 6-9, August 1994 (ML100780158).

6.2.1.5 Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies

6.2.1.5.1 Introduction

Following a LOCA in a PWR plant, the ECCS supplies water to the reactor vessel to reflood and thereby cool the reactor core. The rate of core flooding is restricted to the capability of ECCS water to displace the steam generated in the reactor vessel during the core reflooding period. This core flooding rate increases with increasing containment pressure.

Therefore, as part of the overall evaluation of ECCS performance, it is important to consider the minimum containment pressure possible during the time until the core is reflooded following a LOCA to confirm the validity of the containment pressure in ECCS performance capability studies. The review includes assumptions for the operation of engineered safety feature heat removal systems, the effectiveness of structural heat sinks within the containment to remove energy from the containment atmosphere, and other heat removal processes.

The review is for all PWR containment types (i.e., dry, subatmospheric, and ice condenser containments). It is noteworthy that the minimum containment pressure analysis for ECCS performance evaluation differs from the containment functional performance analysis in that the conservatisms and margins are generally taken in opposite directions; thus, the minimum containment pressure analysis required by the regulations for ECCS performance evaluation is not conservative as to peak containment pressure in a LOCA and cannot be used to determine the containment design basis.

6.2.1.5.2 Summary of Application

DCD Tier 1, Revision 1 contains no information related to this area of review.

The minimum containment pressure analysis used to support performance capability studies of the ECCS system for the APR1400 is presented in DCD Tier 2, Section 6.2.1.5. Calculations for the M&E release associated with the LOCA are performed in accordance with the "Realistic Evaluation Methodology for Large-Break LOCA of the APR1400" report discussed in Section 6.2.1.3 of this SER. The M&E releases used to calculate the minimum pressure are tabulated and discussed in DCD Tier 2, Section 6.2.1.5.2, "Mass and Energy Release Data." Section 6.2.1.5.3, "Initial Containment Internal Conditions," provides the initial containment conditions, which are discussed further below in the technical evaluation Section 6.2.1.5.4.

Other parameters necessary to complete the analysis, such as the nature of containment heat sinks, the assumed containment free volume (3,431,000 ft^3), and the conditions for the spray system, are also provided in DCD Tier 2, Section 6.2.1.5. Four containment spray pumps are assumed to provide a total of 20,000 gpm of flow, cooled by heat exchangers at the minimum temperature of 10 °C (50 °F). In addition, the nonsafety-related containment cooling system is assumed to be in operation and a containment purge is assumed in progress at the initial phase of the transient. The containment pressure is calculated by the applicant using the CONTEMPT4/MOD5 code, and the resultant containment pressure and temperature and IRWST temperature are presented in the DCD Tier 2, in Figures 6.2.1-50, "1.0 x Double-Ended Guillotine Break in Pump Discharge Leg (Min. Containment Pressure for ECCS Performance Analysis)," through 6.2.1-52, "1.0 x Double-Ended Guillotine Break in Pump Discharge Leg (In-containment Refueling Water Storage Tank)." TS associated with the ECCS and the containment are specified in DCD Tier 2, Chapter 16, Sections 3.5, "Emergency Core Cooling Systems (ECCS)," and 3.6, "Containment Systems."

6.2.1.5.3 Regulatory Basis

The relevant requirements for the Commission regulations for the minimum containment pressure analysis for ECCS performance capability studies, and the associated acceptance criteria, are given in Section 6.2.1.5 of NUREG–0800. The application is required to satisfy the following:

- The regulations in 10 CFR 50.46(a)(1)(ii), which requires that an ECCS evaluation model be developed in compliance with 10 CFR 50, Appendix K, paragraph I.D.2. Section I.D.2 of Appendix K to 10 CFR Part 50 requires, in part, that the applicant calculate a containment pressure used for ECCS reflood conditions that shall not exceed a pressure conservatively calculated for that purpose.

- The regulations in 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations.

The related acceptance criteria are as follows:

- RG 1.157, "Best-estimate Calculations of Emergency Core Cooling System Performance," which allows for the use of best-estimate models in calculating cooling effectiveness during the post-blowdown phase of a loss-of-coolant accident, provided their technical basis is demonstrated with appropriate data and analyses.

- BTP 6-2 provides guidance for using conservative estimates for the heat transfer correlations and thermal properties associated with containment internal structures and for calculating the associated heat transfer from the containment atmosphere. These estimates are conservatively high for the purpose of minimizing the containment pressure.

The staff reviewed the analysis conducted by the applicant to determine the minimum containment pressure existing during the period of time until the core is reflooded following a LOCA. The review was conducted in accordance with the criteria of SRP Section 6.2.1.5, BTP 6-2, and Section C.I.6.2.1.5 of RG 1.206.

6.2.1.5.4 Technical Evaluation

DCD Tier 2, Section 6.2.1.5, presents the containment analysis used to determine the minimum backpressure for input as a boundary condition in the ECCS evaluation model. Following a LOCA, the core flooding rate of a PWR is dependent on the ability of the ECCS fluid to displace steam generated in the reactor vessel. The core flooding rate increases with increasing containment pressure and, therefore, minimizing the containment pressure used in the ECCS analysis is conservative. The purpose of the staff's review is to confirm that the pressure used as a boundary condition in the ECCS performance studies, as it relates to DCD Tier 2, Section 15.6.5, "Loss-of-Coolant Accidents Resulting from the Spectrum of Postulate Piping Breaks within the Reactor Coolant Pressure Boundary," complies with the regulatory requirement as specified in 10 CFR 50.46.

The staff reviewed the inputs and assumptions used in calculating the minimum pressure and determined whether they match the guidance in SRP Section 6.2.1.5 and BTP 6-2. For any discrepancies, the staff assessed whether the assumptions used by the applicant are conservative. The staff then used the conditions provided by the applicant to perform an independent confirmatory calculation.

The applicant used the CONTEMPT4/MOD5 code to perform analyses calculating the minimum peak containment pressure and temperature. CONTEMPT4 is a computer code used to perform multi-compartment analyses of reactor containments subjected to a high energy line break.[1] The code has been reviewed and accepted for use by the NRC for analyses of this type for operating licensees. In DCD Tier 2, Figure 6.2.1-50 and -51, the applicant depicted the calculated containment pressure and atmosphere temperature used for the APR1400 ECCS analysis. The conservatively calculated minimum peak containment pressure is approximately 268.7 kPa (39.0 psia), as compared to the peak pressure of approximately 453.7 kPa (65.8 psia) conservatively calculated for containment design and leakage considerations.

In accordance with SRP Section 6.2.1.5 and BTP 6-2, the conditions that used that result in this minimum containment pressure are:

- Minimum allowable initial pressure of 100.4 kPa (14.56 psia)
- Minimum allowable initial containment temperature of 10 °C (50 °F)

[1] "CONTEMPT4/MOD5: An Improvement to COMTEMPT4/MOD4 Multicompartment Containment System Analysis Program for Ice Containment Analysis," NUREG–CR-4001, NRC, Washington, DC, September 1984.

- Initial relative humidity of 90 percent
- Addition of 6.4 percent to the containment volume, when compared to the nominal volume
- Increase of passive heat sink surface areas by a factor of 1.1, with a corresponding reduction in thickness, to remain consistent with the nominal volume
- During the blowdown period, use of the Tagami heat transfer correlation with a multiplier of 4
- Following the blowdown period, use of the Uchida heat transfer correlation with a multiplier of 1.2
- The containment purge system was assumed to be in operation through the 20.3 cm (8 in.) diameter line until the line is isolated following a containment isolation signal, with a maximum purged mass of 600 lbs. of dry air (approximately 0.2 weight percent of the containment air volume)

The validity of these parameters is discussed in detail below.

The initial pressure and temperature correspond to the minimum value allowed by TS. Minimum values for these parameters bias the containment pressure for the transient lower and, therefore, the values are acceptable. The relative humidity chosen is a reasonably conservative initial value based on the initial conditions. Confirmatory analyses performed by the staff indicate that the calculated containment pressure is relatively insensitive to the initial relative humidity inside containment, although BTP 6-2 states the maximum relative humidity for normal operating conditions should be used to conservatively minimize pressure. As such, the use of 90 percent (rather than 100 percent, the absolute maximum) relative humidity is acceptable. No information is provided with respect to the temperatures external to the containment, including the outside ambient air design temperature. Therefore, on June 23, 2015, the staff issued RAI 49-7825, Question 06.02.01.05-4 to address this issue (ML15174A084). In its response to RAI 49-7825, Question 06.02.01.05-4, dated September 24, 2015 (ML15267A720), the applicant clarified that the design range for external temperatures is -40° F to 115° F and these temperatures have no impact on the calculation, as the containment features exposed to the environment (the outside wall and dome) are assumed adiabatic in the maximum pressure analysis (which is conservative) and performed a sensitivity calculation in the response for the minimum pressure analysis showing negligible impact.

DCD Tier 2, Section 6.2.1.5.4, "Containment Volume," stated the containment volume and explained that the volume used to calculate the minimum pressure in Section 6.2.1.5 included a consideration of uncertainty but did not include any specifics on what was entailed by a consideration of uncertainty. Therefore, on June 23, 2015, the staff issued RAI 49-7825, Question 06.02.01.05-1 to address this issue (ML15174A084). In its response to RAI 49-7825, Question 06.02.01.05-1, dated September 24, 2015 (ML15267A720), the applicant clarified the sub-volumes that make up the containment free volume, including uncertainty as margin associated with each sub-volume. Blocked volumes in containment were also enumerated in the response. The applicant recalculated the containment volume and arrived at slightly lower maximum value of 3,431,000 ft^3 (97,155 m^3) that is reflected in DCD, Revision 1.

Sufficient information regarding passive heat sinks was not provided. Therefore, on June 23, 2015, the staff issued RAI 49-7825, Question 06.02.01.05-3 to address this issue (ML15174A084). The response to RAI 49-7825, Question 06.02.01.05-3, was incorporated with the response to issued RAI 49-7825, Question 06.02.01.05-2, which asked for a justification for the thermal properties of the heat sinks as compared with discrepancies with the values suggested in BTP 6-2. In its response to RAI 49-7825, Question 06.02.01.05-2, dated September 24, 2015 (ML15267A720), the applicant provided an updated, more detailed list of containment heat sinks, with minimum, nominal, and maximum values for area and thickness broken down into various categories. These updated values, in concert with other changes made by the applicant, were used to produce a new containment pressure curve for the minimum pressure analysis. As part of a response to RAI 399-8510, **Question 15.06.05-7**, the applicant provided further revisions to these values; this response is evaluated in its entirety below, and is being tracked as a **confirmatory item.**

The use of the Tagami correlation during the blowdown phase and the Uchida correlation with a multiplier of 1.2 following blowdown is in accordance with the stipulations of BTP 6-2 and therefore is acceptable.

Assumption of the purge system operation at the start of the transient is conservative as it contributes to minimizing the air mass inside containment and thereby lowers the pressure. The maximum mass assumed to be purged corresponds to approximately twice the maximum allowable leak rate in a day from the containment. Given the purge line is isolated upon receiving a high containment pressure signal that would be received within a few seconds in the event of a high energy line break, the maximum purged air mass of 272 kg (600 lbm) is acceptable.

The methodology used to obtain the M&E releases used in the minimum containment pressure analysis is described in DCD Tier 2, Section 15.6.5, and is consistent with the guidance in SRP Section 6.2.1.5 and the requirements of Appendix K to Part 50, "ECCS Evaluation Models." The methodology used to conform with BTP 6-2 by the applicant in the DCD appropriately considers the mixing of subcooled ECCS water from the break with the steam atmosphere to further minimize the pressure. The CONTEMPT4/MOD5 analysis also conservatively treats the spilled SI water as an additional heat sink for pressure minimization.

The data for the M&E release in DCD Tier 2, Table 6.2.1.39, "Blowdown and Reflood Mass and Energy Release for the Minimum Containment Pressure Analysis," was inconsistent with the results displayed in DCD Tier 2, Figure 6.2.1-50. Therefore, on June 23, 2015, the staff issued RAI 49-7825, Question 06.02.01.05-3 to address this issue (ML15174A084). In its response to RAI 49-7825, Question 06.02.01.05-3, dated September 24, 2015 (ML15267A720), the applicant provided corrections to the table, in addition to performing a review of DCD Tier 2, Section 6.2.1.5 to correct additional inconsistencies.

In conjunction with the more detailed containment volume analysis and the refinement of the heat sinks in the model, the changes to Table 6.2.1-39 prompted revision of the containment minimum pressure analysis. As part of its response package, the applicant prepared updated figures reflecting the new analysis that was performed. This analysis resulted in slightly higher pressures and temperatures than were calculated in the initial analysis. The applicant updated Figure 6.2.1-50 and Table 6.2.1-39, along with associated changes in the text as part of Revision 1 to the DCD. As part of a response to RAI 8510, Question 15.06.05-7, the applicant provided further revisions to the mass and energy release values in Table 6.2.1-39. During review of the RAI, the applicant determined there were potentially incorrect inputs to fuel parameters that required revision (moderator temperature coefficient and fuel thermal conductivity degradation). Additionally, the applicant found that the data transfer between

RELAP and CONTEMPT4 was sensitive to the frequency of mass and energy data reading. This data reading issue caused CONTEMPT to over-predict the resulting containment pressure. By increasing the data transfer frequency, the applicant found that CONTEMPT calculated a more conservative (lower) containment pressure. As a result, in its response, the applicant calculated a new minimum containment pressure, accounting for the new moderator and fuel parameters, accounting for the revised heat sink values discussed above and using a more frequent time step sampling between CONTEMPT4 and RELAP. The revised discussion in DCD Tier 2, Section 6.2.1.5 and revised mass and energy inputs and resulting calculated figures (DCD Tier 2 Tables 6.2.1-23 and 6.2.1-39, and Figures 6.2.1-49 through 6.2.1-52) are being tracked as a **Confirmatory item**.

The staff performed confirmatory analysis of the minimum containment pressure response following a LOCA using MELCOR. MELCOR is an independently developed, lumped-parameter thermal hydraulics code used by the NRC staff for containment design bases and severe accident analyses. The staff's confirmatory analysis, using values consistent with those in the revised RAI response for DCD Tier 2, Revision 1, Table 6.2.1-39 and the analysis assumptions in DCD Tier 2 Section 6.2.1.5, showed reasonably close agreement. The staff calculation yielded values that were slightly higher than those calculated by the applicant. A lower calculated containment pressure is conservative for this case. Staff notes that this confirmatory calculation was performed on the values reflected in the RAI response to Question 06.02.01.05-3. As part of a response to RAI 8510, Question 15.06.05-7, the applicant provided further revisions to the mass and energy release values in Table 6.2.1-39, and this response is evaluated above. Ultimately, the confirmatory analysis performed by the staff still serves to provide a degree of confidence that the model used by the applicant is appropriately conservative and reflects the design values used in the DCD. As such, based on the confirmatory analysis in conjunction with the staff's finding that the applicant used the appropriate guidance followed as indicated above, the staff finds the applicant's calculation appropriately conservative for use in determining the minimum pressure for ECCS analysis.

6.2.1.5.5 Combined License Information Items

There are no COL information items associated with Section 6.2.1.5 of the APR1400 DCD.

6.2.1.5.6 Conclusion

In conclusion, subject to the closure of the confirmatory items noted above, the staff finds that the applicant has satisfied the Appendix K requirement to 10 CFR Part 50, which requires a conservative backpressure to be calculated and used in ECCS reflood analyses. Specifically, the applicant performed a minimum containment pressure analysis, using assumptions listed above that minimize the calculated containment pressure and which are consistent with those assumptions acceptable to the staff, by following the guidance given in BTP 6-2, SRP Section 6.2.1.5, and Section C.I.6.2.1.5 of RG 1.206. The validity of the methodology for calculating the M&E release has been evaluated and found acceptable in Section 6.2.1.3 of this SER.

On the basis of the aforementioned considerations, the staff finds the minimum containment pressure analysis to be acceptable. The credited minimum pressure has been evaluated in the overall context of the ECCS performance capability studies. Section 15.6.5 of this SER provides the staff's evaluation of the ECCS performance.

6.2.2 Containment Heat Removal Systems

6.2.2.1 Introduction

In order to maintain containment integrity and conform to the requirements of GDC 38, GDC 39, "Inspection of containment heat removal system," and GDC 40, "Testing of containment heat removal system," reactors are equipped with systems to remove heat from the containment. These systems take various forms in different designs, and can include sprays and residual heat removal heat exchangers. In the case of PWRs, the systems must be capable of removing heat both from the containment atmosphere and the containment sump (or equivalent long term recirculation source of water). The NRC staff reviews the capability of the system to withstand a SF, the heat removal capability of the system, the performance characteristics of heat exchangers in the system, the proposed inspection and testing programs, the availability of net positive suction head (NPSH) to the ECCS, and/or containment heat removal pumps, the design of any water sources for those pumps, and any impacts from accident generated debris or chemical effects on long-term core cooling.

6.2.2.2 Summary of Application

6.2.2.2.1 Containment Spray System

The CHRS credited in the APR1400 DCD is the CSS. The CSS acts to reduce containment pressure and temperature following a MSLB or a LOCA. It consists of two divisions, each with a pump, heat exchanger, spray header and associated piping, and valves capable of delivering 100 percent of the required flow. In addition, the CSS system serves as a functional backup to the shutdown cooling system, and each system can perform the function of the other interchangeably with the realignment of remotely operated valves.

As a safety-related system, the CSS is classified as seismic Category I, protected from postulated natural events and phenomena, and each train is powered from its own separate class 1E electrical bus energized by an EDG. The CSS pumps start on the receipt of a safety injection actuation signal (SIAS) or a containment spray actuation signal (CSAS). As stated in DCD Tier 2, Section 6.2.2, the design bases of the CSS are to remove heat such that the containment design pressure and temperature are not exceeded, to reduce containment pressure to half of the peak pressure within 24 hours after a high-energy line break, and to remove fission products (a function evaluated in Section 6.5 of this SER). All of these functions should remain possible in the event of a SF in the system.

The CSS accomplishes these functions by pumping water from the IRWST through a heat exchanger into the containment spray headers in the containment dome, where the flow is then emitted in the form of droplets by the spray nozzles. The droplets fall through the containment atmosphere, cooling the atmosphere as they do so, until they evaporate or reach a surface and eventually drain back down into the holdup volume tank (HVT), which spills back into the IRWST. An emergency containment spray backup system is also provided, but is not credited for the DBA analyzed here.

6.2.2.2.2 In-containment Water Storage System

DCD Tier 1, Section 2.4.2, "In-containment Water Storage System," contains a description of the in-containment water storage system (IWSS), which consists of the IRWST, HVT, and cavity flooding system. Section 2.4.2 describes the functional arrangement of the IWSS, lists the piping classifications and components used in the system, and describes the controls associated with the IWSS and the system functionality. Classifications of piping, equipment, and instrumentation associated with the system are specified in DCD Tier 1, Tables 2.4.2-1, "In-containment Water Storage System Equipment and Piping Location/Characteristics," through 2.4.2-3, "In-containment Water Storage System Instrument List." In addition to the equipment associated with the IRWST and HVT, post LOCA pH control is a part of the IWSS and is accomplished via tri-sodium phosphate (TSP) baskets, also described in DCD Tier 1, Section 2.4.2. ITAAC for the IWSS is located in DCD Tier 1, Table 2.4.2-4.

DCD Tier 1, Section 2.11.2, describes the arrangement and functions of the components of the CSS. Classifications of piping, equipment and instrumentation associated with the system are specified in DCD Tier 1, Tables 2.11.2-1, "Containment Spray System Equipment and Piping Location/Characteristics," through 2.11.2-3, "Containment Spray System Instruments List." ITAAC for the CSS is located in Table 2.11.2-4, "Containment Spray System ITAAC." Additional features related to the IWSS and CSS are described in DCD Tier 1, Section 2.2.1, such as the containment building and internal structures. Both the IRWST and HVT are an integral part of the containment building, and thus have structural components that appear in DCD Tier 1, Sections 2.2.1 and 2.4.2.

DCD Tier 2, Section 6.2.2, describes the functional arrangement of the CSS, the pump and heat exchanger performance, and the nozzle characteristics. The CSS contains two normally open valves in the piping leading from the IRWST to the spray piping up to the header, which contains two normally closed valves isolating the spray header and two normally open block valves used to double isolate the spray header in the event of spray pump testing or shutdown cooling. The nozzles themselves are arranged in an overlapping pattern, arranged on the nozzle header at different angles in order to cover the vast majority of the containment volume with the spray pattern, suitable both for heat removal and fission product removal, which is evaluated in Section 6.5 of this SER.

TR APR1400-E-N-NR-14001-P, "Design Features to Address GSI-191," Revision 0 (referenced in the DCD), provides an assessment of the means the APR1400 design employs to meet the guidelines of RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant-Accident." The technical report contains a detailed discussion of the break selection criteria, the debris generation, classification and transport, the available net positive suction head (NPSH), and the upstream and downstream effects resulting from debris. DCD Tier 2, Section 6.8.4.5, "Performance of the IRWST Sump Strainer," summarizes each of these contributions to the design. DCD Tier 2, Section 6.8.4.5.2, "Debris Generation," states that all of the insulation on piping takes the form of reflective metal insulation (RMI). DCD Tier 2, Section 6.8.4.5.9, states that only 6.8 kg (15 lbm) of latent fiber is present in the design, and that this is the only fibrous debris loading. This value is to be controlled by a cleanliness and foreign material exclusion program to be implemented by the COL applicant.

DCD Tier 2, Section 6.8, describes the IWSS. Both the IRWST and HVT are integrated into the containment building structure, and take the form of large concrete volumes lined with stainless steel. The IRWST is the primary water source for both the CSS and the safety injection (SI) system during normal operation and under accident conditions, the heat sink for discharges from RCS relief and vent lines, and is also the source of water used to fill the refueling cavity during refueling operations. The IRWST is outfitted with pressure relief panels to accommodate the pressure build-up that could result from POSRV discharge, and the panels discharge to the bulk containment atmosphere. At the bottom of the IRWST, four debris strainers are submerged, one over the suction source for each of the ECCS trains. The HVT is the low elevation collection volume for water following a postulated accident and serves as an intermediate volume between the IRWST and the reactor cavity to protect against leakage into the cavity and receive the IRWST water in the event cavity flooding is required. A trash rack and vertical screens line the entrances to the HVT to prevent large debris from entering the HVT and thereby the IRWST, and the spillway connecting the two also assists in preventing debris from entering the IRWST and interfering with recirculation.

TR APR1400-E-N-NR-14001-P also addresses the strainers located in the IRWST. The four strainers are located such that they are fully submerged during all postulated accidents and are sized such that sufficient NPSH is available for the ECCS pumps even under the worst case design basis debris loading.

TS associated with the CSS and IRWST are specified in DCD Tier 2, Chapter 16, Sections 3.5, “Emergency Core Cooling Systems (ECCS),” and 3.6, “Containment Systems.”

Initial testing of the CSS and the IRWST is discussed in DCD Tier 2, Section 14.2, “Initial Plant Test Program,” which describes the testing performed prior to plant operation. Specifically, pre-operational testing for these systems is addressed by Section 14.2.13.1.38, “Containment Spray System Test” and 14.2.13.1.40, “In Containment Water Storage System Test.” Other tests are performed that utilize the systems described in this Section, such as the SIS test and the safety and relief valve testing.

6.2.2.3 Regulatory Basis

The relevant requirements for the Commission regulations for Section 6.2.2 and Section 6.8, and the associated acceptance criteria, are given in Section 6.2.2 of NUREG–0800. The application is required to satisfy the following Commission regulations:

- GDC 4 in 10 CFR Part 50, Appendix A, as it relates to the ability of structures, systems, and components (including pumps, valves, and strainers) important to safety to accommodate the effects of and to be compatible with the dynamic and environmental conditions associated with postulated accidents.

- GDC 35, as it relates to providing abundant emergency core cooling to transfer heat from the reactor core following a postulated design basis accident.

- GDC 38, which requires that:

    - The CHRS be capable of rapidly reducing the containment pressure and temperature following a LOCA and to maintain these parameters at acceptably low levels.

    - The CHRS performs in a manner consistent with the function of other systems.

    - The safety-grade design of the CHRS provides suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capability to ensure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished in the event of a SF.

- GDC 39, as it relates to the design of the CHRS to permit periodic inspection of components.

- GDC 40,as it relates to: (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

- The regulations in 10 CFR 50.46(b)(5), as it relates to the requirements for long-term cooling, including adequate NPSH margin in the presence of LOCA-generated and latent debris.

Acceptance criteria adequate to meet the above requirements include:

- The CHRSs should meet the redundancy and power source requirements for an ESF (i.e., the results of failure modes and effects analyses of each system should ensure that the system is capable of withstanding a SF without loss of function). This conforms to the requirements of GDC 38.

- With regard to GDC 38, as it relates to the capability of the containment system to accomplish its safety function, the spray system should be designed to accomplish this without pump damage caused by cavitation. A supporting analysis should be presented in sufficient detail to permit the staff to determine the adequacy of the analysis. This analysis should also demonstrate that the available NPSH is greater than or equal to the required NPSH. RG 1.82, Revision 4[2], describes methods acceptable to the staff for evaluating the NPSH margin. If containment accident pressure is credited in determining available NPSH, an evaluation of the contribution to plant risk from inadequate containment pressure should be made.

- In evaluating the performance capability of the CSS to satisfy GDC 38, the analyses of its heat removal capability should be based on the following considerations:

    - The locations of the spray headers relative to the internal structures.

    - The arrangement of the spray nozzles on the spray headers and the expected spray pattern. The spray systems should be designed to ensure that the spray header and nozzle arrangements produce spray patterns, which maximize the containment volume covered and minimize the overlapping of the sprays.

    - The spray drop size spectrum and mean drop size emitted from each type of nozzle as a function of differential pressure across the nozzle.

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[2] Regulatory Guide 1.82, Revision 4, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident," March 2012 (ML111330278).

- o The effect of drop residence time and drop size on the heat removal effectiveness of the spray droplets.

- In evaluating the heat removal capability of the CHRS to satisfy GDC 38, the potential for surface fouling of the secondary sides of fan cooler, recirculation, and RHR heat exchangers by the cooling water over the life of the plant and the effect of surface fouling on the heat removal capacity of the heat exchangers. The application should discuss the results of the analysis. The results will be acceptable if they demonstrate that provisions such as closed cooling water systems are provided to prevent surface fouling or that surface fouling has been taken into account in the establishment of the heat removal capability of the heat exchangers.

- To satisfy the requirements of GDC 38 and 10 CFR 50.46(b)(5), regarding the long-term spray system(s) and ECCS(s), the containment emergency sump(s) in PWRs and suppression pools in boiling-water reactors (BWRs) should be designed to provide a reliable, long-term water source for ECCS and CSS pumps. The containment design should allow for the drainage of spray and emergency core cooling water to the emergency sump(s) or suppression pool and for recirculation of this water through the CSS and ECCS. The design of the sumps or suppression pools and the protective strainer assemblies is a critical element in ensuring long-term recirculation cooling capability. Therefore, adequate design consideration of: (1) sump and suppression pool hydraulic performance, (2) evaluation of potential debris generation and associated effects including debris screen blockage, (3) RHR and CSS pump performance under postulated post-LOCA conditions, and (4) impacts of debris penetrating strainers on long-term coolability of the core is necessary. RG 1.82, Revision 3, as modified and supplemented for PWRs by the Nuclear Energy Institute (NEI) Guidance Report (GR) and the NRC SE, "NRC Safety Evaluation of NEI GR , 'Pressurized Water Reactor Containment Sump Evaluation Methodology'" (ML043280631) provide guidance for PWR debris evaluations.

- In meeting the requirements of GDC 39 and GDC 40, regarding inspection and testing, the design of the CHRSs should provide for periodic inspection and operability testing of the systems and system components such as pumps, valves, duct pressure-relieving devices, and spray nozzles.

  To satisfy the system design requirements of GDC 38, instrumentation should be provided to monitor the performance of the CHRS and its components under normal and accident conditions. The instrumentation should determine whether a system is performing its intended function or whether a system train or component is malfunctioning and should be isolated.

The staff reviewed DCD Tier 2, Revision 0, Section 6.2.2 and 6.8 in accordance with SRP Section 6.2.2, Revision 5, BTP 6-5, "Reactor Systems Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps," Revision 3, RG 1.206, Revision X, and RG 1.82, Revision 4.

The staff's review was supplemented by the following guidance:

- NEI Guidance Report NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, (ML050550138)

- "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," (ML050550156)

- RG 1.82 also describes methods acceptable to the staff for evaluating the NPSH margin.

- "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated April 6, 2010 (ML100960495)

- Enclosure 2, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation," to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated March 28, 2008 (ML080230234)

- "Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report (TR) WCAP-16406-P-A, Revision 1, 'Evaluation of Downstream Sump Debris Effects in Support of GSI-191,' Pressurized Water Reactor Owners Group, Project No. 694," NRC, Washington, DC, December 2007. (ML073520295)

6.2.2.4 Technical Evaluation

In reviewing the CHRS, the staff examined a number of specific areas of review, including: the consequences of SF in the system, the capability of the heat removal of both the spray water and whatever residual heat removal system exists, the design provisions providing for testing and inspection of the system, and the availability of the water source for both the ECCS and the CSS. By focusing on these areas, the staff ensured the application meets the requirements associated with GDC 38, GDC 39 and GDC 40, as well as 10 CFR 50.46(b)(5).

Consequences of SF in the system were evaluated through examination of the system layout and the failure modes and effects analysis presented by the applicant. Evaluation of the CHRS consisted of two parts: the CSS and the IRWST, the long term recirculation water source for both the CSS and the ECCS systems. The CSS is the primary means of removing heat from the containment atmosphere following a postulated break, and is discussed in Section 6.2.2.4.1 of this SER. The IRWST is a large tank of water (627,000 gal) that serves as the initial water source for the CSS and ECCS systems and later functions as the collection point for recirculating water for longer term heat removal. At the bottom of the IRWST, four strainers assist in filtering some debris from traveling downstream in the recirculation path. In Section 6.2.2.4.2 of this SER, the safety functions of the IRWST and the rest of the long term water source are evaluated.

6.2.2.4.1 Containment Spray System

The CSS consists of two trains, each of which is independently able to deliver 100 percent of the required flow for cooling containment during DBA conditions. The CSS also serves as a fission product removal system, and this function is evaluated in Section 6.5 of this SER.

Due to its nature as a system with two independent trains, the CSS is relatively well protected against SF. A failure modes and effects analysis is presented in DCD Tier 2, Table 6.2.2-3; in general, each system feature results in the loss of only one train of spray. Provisions for venting spray lines due to voiding, a potential common cause failure, were not discussed in Section 6.2.2. Therefore, on June 11, 2015, the staff issued RAI 25-7844, Question 06.02.02-4 to address this issue (ML15162A004). As a result, the applicant revised DCD Revision 1 to discuss the design provisions made to mitigate against voiding, and included information related to high point vents. Further, the applicant added ITAAC for the SI, shutdown cooling (SC), and CS systems. These ITAAC require an analysis of the potential gas intrusion mechanisms, provide for periodic monitoring of systems, and provide for inspection to ensure the as-built design has installed high point venting available, which would act to preclude voiding.

The two trains, however, do share a common source of water in the IRWST. While the IRWST is designed in a fashion such that it will survive and continue to perform its function during design basis considerations including a SF, the IRWST does have severe accident functions to flood the reactor cavity via the HVT using linked valves located low in the tank. These linked motor-operated valves (MOVs) are locked closed and disconnected from all power sources except under severe accident scenarios where they are used. In Revision 1 of the DCD, the applicant updated the description of these valves. Based on the above considerations, due to the fact that no single active failure can disable functionality, the staff finds the CSS meets the redundancy criteria required in GDC 38.

The capability to periodically test and inspect the CSS is assured through ITAAC, the initial test program, and TS surveillances. ITAAC for the CSS ensure the system, as installed: performs as designed by verifying the components in the system meet the necessary classifications; can withstand the environment following a postulated accident; has sufficient separation between trains; provides the necessary controls and alarms for controlling the system; and provides sufficient flow under postulated post-accident conditions. As part of DCD Revision 1, ITAAC was revised to provide analysis for the containment holdup volumes, a key input for NPSH for the ECCS systems.

The initial test program ensures the IRWST and CSS adequately satisfy the design assumptions. The initial test program, as described in DCD Tier 2, Section 14.2, “Initial Test Program,” did not sufficiently ensure that the design assumptions, such as adequate NPSH under postulated accident conditions, set forth in DCD Tier 2, Section 6.2.2.2, were achieved. Therefore, on June 1, 2015, the staff issued RAI 19-7899, Question 14.02-1, to address this issue (ML15152A518). As part of DCD Revision 1, the applicant provided additional items to clarify the test method in DCD Tier 2, Section 14.2.12.1.38, “Containment Spray System Test.” The added items make clear that the CSS system head requirements are characteristics of the entire system, not only the pumps, and that the static head of the as-built pumps conforms to the assumptions in the applicant’s analyses in DCD Section 6.2.1. TS surveillances ensure that valves in the CSS system are correctly positioned, pumps output the amount of flow at the appropriate pressure assumed by the safety analyses, that equipment repositions appropriately upon receipt of a signal, and that the spray nozzles are unobstructed, in accordance with SRP Section 6.2.2. Taken together, staff finds these provisions allow the CSS to meet the periodic inspection requirements of GDC 39 and testing requirements associated with GDC 40.

The spray nozzle orientation and locations are detailed in Figures 6.2.2-2, “Containment Spray Nozzle Orientation,” through 6.2.2-4, “Spray Nozzle Header Elevations,” of Tier 2 of the DCD. Based on this orientation, the applicant assumes the entirety of the containment volume above the operating floor to be sprayed, which equates to 75 percent of the free volume of the containment (DCD Tier 2, Table 6.5-3, “Sprayed and Unsprayed Regions”). Furthermore, the design incorporates auxiliary spray nozzles below the operating floor that are not credited in coverage volume, which can be used to further stimulate mixing and cooling in the region. The approach taken by the applicant is simplified but acceptable for containment mixing, similar to assumptions made in previously approved designs using sprays for containment cooling. Although high in the dome region near the spray rings, coverage may not be fully achieved however flow paths leading below the operating deck exist and will allow for mixing. On the basis of the substantial coverage provided by the nozzle header arrangement combined with their locations high in containment, the staff finds the sprays adequately cover the containment volume.

At the design pressure differential, the mean diameter of the spray droplets is approximately 300 microns; at half the design pressure, the mean drop size is approximately 370 microns. As larger droplet sizes are more conservative with respect to heat transfer (due to the smaller surface area per volume of spray) the applicant assumed a mean diameter of 1000 microns for the containment response analysis using the GOTHIC code. Staff calculations indicated a 1000 micron droplet is capable of fully participating in heat transfer with the containment atmosphere in approximately 1 second, long before a droplet could fall the shortest distance possible in the APR1400 design (from the lowest spray header to the top of the pressurizer compartment), and therefore the staff finds the applicant’s modeling choice is acceptable.

SRP Section 6.2.2 specifies that the application should contain a spectrum of expected spray drop sizes as a function of pressure in order to provide assurance that the values assumed in the DCD bound the droplet sizes expected to be seen during system operation. Although the DCD contained an average value, no such spectrum was provided. In its response to RAI 25-7844, Question 06.02.02-5(ML15204A708), dated July 23, 2015, the applicant provided the drop size spectrum and relative frequency, and updated DCD Revision 1, to contain the expected spectrum of drop sizes.

The heat removal capability of the system with respect to coverage, drop size and drop residence time is not as restrictive as the iodine removal capability of the system. As stated in SRP Section 6.2.2, the staff has determined that if the CSS is capable of adequately removing iodine, it is also capable of satisfactory heat removal. In Section 6.5 of this SER, the staff’s findings related to the iodine removal capability of the system are discussed. The staff finds the heat removal capability of the system to be acceptable, considering that: the spray droplets reach thermal equilibrium; the system is capable of removing iodine adequately; and the spray system was found to reduce containment pressure and temperature following a postulated accident, as evaluated by the staff in the discussion of containment peak pressure and temperature analyses in Section 6.2.1.1.A, “PWR Dry Containments, Including Subatmoshperic Containments,” of this SER.

The fouling factors used in the CSS heat exchangers are an important parameter determining the heat removal provided by the CSS. During the lifetime of the heat exchangers, chemical and/or biological buildup and corrosion will contribute to fouling of the tubing in the heat exchanger, slightly reducing its efficiency over time. In its response to RAI 25-7844, Question 06.06.06-7 (ML15204A708), dated July 23, 2015, the applicant provided additional data on the CS heat exchanger, including the modeling conditions in the containment analyses and updated the DCD to provide the basis for the fouling factor used in the analyses.

The containment spray pumps are also capable of serving interchangeably with the shutdown cooling pumps (SCPs). The adequacy of the SCP is assessed in Section 5.4.7, “Residual Heat Removal (RHR) System,” of this SER.

Instrumentation associated with the CSS is listed in DCD Tier 2, Table 6.2.2-4, “Containment Spray System Display Instrumentation,” and the role of the instrumentation in the ESF actuation system is described in DCD Tier 2, Section 7.3, "Engineered Safety Features Systems." Specifically, the CSS actuates on a high-high containment pressure signal of 138.1 kPa (20.03 psia), as compared with the normal operating range of containment pressure of 0.0 to 6.2 kPa (0.0 to 0.9 psia). Monitoring of CSS operation is accomplished through instrumentation providing indication in the control room for: pump suction and discharge pressure, flow, valve position, pump motor parameters, and heat exchanger temperature. Taken as a whole, these instruments meet the requirement for the design to have instrumentation capable of monitoring the system during normal and accident conditions and therefore meet the design instrumentation requirements that are part of GDC 38.

6.2.2.4.2 In-Containment Refueling Water Storage Tank (IRWST)

The IRWST serves as the water source for the SI and CSSs, both as the water source for the initial phase of postulated accidents and, in the long term, as the collection point for recirculating water. Initially, the IRWST is filled with borated water both for refueling operations and the ECCS suction source.

In order to fulfill its design function, the IRWST must accommodate projected operation conditions. The IRWST contains spargers, which accommodate discharges from the POSRVs, head vent valves and pressurizer vent valves. The spargers are designed to quench a steam release from the aforementioned RCS vent connections. The location of the spargers is identified in DCD Tier 2, Figure 6.8-1, "IRWST and HVT Plan View," and the flow paths to the spargers are displayed in DCD Tier 2, Figures 5.4.12-1, "Reactor Coolant Gas Vent System Flow Diagram," and 5.1.2-3, "Pressurizer and POSRV Flow Diagram."

The IRWST is protected against pressure transients via swing panels on the sides of each of four vent stacks atop the IRWST. Each vent stack has one panel designed for vacuum protection and two panels for overpressure protection. The panels are designed to accommodate differences in pressure resulting from discharges from the spargers, operation of ECCS equipment, as well as normal operating functions without the need for operation. On April 28, 2016, the staff issued RAI 472-8564, Question 06.02.05-9, requesting the applicant to provide additional detail on the IRWST vent operating criteria, dimension, and location. In its response to RAI 472-8564, Question 06.02.05-9, dated July 7, 2016 (ML16189A319), the applicant stated the outflow swing panels open at a differential pressure of 0.5 psi and the inflow swing panels open at a differential pressure of 1.5 psi. Additional detail drawings with the locations of the vent stacks were also provided in the response. The efficacy of the panels for mitigation of combustible gas concentrations is evaluated in Section 6.2.5 of this SER.

The IRWST is designed to withstand the hydrodynamic loads on the walls and structures resulting from the actuation of the spargers. The spargers provide a controlled release of steam flow that is quenched in the IRWST following a POSRV actuation. Discussion related to the hydrodynamic loading on the structure of the IRWST due to actuation of the spargers is located in Section 3.8.3 of this SER.

As part of a response to RAI 8245, Question 03.08.03-1 (ML15313A533 and ML16273A563), the staff requested the applicant to provide further detail regarding the pressure transient resulting from sparger actuation in the IRWST. The applicant used the same method used in the System 80+ design, scaled to account for differences in the plant designs. That method was based on methods reviewed and accepted by the staff for calculating hydrodynamic loads in BWRs (SRP Section 6.2.1.1.C), and the mechanics of the sparger discharge in the IRWST are practically identical to those in suppression pool discharge. The staff reviewed the methodology and scaling used by the applicant in its response, and in conjunction with the margin factors applied to the calculated pressure and frequency by the applicant, views them as appropriately conservative.

ITAAC for the IRWST are included with the IWSS in Tier 1, Table 2.4.2-4 of the DCD. The ITAAC ensure the IWSS conforms to the design as described in the DCD by providing the classifications for components and provides the necessary controls and alarms for monitoring system operation. In addition, ITAAC are provided to ensure the IRWST water volume meets the minimum requirements specified in Section 6.8 of the DCD. Section 2.4.2 of DCD Tier 1 provides a design description for the IWSS; tables listing system equipment, piping, components, and instruments; proposed ITAAC; and a functional diagram of the system. The staff ensured that the Tier 1 information is consistent with DCD Tier 2. In addition, the staff determines that the proposed ITAAC in Table 2.4.2-4 of DCD Tier 1 are necessary and sufficient to provide reasonable assurance that, if met, the IWSS will be in conformity with the certified design, the applicable regulations, in accordance with 10 CFR 52.47(b)(1). This conclusion is based on the inclusion of comprehensive inspections, tests, and analyses that verify the acceptability the IWSS. Specific ITAAC are discussed in further detail below. TS surveillances verify that the strainers are operable and the IRWST fluid conditions remain within the boundaries assumed in DCD Tier 2 Chapter 6 analyses.

6.2.2.4.3 Long-Term Recirculation Water Source

DCD Tier 2, Section 6.8, describes the IWSS, which encompasses collection and storage of water as well as serving as a heat sink during both normal operation and accident conditions. Thus, a large part of the safety function of the system involves its role as the long term water source for both the SI as part of SI and the CSS for containment pressure and temperature suppression during postulated accidents and radionuclide removal during severe accidents.

Operational experience has revealed that insulation and other materials inside containment have the capability to form a debris bed on the sump or recirculation source. In the event of a LOCA, debris generated in the vicinity of the break combined with latent debris existing in containment will transport to the containment sump strainer. In the absence of design considerations to mitigate against this, the debris that forms on the strainer could result in excessive head loss, which could impede the flow of water into the SI and/or CSSs. This issue is referred to as GSI-191.

The following Sections, 6.2.2.4.3.1, “Break Selection,” through 6.2.2.4.3.12, “Chemical Effects,” of this SER, form the basis for the applicant’s compliance with the requirements of GDC 35 to provide abundant emergency core cooling to remove decay heat and 10 CFR 50.46(b)(5) for long-term cooling, including demonstrating adequate NPSH margin in the presence of LOCA-generated and latent debris.

In the APR1400 design, the IRWST serves as the initial source of water for the CSS and SISs. In the long term, the containment floods to the point where the HVT fills and spills over back into the IRWST. The efficacy of debris mitigation following a limiting LOCA and the criteria determining the limiting break are discussed below.

6.2.2.4.3.1 Break Selection

The goal of break selection with respect to the long-term recirculating water source is to identify the break location that results in the most limiting pressure drop across the strainer. RG 1.82, regulatory position C.1.3.2.3 and NEI 04-07, Sections 3.3.4, “Identifying Break Locations,” and 4.2.1, “Break Selection,” and the associated SE provide criteria that should be considered to identify the limiting break. The applicant's break selection is discussed in Section 3.1.2, "Selection of Postulated Break Location," of APR1400-E-N-NR-14001-P, "Design Features to Address GSI-191," and Section 6.8.4.5.1, "Break Selection," of the DCD, Tier 2.

The primary sources of debris in the APR1400 design are insulation, coating, and latent debris. By the nature of the latent debris, its loading is independent of the selected break – all 200 lbm of latent debris is assumed by the applicant to be a part of the debris source regardless of the break. Similarly, the debris resulting from chemical effects does not depend on the break location. In addition, the applicant stated no fibrous insulation is present inside the APR1400 containment. As such, limiting breaks are defined by size and potential to produce the largest and most varied debris loading.

NEI 04-07 and its associated SE state that main steam and feedwater lines need only be analyzed as potential break location in plants where ECCS recirculation is required to mitigate the effects of breaks in these lines. Because APR1400 relies on establishing decay heat removal from the SGs if the RCS is intact, no ECCS recirculation is required in the event of a feed system break. A MSLB is considered in the debris generation analysis as such an event would result in an SIAS and CSAS.

Three pipe break Sections were analyzed: the 42" diameter RCS hot leg, the 30" RCS cold leg, and 30.907" main steam line. As stated by the applicant in DCD Tier 2, Section 6.8.4.5.1, because the SG compartment has the largest volume of reflective metal insulation (RMI) and coatings, as well as the largest volume of piping, the limiting break was determined to be the junction of the RCS hot leg and the SG. According to the applicant, this break location generates the largest amount of debris. The staff reviewed the detail drawings associated with the break locations and determined they were located in the areas with the highest concentration of surrounding piping insulation, and that it was reasonable to conclude that these areas would also produce a limiting loading of coatings. Given the relatively straightforward break selection process, due to the lack of fibrous insulation and relatively location independent debris source terms, as well as the consistency of the approach with the guidance in RG 1.82 and the SE associated with NEI 04-07, the staff finds the selection of limiting break acceptable.

6.2.2.4.3.2 Zone of Influence and Debris Generation

Debris is generated from a break in a zone of influence (ZOI) around the break. The ZOI is the volume around the break in which the forces resulting from the break release would be sufficient to damage materials, and the size of the ZOI is dependent on the material. The applicant evaluated ZOIs consistent with NEI 04-07, Sections 3.4.2, "Zone of Influence," and 4.2.2, "Debris Generation," and the associated staff SE, and used the suggested destruction pressures (and associated ZOI radii) from Table 3-2, "Revised Damage Pressures and Corresponding Volume-Equivalent Spherical ZOI Radii," of the SE for NEI 04-07. These ZOI radii, corresponding to experimentally tested destruction pressures, are calculated by applying the standard jet expansion model ANSI/ANS-58.2, which transforms a free-expanding jet for a given break into an equivalent spherical volume for use with a pipe break. The validity of the ANSI/ANS-58.2 and the approved ZOIs for various materials are discussed in Appendix I of the staff's SE for NEI 04-07.

The applicant's ZOI and debris generation evaluations and methods are presented in APR1400-E-N-NR-14001-P, Appendix B, "Debris Generation Evaluation for the APR1400." The applicant used material-specific ZOIs, consistent with staff guidance. By superimposing the ZOIs for the different materials onto design drawings and evaluating the specifications for components within the ZOI, the debris for each postulated break can be calculated. For RMI, a ZOI of 2 pipe diameters (2D) is used; for coatings, a ZOI of 4D (epoxy) and 10D (IOZ) is used. These zones of influence are consistent with the jet impingement damage zones specified in Table 3-2 of the staff's SE on NEI 04-07.

Staff experience has shown that there may be other sources of debris from miscellaneous sources, such as instrumentation and power cables, in the ZOI. Some of this debris could be fibrous in nature and would, therefore, challenge the assumptions made in the debris analysis for the APR1400. Therefore, on May 22, 2015, the staff issued RAI 12-7902, Question 06.02.02-1 to address this issue (ML15142A447). In its response to RAI 12-7902, Question 06.02.02-1 (ML15173A087), dated June 22, 2015, the applicant stated that the cables are made of a polypropylene material and do not contain fiber. In addition, the applicant stated that all sources of fiber that could affect ECCS capability were assessed and none were present in a break ZOI. As such, latent fibrous debris makes up the only fiber input to the strainer head loss and downstream analyses. This criteria is ensured through a foreign materials exclusion program that is to be instituted by a COL applicant, as specified by COL action item 6.8(2) (discussed below).

The staff reviewed the applicant's ZOI and debris generation evaluations, as presented in APR1400-E-N-NR-14001-P, relying on the methods approved in NEI 04-07 and the associated staff SE for NEI 04-07 as a guide. The staff finds that the applicant's approach, which is consistent with staff guidance on debris-specific ZOIs, is acceptable.

6.2.2.4.3.3 Debris Characteristics (including Latent and Miscellaneous Debris)

Different types of debris are present in the post-accident environment following a LOCA. Guidance for characterizing debris can be found in NEI 04-07, Section 3.4.3, "Quantification of Debris Characteristics," and the associated SE. The debris characteristics are discussed in DCD Tier 2, Section 6.8.4.5.3,"Debris Characterization," and detailed in Section 3.3, "Debris Characteristics," of APR1400-E-N-NR-14001-P. Four types of debris are characterized – RMI, latent debris, coatings, and chemical precipitates. Coatings are discussed below in Section 6.2.2.4.3.4, and the chemical precipitates are discussed in Section 6.2.2.4.3.12 below.

RMI, which surrounds most of the piping in containment as insulation, is assumed by the applicant to be destroyed and broken into debris within the ZOI of the break. The debris is assumed to consist of 25 percent small pieces ("fines" in the DCD) and 75 percent larger pieces. This size breakdown is consistent with the guidance offered in NEI 04-07 and endorsed by the NRC in RG 1.82.

Latent debris is characterized as dirt, dust, fiber, and miscellaneous materials present (not intentionally) inside containment during operation prior to a postulated break. Latent debris is further subdivided into two types, fiber and particulate. Latent fiber could theoretically take a number of forms, and so for testing an equivalent material, Nukon fiberglass, is used. In the SE for NEI 04-07, Section 3.5.2.3, "Sample Calculation," the staff found Nukon to be a suitable surrogate for the latent debris found inside containment. Most importantly, properties for water submerged or suspended fiberglass are similar to that of the miscellaneous fiber found inside containment, given that an appropriate size distribution is used.

In the APR1400 design, all fibrous debris is assumed to take the form of fines, in the form of class I or II fibrous debris in NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance." This smaller size results in more of the fiber making its way to the IRWST. In an audit (ML17037A756), staff witnessed the preparation of the surrogate fibrous material used in the testing and agreed with the applicant's determination that the fiberglass fines used in the head loss testing credited in APR1400-E-N-NR-14001-P were a reasonable approximation of latent fiber that would be found in containment during operation.

Fibrous debris makes up only a portion of the latent debris in containment; the majority of the latent debris is in the form of particulate (dirt, sand, etc.). NEI 04-07 suggests that 90.7 kg (200 lbm) of latent debris split 15 percent as fiber and 85 percent as particulate is an appropriate assumption for a clean containment debris loading. While APR1400 assumes 90.7 kg (200 lbm) of debris, the split assumed is 7.5 percent fiber and 92.5 percent particulate. As fiber is a more limiting load on the strainers, this could present a non-conservative assumption. Therefore, on May 22, 2015, the staff issued RAI 12-7902, Question 06.02.02-2 to address this issue (ML15142A447). In general, the staff finds no issue with a more strict administrative control on fiber, but the established cleanliness, housekeeping, and foreign material exclusion (FME) program instituted should contain provisions to limit the latent fiber in containment to 15 lbm, as indicated in DCD Tier 2, Table 6.8-3. In DCD Revision 1, the applicant revised the cleanliness program to include an acceptance criteria of 200 lbm (90.7 kg) of latent debris of which at most 15 lbm (6.8 kg) can be fiber.

Additional miscellaneous debris, such as equipment tags, labels, and tape, also need to be accounted for as part of the debris loading on the strainer. The applicant assumed that 9.29 m^2 (100 ft^2) per strainer is blocked by miscellaneous debris in APR1400-E-N-NR-14001-P but notes that the exact area is to be determined by the COL applicant as part of the containment cleanliness program. Such a value serves as the upper bound for the amount of miscellaneous debris and is consistent with other DC applications reviewed and accepted by the staff.

In order to ensure that debris remains below the limits specified above, the DCD includes a COL information item for the COL applicant to establish programmatic controls that prevent debris from being introduced into containment that would impact the function of the IRWST, strainers, or ECCS systems. These controls are described in DCD Tier 2, Section 6.8.4.5.10. Specifically, COL applicants should do the following, per the discussion in DCD Tier 2 Section 6.8.4.5.10 and COL items 6.8(1), 6.8(2), 6.8(3) and 6.8(6):

- Establish a cleanliness, housekeeping, and foreign material exclusion (FME) program that restricts latent debris and miscellaneous debris to the values found acceptable above.
- Implement procedures to ensure that administrative controls for plant modifications and temporary changes consider and evaluate, in accordance with 10 CFR 50.59 and 10 CFR 52.63, materials that could contribute to sump strainer blockage.
- Evaluate the acceptability (by assessing and managing the risk associated with) of maintenance and temporary changes.
- Establish a containment coatings monitoring program in accordance with the guidance in RG 1.54, Revision 2.

These items address the requirements to account for and control debris that could be introduced into containment; however, no acceptance criteria were included to ensure that the cleanliness program restricts the containment debris to the amount assumed in the strainer performance analysis. The applicant added the above acceptance criteria for the cleanliness program as part of Revision 1 of the DCD. The staff finds the treatment and characterization of the debris to conform to the staff approved guidance and is therefore acceptable.

COL Items

Item No.	Description	DCD Tier 2 Section
COL 6.8(1)	The COL applicant is to provide the operational procedures and maintenance program for leak detection and contamination control. Note: this COL item is also reflected as part of COL Items 6.3(1) and 6.5(1).	6.2.4.1.1, 6.3.6, 6.8.4.5.10
COL 6.8(2)	The COL applicant is to provide the preparation of cleanliness, housekeeping, and foreign materials exclusion program.	6.8.4.5.10
COL 6.8(3)	The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations. Note: this COL item is also reflected as part of COL Items 6.3(2) and 6.5(2).	6.8.4.5.10
COL 6.8(6)	The COL applicant is responsible for the establishment and implementation of the Maintenance Rule program in accordance with 10 CFR 50.65.	6.1.2.1, 6.8.4.5.10

6.2.2.4.3.4 Coatings Evaluation

In reviewing whether the APR1400 design meets the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5) with respect to protective coatings (paint) in containment, the staff used the guidance documents listed above in Section 6.2.2.3.

DCD Tier 2, Section 6.1.2, “Organic Materials,” contains a discussion of the protective coatings used inside containment. The staff’s evaluation is included in Section 6.1.2 of this SER. SL I coatings are used inside containment in areas where failure of the coatings could adversely impact post-accident system operation, such as washdown transport to the strainer, which could result in recirculation flow blockage. Qualified coatings remain intact outside the ZOI (Section 3.3 of APR1400-E-N-NR-14001-P), and all unqualified coatings in containment will fail. As per Section 2.6, “Coatings,” of APR1400-E-N-NR-14001-P, there will be no unqualified coatings inside containment.

The applicant assumed that coatings debris would be in the form of small particles. In its response to RAI 391-8462, Question 06.02.02-32, dated July 1, 2016(ML16183A314), the applicant provided the basis for assuming only the particulate form. Since all coatings in containment are qualified with respect to DBA conditions, coatings will be assumed to fail only within the ZOI and in the form of particles approximately 10 micrometers (0.00039 in) in diameter. The staff finds this acceptable because it conforms to the staff guidance on coatings listed above. In developing its guidance the staff determined that small particles are conservative for plants that can form a fibrous debris bed that can trap particles. The response proposed changes to DCD Tier 2, Section 6.8.4.5.3 and APR1400-E-N-NR-140001-P Section 3.3 to supplement the discussion of the coating debris form. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, and APR1400-E-N-NR-140001, Revision 1, dated March

31, 2017, were revised to incorporate the proposed changes. Therefore, RAI 391-8462, Question 06.02.02-32 is resolved and closed.

Coatings are stated in DCD Tier 2, Section 6.8.4.5.3 to fail within the ZOI for a given break. As discussed in Section 6.2.2.4.3.2, for epoxy coatings, a ZOI of 4D is used and for inorganic zinc, a ZOI of 10D is used; these values have been approved by the staff in the reports referenced above. The amount and type of coatings is location dependent – different break locations may have different sets of structures or components within the ZOI and therefore different coatings loads shall be calculated for the different breaks.

The total volume of coatings was calculated by identifying the coatings that exist within a given ZOI, then multiplying the surface area of those coatings by the thickness. Coating thickness depends on the type of substrate, substrate material, and coating type, as listed in Table B.3-1, "Coating Materials and Coating Thickness inside Containment," of APR1400-E-N-NR-140001-P. For the pipe breaks considered, the applicant assumed a coating thickness of 0.005 in. for IOZ, 0.005 inch for epoxy on steel substrates, and 0.025 inch for epoxy on concrete surfaces. These values correspond to the maximum values of the ranges listed in Table B.3-1. In its response to RAI 391-8462, Question 06.02.02-33, dated July 29, 2016 (ML16211A347), the applicant stated that these thickness ranges are determined by project coating specifications from the reference plants. Since using higher thickness values within the specified ranges is conservative with respect to the calculated amount of coating debris quantity, the staff finds the applicant's coating debris quantity acceptable.

The response to RAI 391-8462, Question 06.02.02-33, also provided the basis for the coating density values assumed in APR1400-E-N-NR-140001-P, Table 3.3-2, "Debris Properties." The values are based on Table 3.3, "Coating Debris Characteristics," of NEI 04-07, which was approved in the staff's SER to NEI 04-07. The response also proposed adding the reference for the density values to APR1400-E-N-NR-140001-P, Table 3.3-2. The staff confirmed that Revision 1 of APR1400-E-N-NR-140001, was revised to incorporate the proposed changes. Therefore, RAI 391-8462, Question 06.02.02-33 is resolved and closed.

Strainer testing performed by the applicant uses surrogates for epoxy and IOZ coatings. Silicon carbide, which has a density of 3.2 g/cm^3, was used as the surrogate. The material has a density somewhat higher than epoxy, but has a comparable characteristic particle size of 10 microns and has been used as a suitable surrogate in other testing reviewed by the staff. Based on the surrogate particle sizing and density, the staff determined that the applicant's surrogate material is considered acceptable, as it is comparable to the plant coating materials for the purpose of strainer head loss testing. Section 6.2.2.4.3.6 of this SER discusses the head loss test program and results.

In addition to the COL information items discussed above, COL applicants are also required to address the implementation of a containment coatings monitoring program, described further in Section 6.1.2 of this SER. This treatment is consistent with the effects on coatings described in Section 3.4.3.2 of NEI 04-07 and the associated SE and is therefore acceptable.

6.2.2.4.3.5 Debris Transport

Debris transport refers to analysis determining the fraction of the debris generated that reaches the strainers. In APR1400-E-N-NR-14001-P, the applicant cites guidance provided in Section 3.6.1, "Definition," of NEI 04-07, which considers four mechanisms for debris transport:

- Blowdown transport: the horizontal and vertical transport (to the containment floor) of debris throughout containment by the break jet.

- Washdown transport: the downward transport of debris due to containment spray flow and the postulated break flow.

- Pool fill transport: the horizontal transport of debris by break flow and containment spray flow to areas that may be active or inactive during recirculation.

- Pool recirculation transport: the horizontal transport of debris from the active portions of the containment pool to the suction strainers through flows induced by the operation of the SI and CSS in recirculation mode.

The applicant neglected washdown transport, as all debris that would have been transported by washdown is assumed to have been transported to the containment floor by blowdown. In the case of pool fill transport, the applicant conservatively assumed that all debris transported out of the SG compartments makes its way to the holdup volume tank, and no transport occurs to inactive volumes. Recirculation transport is assumed by the applicant to affect all fiber, coating and particulate debris.

RMI debris is broken down into two categories: large (75 percent of the debris) and small (25 percent). Larger pieces cannot reach the inside of the HVT due to the trash rack and screens at the entrance. Some smaller pieces may reach the HVT instead of settling out in containment on the path to the HVT, but because the terminal settling velocity is larger than the approach flow velocity in the sump, the applicant stated RMI debris that gets to the IRWST settles at the bottom and does not transport to the strainers.

In general, the staff finds RMI is unlikely to accumulate on a strainer configuration of the type used in APR1400 (strainer elevated slightly above pool elevation, not a "pit" type). Due to its porosity, RMI is likely to settle to the bottom of a pool, assuming low approach velocities[3]. Given the importance of the approach velocity, on June 11, 2015, the staff issued RAI 25-7844, Question 06.02.02-5, requesting the applicant to provide more information about how the approach velocity was calculated for the design. In its response to RAI 25-7844, Question 06.02.02-5 (ML15204A708), the applicant clarified how the approach velocity was calculated and identified that the sump approach velocity was 78 percent of the terminal velocity of fine RMI debris (the debris with the highest likelihood of impacting the strainer) and the conservatisms involved in calculating that velocity, including assuming a single SI and CS pump taking suction on a single strainer plus margin. Given there are four ECCS strainers, substantially lower flow rates would be expected. In addition, the applicant makes conservative assumptions regarding debris transport to the IRWST, when in fact, due to the previously mentioned pathway through the HVT, much of the debris is unlikely to make its way to the IRWST. As such, the staff finds that the approach velocities assumed are acceptable.

Other potential sources of nonfibrous transportable material in containment, such as signs, equipment tags, and other miscellaneous material are assumed by the applicant to transport to

[3] "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing," March 2008 (ML080230038).

the strainer and as such are allocated to a sacrificial area of the strainer assumed to be covered by these materials. This results in a conservatively assessed strainer area.

For the fiber, coating and particulate debris, the applicant conservatively assumes that all debris resulting from the break is transported to the strainer. Because making this assumption results in the highest head loss and potential fiber load downstream, the staff finds these assumptions acceptable for evaluating debris loads used for head loss and bypass testing.

6.2.2.4.3.6 Strainer Head Loss

The applicant conducted strainer head loss testing to demonstrate that the NPSH losses assumed in the SI and CSS pump evaluations are appropriate. Due to the plant specific nature of debris, strainer design and system configuration, the staff's evaluation of strainer head loss is based on the applicant's prototypic testing. Specifically, the staff evaluates strainer performance inputs to NPSH losses based on the results of the testing performed by the applicant.

During a postulated break inside containment, debris in various forms (insulation and coatings) discussed above may be generated by impacts resulting from jet forces. Further debris exists in the containment in small quantities prior to the postulated break. In addition, chemical precipitants may be created from the interaction of RCS coolant, containment buffering chemicals, and various plant materials. The chemical effects are discussed later in this report in Section 6.2.2.4.3.13. All of this debris is then transported to the HVT.

The HVT is lined with a trash rack at the entrance above the tank. The trash rack is a vertical structure with a maximum opening of 3.8 cm by 3.8 cm (1.5 in. by 1.5 in.). Once debris and coolant fluid enter the HVT, it will pass through a spillway structure located up the side of the tank such that some debris settling occurs. Then, the strainers in the IRWST further restrict debris from entering the SI and CSS systems. Debris built up on the strainer and further interacted with by other debris and chemical effects causes a pressure drop over the strainer greater than the clean strainer. The combination of the clean head loss plus the head loss resulting from maximum debris loading will not exceed the total strainer head loss assumptions that can be accommodated by the pumps.

DCD Tier 2, Section 6.8.4.5.5, "Debris Head Loss," of the DCD addresses the testing performed to determine the maximum head loss from debris, and APR1400-E-N-NR-14001-P describes the testing in detail. The applicant first performed a clean strainer head loss test to determine the pressure drop over the clean strainer under prototypic conditions.

The NRC staff conducted a regulatory audit from August 28 through September 9, 2013, to witness strainer testing performed to qualify the design used for the APR1400 (ML13353A401). The staffs witnessed clean strainer head loss testing, debris preparation, debris introduction and the ensuing head loss data collection, vortex testing, and pool drain down and strainer inspection. In addition, staff audited bypass testing, including filter bag preparation, installation and removal, debris preparation, debris introduction and transport, and data collection. The staff had concerns that the debris preparation was not prototypic of the latent fiber expected to be present in the APR1400 for the first test witnessed during the audit. These concerns were rectified in a subsequent test the staff witnessed and are reflected in the testing documented in APR1400-E-N-NR-14001-P.

Tests were performed for a clean strainer, with fiber only to determine a limiting bypass fraction, and with all transportable debris (that is, excluding RMI) to determine a limiting head loss. Head loss testing witnessed by the staff resulted in a pressure drop over the strainer of approximately 0.65 ft-water; subsequent testing by the applicant using more consistent debris preparation and a reduced strainer area to account for sacrificial area covered by miscellaneous debris resulted in a higher head loss of 0.81 ft-water. Based on the staff's audit observations and review of APR1400-E-N-NR-14001-P, use of this value for the strainer head loss term is consistent with the guidance in NEI 04-07 and the associated SE and is therefore acceptable.

6.2.2.4.3.7 Net Positive Suction Head

In order to ensure the ECCS pumps operate in an acceptable manner and to prevent cavitation from occurring inside the pump and potentially damaging the equipment, adequate net positive suction head (NPSH) is required. The available NPSH for the SI and CS pumps have to be greater than the required NPSH, accounting for uncertainty.

Available NPSH is calculated by adding the atmospheric pressure head on the fluid, the static head resulting from the difference in height between the fluid surface and the pump suction centerline and subtracting the head losses due to friction and line losses and the head equivalent to the vapor pressure of the fluid. In order to perform its design function, the available NPSH to the pump must be greater than the NPSH required by the pump. In accordance with the guidance in SECY-11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," the applicant applied an uncertainty of 21 percent to the vendor tested NPSH required. This uncertainty accounts for slight variations in pump speed and piping configurations in the real use cases (as compared with the testing) as well as differences in fluid properties and decreasing NPSH required with increasing temperature. As such, although the vendor-tested NPSH required for the CSS and SIPs is 14.4 and 18.23 ft of water (ft-water), respectively, the design basis NPSH values used are 17.5 ft-water for the CS pumps and 22.0 ft-water for the SIPs.

For the APR1400 design, the atmospheric head term and the vapor pressure loss term are linked. While the temperature is less than 100 °C (212 °F), the atmospheric head term is assumed to be the initial containment pressure prior to the LOCA, and the vapor pressure loss term is equivalent to the vapor pressure at the temperature of the IRWST. This is representative of the physical condition of the system, and NPSH is generally not a concern at these lower fluid temperatures. When the temperature exceeds 100 °C (212 °F), the atmospheric head term is assumed equal to the vapor pressure loss term. This results in the atmospheric head term being larger than the initial containment pressure at the start of the transient (approximately atmospheric pressure). This is referred to as crediting containment accident pressure for NPSH. As the pressure inside containment following an accident will be higher than the saturation pressure of the water in the IRWST, the staff finds this acceptable subject to the additional analysis provisions discussed below.

As directed by the SRP Section 6.2.2, if the containment accident pressure is credited in determining available NPSH, an evaluation of the contribution to plant risk from inadequate containment pressure should be made. No such evaluation was provided. Therefore, on June 11, 2015, the staff issued RAI 25-7844, Question 06.02.02-6 to address this issue (ML15162A004). In its response to RAI 25-7844, Question 06.02.02-6, dated May 19, 2016 (ML16141A053), the applicant provided additional detail on the uncertainty involved in the pump calculation and provided a risk assessment for the usage of containment accident pressure in the accident analyses.

Additionally, portions of the analysis that have been performed using methods the NRC staff has accepted (in this case, those parameters aside from the pump evaluation) do not require quantification of uncertainty, as the conservatisms inherent in the model through the use of limiting values are considered to compensate for the lack of explicit uncertainty.

In APR1400-E-N-NR-14001-P, Section 3.5.3.3, "Total Strainer Head Loss," the applicant calculates the total strainer head loss value as the clean strainer head loss plus debris head loss. This calculation is done by taking the analytical clean strainer head loss and adding the tested debris head loss, which, according to the applicant, incorporates a clean head loss and, thus, double counts the clean head loss. Performing the calculation in this fashion results in a head loss of 1.06 ft-water (0.25 ft-water plus 0.81 ft-water). Although the analytical clean head loss at 60 °C (140 °F) (0.25 ft-water) is less than the tested clean head loss at approximately 31.1 °C (88 °F) (0.52 ft-water), the applicant used an allowable value for head loss over the strainer of 2 ft-water, and use of either value for the clean head loss still results in a head loss of less than 2 ft-water. Thus, the staff finds that the use of 2 ft-water as the limiting head loss for the strainer is acceptable.

The static head term used by the applicant to calculate available NPSH is the difference between the height of the minimum water level in the IRWST following an accident and the center of the pump suction (different for the CS and SIPs). The staff reviewed the data supporting the minimum water level in the IRWST below in Section 6.2.2.4.3.8 of this SER. As part of an audit (ML15212A244) staff reviewed the calculated friction loss terms and confirmed the values are representative of the piping dimensions of the proposed design. In addition, to confirm the as-built NPSH (in order to account for any changes in piping runs, as well as other potential design changes conducted under 10 CFR 52.63(b)(2)) is greater than the requirements, the applicant is required to perform testing as part of ITAAC for the CS (DCD Tier 1, Table 2.11.2-4, "Containment Spray System ITAAC, item 10), SI (DCD Tier 1, Table 2.4.3-4, "Safety Injection System ITAAC," item 9.c) and SCPs (DCD Tier 1, Table 2.4.4-4, "Shutdown Cooling System ITAAC," item 9.d). The staff finds the design criteria for the loss terms, in conjunction with the ITAAC, are sufficient to provide reasonable assurance that the as-built design will have sufficient net positive suction head for the ECCS pumps.

6.2.2.4.3.8 Upstream Effects

In order for the IRWST to have sufficient inventory to provide flow to the spray system in the long term, sufficient inventory is required such that the long term water level in the IRWST does not fall below the minimum water level required by the SI and CS pumps to prevent cavitation. A sufficient water level is achieved through design considerations so that following the limiting postulated accident, the maximum amount of water holdup in locations other than the IRWST does not exceed the amount of water available inside containment. That is, the initial inventory of the IRWST plus the loss of fluid from the RCS is sufficient to maintain the IRWST water level above the minimum level required to provide NPSH to the pumps. The information on containment holdup is located in DCD Tier 2, Table 6.8-2.

To verify this information, the staff conducted a regulatory audit (ML17037A756). In the calculation supporting Table 6.8-2, the applicant conservatively calculates the total amount of dead-end volumes below the elevation of the HVT flow path as well as the other miscellaneous volumes that retain water (the containment atmosphere, film on various surfaces, the ECCS piping, etc.) under conditions following a postulated accident. The applicant's holdup analysis finds that the IRWST level is sufficient, given that: at the start of the transient, the RCS begins full; all compartments are flooded below the main HVT spillway; other surfaces are covered in condensate films; and one accumulator fails to open (and therefore its inventory is not contributed to the available water level). Given the above discussion, the applicant calculated a minimum water level in the IRWST of 4.75 feet above the bottom surface. The applicant used appropriately conservative assumptions, especially when considering that, for long term recirculation, pressure is expected to be somewhat lower than the calculated peak pressure during a design basis event (which occurs fairly early in the transient). A lower pressure would result in a lower partial pressure of steam (which means the actual water held up in containment would be lower) and a lower saturation pressure, which reduces many of the terms assumed in the analysis (thereby increases the available water inventory). Based on the information reviewed by the staff during the audit in conjunction with the information presented in the DCD, the staff finds the treatment of upstream holdup volumes acceptable, as the applicant has conservatively assessed the available volume of water in containment and adequate inventory remains for ECCS functions.

Structural analysis associated with upstream components is located below in Section 6.2.2.4.3.10.

6.2.2.4.3.9 Ex-Vessel Downstream Effects

The term “ex-vessel downstream effects” refers to effects of post-loss of coolant accident (post-LOCA) debris on the systems and components in the ECCS and CSS flow path (excluding the reactor vessel) located downstream of the IRWST sump strainer. Debris that bypasses the IRWST sump strainer may be carried downstream in the ECCS and CSS during post-LOCA operation, thus causing blockage or wear and abrasion in system components. Areas of concern for ex-vessel downstream components include (1) blockage of system flowpaths at narrow flow passages (e.g., containment spray nozzles, pump internal flow passages, and tight-clearance valves), and (2) wear and abrasion of surfaces (e.g., pump running surfaces) and heat exchanger tubes and orifices.

The ECCS removes heat from the reactor core following postulated DBAs. The function of the APR1400 ECCS is performed with the SIS. The CSS is designed to reduce containment pressure and temperature from a MSLB or a LOCA and remove fission products from the containment atmosphere following a LOCA. Both systems take suction from the IRWST through the sump strainer.

Design parameters for the SIS and its components are described in DCD Tier 2, Section 6.3, “Safety Injection System.” Design parameters for the CSS and its components are described in DCD Tier 2, Section 6.2.2. DCD Tier 2, Section 6.8.2.2.1, “In-Containment Refueling Water Storage Tank,” references Technical Report APR1400-E-N-NR-14001-P (Proprietary), “Design Features to Address GSI-191,” Revision 1, dated March 2017, to provide additional description of the APR1400 design features that limit the impact of post-accident debris accumulation on the SIS and CSS components.

The staff reviewed the applicant’s evaluation of ex-vessel downstream effects in Section 4.2, “Ex-Vessel Downstream Effects,” of Technical Report APR1400-E-N-NR-14001-P for conformance to RG 1.82, Revision 4, to provide reasonable assurance that the ECCS (SIS) and CSS and their associated components will function as designed under post-LOCA fluid conditions for the required mission time. Section 4.2 of Technical Report APR1400-E-N-NR-14001-P contains subsections that address the system descriptions, design inputs and evaluation assumptions, and evaluations of specific components (e.g., pumps, heat exchangers, valves). The following sections of this SER provide the staff’s evaluation of these specific topics.

6.2.2.4.3.9.1 Introduction

Technical Report APR1400-E-N-NR-14001-P, Section 4.2, "Introduction," states that the objective of ex-vessel downstream effects evaluation is to assess the systems and components of the APR1400 ECCS and the CSS to ensure that these systems are designed to be operable under post LOCA conditions.

On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-12 requesting the applicant define "guarantee," as used in the introductory paragraph, with respect to assessment of ECCS and CSS under post-LOCA conditions. In its response to RAI 63-7983, Question 06.02.02-12 dated August 10, 2015, the applicant stated that the technical report would be revised to change the word guarantee to ensure. The staff finds the applicant's response acceptable because the ex-vessel downstream effects evaluation provides reasonable assurance, and does not guarantee, that these systems are designed to be operable under post LOCA conditions. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-12. Therefore RAI 63-7983, Question 06.02.02-12 is resolved and closed.

6.2.2.4.3.9.2 LOCA Scenarios

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.2.1, "LOCA Scenarios," describes the most limiting large break LOCA (LBLOCA) as a double ended guillotine cold leg break at the discharge of the RCP. The most limiting small break LOCA (SBLOCA) is described as a DVI line break LOCA. For the downstream ex-vessel effects evaluation, the SBLOCA is bounded by the LBLOCA. The debris quantity and the ECCS flows during the SBLOCA are considered much smaller than during the LBLOCA. Therefore, the bounding LOCA scenario for the ex-vessel downstream effects evaluation is the LBLOCA.

6.2.2.4.3.9.3 Mission Time

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.2.2, "Mission Time," defines mission time as the maximum period of time for which a system, structure or component remains to perform their safety function. The mission time is the accident analysis credit time. For the downstream ex-vessel effects evaluation, the mission time is defined as 30 days.

6.2.2.4.3.9.4 Components of Interest

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.2.3, "Components of Interest," states that Table 4.2-1, "Components in the Flow Path during an LBLOCA," of Technical Report APR1400-E-N-NR-14001-P lists the ECCS and CSS components in the downstream effects evaluation. The components in the ECCS and CSS flow path during SBLOCA and LBLOCA operations include pumps, heat exchangers, valves, orifices, containment spray nozzles and piping.

The staff's review noted that Table 4.2-1 in Technical Report APR1400-E-N--NR-14001-P, Section 4.2.2.3, does not include CSS miniflow heat exchanger or the miniflow piping. On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-13 requesting the applicant review and confirm that Table 4.2-1 includes all applicable components in the downstream ex-vessel effects evaluation. In its response to RAI 63-7983, Question 06.02.02-13 dated August 10, 2015, the applicant provided a markup of proposed changes to Table 4.2-1 and stated that Table 4.2-1 will be revised to either add, change, or delete the components required to be included in the downstream effects evaluation and that Technical Report Section 4.2.2.3

will be revised consistent with Table 4.2-1. The staff reviewed the applicant's response and finds the components of interest acceptable because the markup in Table 4.2-1 includes the mini flow components and other applicable changes for components to be included in the ex-vessel downstream effects evaluation. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-13. Therefore RAI 63-7983, Question 06.02.02-13 is resolved and closed.

6.2.2.4.3.9.5 Post-LOCA Fluid Constituents

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.2.4, "Post-LOCA Fluid Constituents," describes the debris generated during a LBLOCA and categorizes the debris according to type and quantity in Table 4.2-3, "Total Quantity of Debris Generated during an LBLOCA." The type of debris generated during an LBLOCA consists of the following:

- Reflective metal insulation (RMI): Consists of Fines and Large Pieces
- Qualified epoxy coating: Consists of Particulates
- Latent debris
    - Particulate (dirt and dust): Consists of Particulates
    - Fiber: Consists of Fines
- Miscellaneous debris (e.g., tape, tags, stickers, labels, rope, fire hoses, ventilation filters, plastic sheeting): Consists of Large Pieces

Table 4.2-2, "Size Range of Debris Material" categorizes the size of debris material as follows:

- Particulates: 0-2.38 millimeter [mm] (0-0.094 inch)
- Fines: less than 101.6 mm (4 inch)
- Large pieces: greater than 101.6 mm (4 inch)

The applicant evaluated the debris generated during a LBLOCA and determined that debris less than or equal to the perforated plate hole diameter of the sump screen (2.38 mm [0.094 inch]) will bypass the sump strainer. The evaluation for each type of debris is described below.

Epoxy coating and latent particulate evaluation: Based on the hole diameter of the sump screen, the ECCS will ingest 100 percent of the qualified epoxy coating and latent particulate because the maximum size of these materials is 2.38 mm (0.094 inch).

Latent fiber evaluation: Bypass testing of latent debris yielded a fiber bypass percentage of less than 25 percent as described in Appendix D, "Bypass Test Report for the IRWST Sump Strainer," of Technical Report APR1400-E-N-NR-14001-P. However, for downstream ex-vessel effects evaluation, the applicant conservatively assumes that 100 percent of the latent fiber bypasses the sump strainer.

RMI evaluation: For RMI, the applicant stated the following:

Results of the NRC debris generation test documented in NUREG/CR-6808, “Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump performance,” show that RMI debris size distribution ranges from 6.35 mm (0.25 inch) to 152.4 mm (6 inch). RMI debris will not bypass the sump screens and enter the ECCS because the size of the RMI debris is greater than the perforated plate hole size in the sump strainer. As a result, this evaluation assumes no RMI bypasses through the sump strainer.

The staffs notes that NUGRE/CR-6808 states the following about RMI testing:

Section 3.2.2.4 of NUREG/CR-6808, “Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance,” provides results of jet impact testing on RMI. NUREG/CR-6808 Figure 3-7, “Typical RMI Debris Generated by Large Pipe Break,” shows a size distribution at ¼ inch of 4.3 percent.

Reference 3-4 of NUREG/CR-6808, “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Bulletin 96-03 Boiling Water Reactor Owners Group Topical Report NEDO-32686, ‘Utility Resolution Guidance for ECCS Suction Strainer Blockage,’” Docket No. PROJ0691, August 20, 1998 (ML092530505) provides more detailed information in Appendix D of NEDO-32686 (pdf file pages 366 to 399), “Structural Properties of Reflective Metal Insulation Installed in U.S. BWR’s.”

NUREG/CR-6808 Figure 3-7, Typical RMI Debris Generated by Large Pipe Break, can be found in NEDO-32686 Appendix D as Figure 6, Size vs. Percent by Weight. NEDO-32686 Appendix D test data analysis identifies Vattenfall/NRC test as a bounding case. Appendix D shows the size of the debris and states ML092530505, pdf file page 387 of 399):

“Photographs of a random sample of each size category are reproduced as Figure 7. Note that in the smallest sample size range, 0.25,” debris smaller than 0.25” is not observed.”

Based on the results of jet impact testing that no RMI debris smaller than 0.25 inches was observed, there is reasonable assurance that RMI will not enter the ECCS because the size of the RMI debris is greater than the diameter of the IRWST sump screen (2.38 mm [0.094 inch]). Therefore, the staff considers the applicant’s evaluation that RMI will not bypass the sump strainer to be acceptable.

On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-14 requesting the applicant describe how the RMI testing referenced in NUREG/CR-6808 is applicable to the APR1400 reactor plant. In its response to RAI 63-7983, Question 06.02.02-14 dated January 7, 2016 and April 25, 2016, the applicant stated that the APR1400 has similar NSSS layout and plant operating conditions as US PWRs and that the DCD will be revised to add a COL item to confirm that RMI used in the APR1400 is similar to that tested in NUREG/CR-6808. The staff reviewed the applicant’s response and finds it acceptable because the configuration and controls established (COL 6.8(4)) ensure that the plant will respond consistent with tests conducted in NUREG/CR-6808 that achieve acceptable post LOCA results. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-14. Therefore, RAI 63-7983, Question 06.02.02-14 is resolved and closed.

Miscellaneous debris evaluation: Miscellaneous debris materials are large pieces of material with a debris size range that is significantly greater than the perforated hole size in the IRWST sump strainer. Therefore, due to their size, the applicant concludes that these materials will not bypass the IRWST sump strainer.

Technical Report APR1400-E-N-NR-14001-P, Table 4.2-5, "Post-LOCA Fluid Constituents Downstream of IRWST Sump Strainer," lists the debris type, quantity, density, and concentration in parts per million (ppm) that will bypass the sump strainer and enter the ECCS. The debris listed in Table 4.2-5 is used in evaluating the systems and components of the ECCS and CSS to provide reasonable assurance that these systems and components are designed to be operable under post-LOCA conditions for the required mission time.

The applicant's methodology to evaluate debris in the post-LOCA fluid constituents is acceptable because it specifies the type and quantity of debris that will bypass the sump strainer and is consistent with the staff SE for WCAP-16406, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191,"[4]

6.2.2.4.3.9.6 ECCS Flow Rate and Flow Velocity

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.2.5, "ECCS Flow Rate and Flow Velocity," describes the ECCS flow rates used for the evaluation of debris settling and component wear during a LBLOCA. The APR1400 is a fixed resistance system under valve wide-open conditions. Emergency Operating Procedures allow for operator action to throttle flow based on main control room (MCR) indication. The range of operation is therefore assumed to be from shutoff head conditions to runout conditions. To evaluate debris settling and component wear during an LBLOCA, the evaluation conservatively assumes ECCS and CSS flow rates ranging from shutoff head conditions to runout conditions.

The SIP flow is assumed to be 303 liters per minute [L/min] (80 gallons per minute [gpm]) for evaluating debris settling in the SIS. The SIP flow is assumed to be 6,057 L/min (1,600 gpm) for component wear rate evaluations. The engineering design range of flow the SIP is 397 L/min (105 gpm) at shutoff and 4,675 L/min (1,235 gpm) at runout.

The CS pump flow is assumed to be 1,514 L/min (400 gpm) for evaluating debris settling in the CSS. The CS pump flow is assumed to be 27,255 L/min (7,200 gpm) for component wear rate evaluations. The engineering design range of flow is 1,817 L/min (480 gpm) at shutoff and 24,605 L/min (6,500 gpm) at runout. The component wear rate and debris settling evaluations are detailed in Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.

Staff considers the applicant's approach to use ECCS and CSS flow rates that are more conservative than the shutoff head conditions and runout conditions to be acceptable for determining component wear rate and debris settling evaluations.

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.2.5 states that shutoff head conditions for SI and CS pumps will be verified by the vendor during procurement. However, the technical report does not address whether runout conditions for SI and CS pumps will be verified by the vendor during procurement. On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-15, requesting that the applicant address runout testing for SI and CS pumps. In its response to RAI 63-7983, Question 06.02.02-15 dated August 10, 2015, the applicant stated that technical report Section 4.2.2.5 would be revised to specify that both

[4] "Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report (TR) WCAP-16406-P-A, Revision 1, 'Evaluation of Downstream Sump Debris Effects in Support of GSI-191,' Pressurized Water Reactor Owners Group, Project No. 694," NRC, Washington, DC, December 2007. (ML073520295)

shutoff head and runout conditions for the SI and CS pumps will be verified by the vendor. The staff reviewed the applicant's response and finds it acceptable because the technical report states that both shutoff head and runout conditions for the SI and CS pumps will be verified by the vendor. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-15. Therefore RAI 63-7983, Question 06.02.02-15 is resolved and closed.

The applicant's methodology to determine ECCS and CSS flow rates and flow velocities used for the downstream ex-vessel evaluation is conservative with respect to the shutoff and runout pump flow conditions and is therefore acceptable.

6.2.2.4.3.9.7 Summary of Assumptions and Conservatisms

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.2.6, "Summary of Assumptions and Conservatisms," describes the assumptions and conservatisms used in the ex-vessel downstream evaluation. These assumptions and conservatisms are summarized as follows:

- 100 percent of all particulates (i.e., coating debris and latent particulates) and 100 percent of latent fiber are assumed to pass through the strainers and enter into the ECCS and CSS. RMI does not pass through the sump strainer because the size of the RMI debris is greater than the sump strainer perforated plate hole size.

- SIP flow is assumed to be 303 L/min (80 gpm) for the purpose of calculating settling velocities. Flow is assumed to be 6,057 L/min (1,600 gpm) for the purpose of component wear rate evaluations. Engineering design range of flow is 397 L/min (105 gpm) at shutoff and 4,675 L/min (1,235 gpm) at runout.

- CS pump flow is assumed to be 1,514 L/min (400 gpm) for the purposes of calculating settling velocities. Flow is assumed to be 27,255 L/min (7,200 gpm) for the purpose of component wear rate evaluations. Engineering design range of flow is 1,817 L/min (480 gpm) at shutoff and 24,605 L/min (6,500 gpm) at runout.

- Wear is calculated from "time zero," i.e. start of the event. Worst case fluid properties are assumed to be present. This assumption is conservative since it does not credit debris transport or the slow increase of fluid properties due to long term mixing.

- Fluid velocity through a single CS heat exchanger tube is assumed to be 4.57 meters per second [m/s] (15 feet per second [ft/s]). A nominal design and operating heat exchanger velocity range is 0.91 to 3.05 m/s (3 to 10 ft/s). Therefore, the use of 4.57 m/s (15 ft/s) is conservative from a heat exchanger design perspective and bounds the heat exchanger design and procurement specifications.

Technical Report APR1400-E-N-NR-14001-P, Table 4.2-5 lists the amount of debris in the post-LOCA fluid (downstream of the IRWST sump strainer) that will be used for confirmatory tests.

The concentration of the post-LOCA fluid constituents listed in Technical Report APR1400-E-N-NR-14001-P, Table 4.2-5, is conservatively estimated based on the assumption that the IRWST contains 250,000 gallons (946.4 m3) of water during post-LOCA operation, which is less than the minimum IRWST water volume of 262,388 gallons (993.2 m^3). Estimating the debris concentration at less than the expected IRWST volume yields a more concentrated debris-laden fluid for confirmatory tests, and produces conservative test results. On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-16, requesting the applicant describe the methodology used to calculate recirculation water volume for determining the concentration (ppm) of post-LOCA fluids as listed in Table 4.2-5 of the technical report. In its response to RAI 63-7983, Question 06.02.02.-16 dated April 25, 2016, the applicant provided additional information describing the methodology and calculations used to determine the recirculation water volume during post-LOCA operation. Based on the calculations in the RAI response, the applicant determined that the IRWST water volume of 250,000 gallons (946.4 m^3) results in a higher concentration of debris than was previously listed in Table 4.2-5 of the technical report and proposed to revise Table 4.2-5 of the technical report with the higher debris concentration values. The staff determined that the applicant's methodology and calculations to determine water volume and the concentration of the post-LOCA fluid constituents is consistent with the methodology approved in staff SE on TR-WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects In Support Of GSI [Generic Safety Issue]-191," and is acceptable. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-16. Therefore RAI 63-7983, Question 06.02.02-16 is resolved and closed.

The applicant's methodology to determine assumptions and conservatisms used in the ex-vessel downstream evaluation is acceptable because the assumptions are conservative for evaluation effect of debris on components and is consistent with the methodologies approved in staff SE on TR-WCAP-16406-P.

6.2.2.4.3.9.8 SI and CS Pump Evaluations

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.1, states that the SI and CS pumps and associated mechanical seals will be qualified to operate with the post-LOCA fluids for at least 30 days, *using the guidance of* ASME Standard QME-1-2007 as endorsed by RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 3. As part of the ASME Standard QME-1-2007 qualification process, Section 4.2.3.1 also lists specific pump qualification criteria for post-LOCA operation that will be evaluated by the vendor.

The post-LOCA fluids used for the design and qualification of pumps includes the debris that bypasses the containment sump strainer identified in Table 4.2-5 of Technical Report APR1400-E-N-NR-14001-P.

RG 1.100, Revision 3 states that ASME QME-1-2007 is an NRC staff acceptable methodology for the qualification of pumps and valves, and when a licensee commits to the use of QME-1-2007 for the qualification of pumps and valves, the criteria and procedures become part of the basis for the qualification program. Therefore, staff does not consider the statement "using the guidance of," ASME QME-1-2007 to be acceptable. Therefore, on July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-17, requesting the applicant to specify that qualification of pumps and valves will be in accordance with ASME QME-1-2007. In its response to RAI 63-7983, Question 06.02.02-17 dated August 10, 2015, the applicant stated that the technical report would be revised to state that qualification of pumps and valves will be in accordance with ASME QME-1-2007. The staff reviewed that applicant's response and finds it acceptable because it is consistent the staff approved methodology in RG 1.100 for the qualification of pumps and valves. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-17. Therefore RAI 63-7983, Question 06.02.02-17 is resolved and closed.

Section 4.2.3.1 of the technical report lists specific pump qualification criteria for post-LOCA operation that will be evaluated by the vendor and states that qualification will be performed using tests and/or analysis. However, to be consistent with ASME QME-1-2007 as accepted by RG 1.100, Revision 3, pump qualification is demonstrated by test or a combination of test and analysis. Analysis alone is not an acceptable method for qualification. Therefore, on July 7, 2015, the staff issued to RAI 63-7983, Question 06.02.02-18, requesting the applicant specify qualification of pumps by test or a combination of test and analysis. In its response to RAI 63-7983, Question 06.02.02-18, dated August 10, 2015, and July 28, 2017, the applicant stated the technical report would be revised to specify that the SI and CS pumps and associated mechanical seals will be qualified to operate with post-LOCA fluids for at least 30 days in accordance with ASME QME-1-2007 as endorsed by RG 1.100 Revision 3. The staff finds the applicant's response acceptable because it is consistent with the staff approved methodology in ASME QME-1-2007 for the qualification of pumps and does not specify the qualification by analysis alone. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 2, was revised as committed in the response to RAI 63-7983, Question 06.02.02-18. Therefore RAI 63-7983, Question 06.02.02-18 is resolved and closed.

The applicant's methodology to evaluate the SI and CS pumps and associated mechanical seals to operate with the post-LOCA fluids is acceptable because the pumps are qualified in accordance with staff approved ASME QME-1-2007 as endorsed by RG 1.00, Revision 3. The SI and CS pump evaluation meet the regulatory requirements of GDC-4.

6.2.2.4.3.9.9 Heat Exchanger Evaluation

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.2, "Heat Exchanger Evaluation," describes the CS heat exchanger evaluations and states that heat exchanger plugging, fouling, and wear are addressed as part the equipment specification. The staff issued RAI 63-7983, Question 06.02.02-20 requesting the applicant describe the methodology for evaluation of the heat exchanger plugging, fouling, and wear as part the equipment specification. In response to RAI 63-7983, Question 06.02.02-20 dated November 4, 2015, the applicant stated the technical report would be revised to describe that heat exchanger plugging, fouling, wear and heat transfer performance in the presence of post-LOCA debris will be evaluated by the vendor during the procurement process with a certificate of compliance to provide verification that the heat exchanger meets procurement specifications. The staff finds the applicant's response acceptable because the CS heat exchanger plugging, fouling, and wear will be evaluated by the vendor during the procurement process and a certificate of compliance will be provided documenting that the heat exchanger meets the procurement specifications. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-20. Therefore RAI 63-7983, Question 06.02.02-20 is resolved and closed.

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.2.1, "Heat Exchanger Plugging," states that CS heat exchanger tube inside diameter is significantly larger than the largest expected particle size. Therefore, the report concludes that plugging or blockage will not occur. The report also states that debris settling will not occur because the CS heat exchanger flow velocity is significantly greater than the terminal settling velocity of the debris as specified in Technical Report Table 4.2-4, "Terminal Settling Velocity of Debris Source Materials." However, the heat exchanger flow velocity is not specified in the technical report. On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-19 requesting the applicant specify the CS heat exchanger flow velocity. In its response to RAI 63-7983, Question 06.02.02-19, dated August 10, 2015, the applicant proposed to revise the technical report to state that the heat exchanger is designed with a tube flow velocity not to be less than 3 ft/s to prevent deposition of suspended material in transition areas of heat exchangers, piping, etc. The staff finds the applicant's response acceptable because it specifies that the CS heat exchanger and CS miniflow heat exchanger flow velocity is greater than the terminal settling velocity of the debris and that debris settling will not occur. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-19. Therefore RAI 63-7983, Question 06.02.02-19 is resolved and closed.

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.2.2, "Heat Exchanger Performance and Wear," states that the CS heat exchanger and CS miniflow heat exchanger are sized and designed to maximize heat transfer efficiency and performance. Chemical effects of precipitates during the 30 day post-LOCA mission time will not degrade the performance of the heat exchangers as discussed in the Chemical Effects Section of the SE. The applicant also stated that any potential reduction in capability due to fouling over the 30 day mission time is gradual and within the nominal heat exchanger design.

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.2.2, states that the CS heat exchanger tubes are constructed of 304 stainless steel that are appropriate for use as heat exchanger material in mildly abrasive applications. Heat exchanger tube wear was calculated using a conservative mass concentration of debris during the 30 day mission time and the tubes were found to have sufficient thickness to withstand the effects of erosion due to the effects of debris particles.

The applicant's heat exchanger evaluation addressed the CS heat exchangers but did not specifically address the CS mini flow heat exchangers. On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-31, requesting the applicant address the effects of debris on the CS mini flow heat exchanger. In its response to RAI 63-7983, Question 06.02.02-31, dated January 18, 2016, the applicant proposed to revise the technical report to address the effects of debris on the CS mini flow heat exchanger. The NRC staff finds the applicant's response acceptable because the proposed technical report revision adequately addresses the effects of debris on the CS mini flow heat exchanger. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-31. Therefore RAI 63-7983, Question 06.02.02-31 is resolved and closed.

The applicant's methodology to evaluate heat exchanger plugging, fouling, wear and heat transfer performance in the presence of post-LOCA debris is acceptable because the effect of debris on the heat exchanger is consistent with the methodologies approved by the staff in the SE for TR-WCAP-16406-P. Therefore, the staff finds that the heat exchanger evaluation meets the regulatory requirements of GDC-4.

6.2.2.4.3.9.10 Blockage Evaluation for Valves, Orifices, Spray Nozzles, and Pipes

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.3.1, "Blockage and Debris Settling Evaluation for Valves, Orifices, Spray Nozzles, and Pipes," describes the evaluation for blockage for valves, orifices, spray nozzles, and pipes during operation with post-LOCA fluids. The applicant stated the following:

The IRWST strainer hole size is 2.38 mm (0.094 inch). Therefore, when the gap of the component is 2.38 mm (0.094 inch) + 0.238 (0.0094 inch) or 2.62 mm (0.103 inch) or less than this value, the flow-path or component may be blocked.

The flow diameters of valves, orifices, spray nozzles, and pipes are significantly larger than the maximum debris size. Gate valves are either full open or full closed. Globe valves are expected to be throttled to a minimum of 50.8 mm (2 inch) open between the valve disc and seat. Therefore, blockage is not expected for valves, orifices, and pipes during operation with post-LOCA fluids.

The applicant's evaluation for blockage for valves, orifices, spray nozzles, and pipes during operation with post-LOCA fluids is acceptable because the flow diameters are significantly larger than the maximum debris size. This evaluation is consistent with the methodology approved in the SE for TR-WCAP-16406-P.

6.2.2.4.3.9.11 Debris Settling Evaluation for Valves, Orifices, Spray Nozzles, and Pipes

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.3.1, "Blockage and Debris Settling Evaluation for Valves, Orifices and Pipes," evaluates debris settling for valves, orifices, pipes, and spray nozzles. The velocity of the debris in the post-LOCA fluid is assumed equal to the velocity of the fluid. If the fluid velocity is greater than the terminal settling velocity of the debris, the debris will not settle. APR1400-E-N-NR-14001-P, Table 4.2-6, "Affected Equipment /Flow Rates," lists the size, flow rate, and maximum settling velocity for the orifices, spray nozzles, and piping in the downstream ex-vessel effects evaluation. The applicant compared the pump shutoff flow velocities with the maximum settling velocity of 0.70 ft/sec for latent debris as calculated in Table 4.2-4 and determined that flow velocities are above the settling velocities in all components except several cases (24 inch, 20 inch, and 10 inch SIP suctions lines, and 12 in. SIP discharge line).

The applicant did not provide technical justification that debris settling would not occur in piping and valves where the flow velocity for latent debris is less than the terminal settling velocity. On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-22, and supplemental RAI 543-8734, Question 06.02.02-46, requesting the applicant address the potential for debris settling in piping where the flow velocity is less than the settling velocity. In its response to RAI 63-7983, Question 06.02.02-22, dated April 25, 2016, and RAI 543-8734, Question 06.02.02-46, dated July 11, 2017, the applicant stated that settling velocities in these locations were conservatively calculated using the pump shutoff flow. The system flow velocity based on the expected pump operation is higher than the conservative flowrate at pump shutoff. The applicant recalculated the system flow velocity based on the expected pump operation during the event in lieu of the pump shutoff flow and determined that debris settling would not occur at the expected pump flowrates. The applicant provided a proposed markup for TR APR1400-E-N-NR-14001-P Section 4.2.3.3.1 and Table 4.2-9 describing that debris settling will not occur when calculated using the expected pump flow. The staff finds the applicant's response acceptable because calculating the debris settling using the expected pump flow is more realistic of actual conditions and results in flow higher than the terminal setting velocity of 0.70 ft/sec for the debris. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 2, was revised as committed in the response to RAI 543-8734, Question 06.02.02-46. Therefore RAI 543-8734, Question 06.02.02-46 is resolved and closed.

6.2.2.4.3.9.12 Wear Rate Evaluation for Valves, Orifices, and Pipes

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.3.2, "Wear Rate Evaluation for Valves, Orifices and Pipes," evaluates the wear rate of valves, pipes and orifices due to the effect of debris. Erosive wear is caused by particles that impinge on a component surface and removes material from the surface because of momentum effects. The wear rate of a material depends on the debris type, debris concentration, material hardness, flow velocity, and valve position. The report states that flow rates of 6,057 L/min (1,600 gpm) and 27,255 L/min (7,200gpm) for SIS and CSS, respectively, are conservatively assumed for the wear rate evaluation of the components listed in Table 4.2-1 of the technical report.

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.3.2, “Wear Rate Evaluation for Valves, Orifices and Pipes,” states that the wear rate of ECCS valves will be provided by the vendor. The vendor will qualify the ECCS valves to operate with the post-LOCA fluids for at least 30 days, using the qualification in accordance with ASME QME-1-2007, endorsed by RG 1.100, Revision 3. As part of the qualification process, the wear evaluation for the ECCS valves in the flow path during an accident is performed. However, Section 4.2.3.3.2 addresses ECCS valve qualification and wear rate but did not address qualification and wear rate evaluation of CSS valves. Therefore, on July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-25, requesting the applicant to address qualification and wear rate evaluation for CSS valves during operation in post-LOCA fluids. In its response to RAI 63-7983, Question 06.02.02-25, dated August 10, 2015, the applicant proposed to revise technical report Section 4.2.3.3.2, to address qualification in accordance with ASME QME-1-2007, and wear rate evaluation for CSS valves similar to the qualification and wear rate evaluation for ECCS valves. The staff finds the applicant’s response acceptable because it specifies that both ECCS and CSS valves are qualified in accordance with ASME QME-1-2007 as endorsed by RG1.100, Revision 3 and as part of the qualification process, the wear evaluation for the ECCS and CSS valves in the flow path during an accident is performed. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-25. Therefore RAI 63-7983, Question 06.02.02-25 is resolved and closed.

Technical Report APR1400-E-N-NR-14001-P, Table 4.2-7, contains the piping, spray nozzle, and orifice wear calculation results and lists the diametric wear and percent flow rate increase for pipe and orifice. The report also states that an analysis will be provided to confirm that the overall system resistance/pressure drop across the ECCS is consistent with the safety analysis results for the 30-day mission time. On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02-24, requesting the applicant describe the analysis that will confirm that the overall system resistance/pressure drop across the ECCS and CSS is consistent with the safety analysis results for the 30-day mission time, including the verification of acceptable ECCS and CSS operation. In its response to RAI 63-7983, Question 06.02.02-24, dated March 16, 2017, the applicant proposed to add Section 4.2.4, “Overall System Analysis,” to Technical Report APR1400-E-N-NR-14001-P. This additional section evaluates potential increase of flowrates in ECCS and CSS due to component wear and concludes that the resulting flows and pressures are consistent or conservative with respect to the inputs used in the accident analysis. The staff finds the applicant’s response acceptable because it specifies that the resulting flows and pressures in ECCS and CSS due to component wear are consistent or conservative with respect to the inputs used in the accident analysis. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 2, was revised as committed in the response to RAI 63-7983, Question 06.02.02-24. Therefore RAI 63-7983, Question 06.02.02-24 is resolved and closed.

The applicant’s methodology to evaluate the wear rate of components during post-LOCA debris conditions is acceptable because it is consistent with the methodologies approved by the staff in the SE for TR-WCAP-16406-P for evaluation of component wear and the COL applicant will verify that any increased flowrates in ECCS and CSS due to component wear are within the maximum allowable flowrates for at least 30 days of post-LOCA operation.

6.2.2.4.3.9.13 Instrument Tubing Clogging Evaluation

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.4, "Instrument Tubing Clogging Evaluation," states that when the instrument tubing lines maintain a solid state prior to ECCS operation, it is determined that tubing integrity is not affected because there is almost no possibility of debris ingestion, and the evaluation shows there are no effects from flow blockage and wear because flow velocities in all cases are above the settling velocities of the post-LOCA fluid. Section 4.2.3.4 also states that all instrument connections are located at the horizontal or above. On July 7, 2015, the staff issued RAI 63-7983, Question 06.02.02.27, requesting the applicant further describe the instrument connections located at horizontal or above. In its response to RAI 63-7983, Question 06.02.02.27, dated August 10, 2015, the applicant stated that all instrument connections used in the APR1400 reactor are located at horizontal or above, meaning at the side or at the top of the pipe and there are no bottom-mounted instrument connections in the SIS and CSS. The technical report will be revised to state that all instrument connections are at the side or at the top of the pipe and the SIS and CSS do not contain any bottom-mounted instrument connections. The staff finds the applicant's response acceptable because all instrument connections are at the side or at the top of the pipe. This is consistent with the staff SE for WCAP-16406. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-27. Therefore RAI 63-7983, Question 06.02.02-27 is resolved and closed.

The applicant's evaluation for instrument tubing is acceptable because the tubing is in a water solid state prior to ECCS operation and all instrument connections are at the side or at the top of the pipe. This is consistent with the methodologies approved by the staff in the SE for TR-WCAP-16406-P.

6.2.2.4.3.9.14 Chemical Effects

Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.5, "Chemical Effects Evaluation," describes the effect of chemical precipitates on the performance of pumps (including mechanical seals), valves, heat exchangers, orifices, and piping in the ECCS and CSS downstream of the sump strainer during the 30 day post-LOCA mission time and concludes that chemical precipitates will not degrade the performance of the ECCS.

In addition, Technical Report APR1400-E-N-NR-14001-P, Section 4.2.3.5, states that the qualification of the ECCS and CSS pumps, performed with conservative amounts of post-LOCA debris in accordance with ASME QME-1-2007, will include confirmation that the internal running clearance of the ECCS pumps is sufficiently large enough to avoid clogging, and supports acceptable pump and seal operation during the 30-day post-LOCA mission time.

However, Section 4.2.3.5 addresses ECCS components for the chemical effects evaluation but does not address CSS components. Therefore, the staff issued RAI 63-7983, Question 06.02.02-29 requesting the applicant address the chemical effects evaluation for CSS components. In its response to RAI 63-7983, Question 06.02.02-29, dated November 4, 2015, the applicant proposed to revise the technical report Section 4.2.3.5 to address the chemical effects evaluation for CSS components. The staff finds the applicant's response acceptable because it specifies the chemical effects evaluation for CSS components. The staff confirmed that Technical Report APR1400-E-N-NR-14001-P, Revision 1, dated March 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-29. Therefore RAI 63-7983, Question 06.02.02-29 is resolved and closed.

The applicant's evaluation that precipitants have no effect on plugging or wear of downstream ex-vessel components is acceptable because it is consistent with staff positions documented in NRC Memorandum, "Basis for Excluding Chemical Effects Phenomenon from WCAP-16406-P Ex-vessel Downstream Evaluations," dated January 21, 2010 (ML093160100), and Technical Report, "Evaluation of Chemical Effects Phenomena Identification and Ranking Table Results," dated March 2011 (ML102280594).

6.2.2.4.3.9.15 ITAAC

The staff reviewed APR1400 DCD Tier 1 Tables to verify that functional qualification ITAAC for pumps and valves in the downstream ex-vessel effects evaluation are included in the tables. The staff identified that APR1400 DCD Tier 1, Table 2.11.2-4, "Containment Spray System," does not contain ITAAC for the functional qualification of the CSS pumps. By RAI 63-7983, Question 06.02.02.30 the staff requested the applicant add the ITAAC for the CSS pumps in Table 2.11.2-4. In its response to RAI 63-7983, Question 06.02.02-30, dated August 10, 2015, the applicant stated that APR1400 DCD Tier 1, Table 2.11.2-4, would be revised to add an ITAAC for the functional qualification of the CSS pumps. The staff finds the applicant's response acceptable because the proposed revision adds ITAAC to confirm the functional qualification of the CSS pumps. SER Section 3.9.6 evaluates the functional qualification ITAAC for pump and valves. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 63-7983, Question 06.02.02-30. Therefore, RAI 63-7983, Question 06.02.02-30 is resolved and closed.

6.2.2.4.3.9.16 COL Item

DCD Tier 2, Section 6.8.6, "Combined License Information," specifies action items for COL applicants related to generic APR1400 effect of post-LOCA debris on components, as indicated in the following table.

Item No.	**Description**	**DCD Tier 2 Section**
COL 6.8(4)	The COL applicant is to confirm that the RMI is one of the tested RMIs in NUREG/CR-6808.	6.8.4.5.3
COL 6.8(7)	The COL applicant is to confirm that the IRWST sump strainer has the total strainer head loss less than the allowable head loss (0.61 m (2ft)).	6.8.4.5.5

6.2.2.4.3.9.17 Summary

The NRC staff concludes that the provisions in the DCD and technical report for the ex-vessel downstream effects in the APR1400 design are acceptable and meet the applicable NRC requirements and guidance. This conclusion is based on the applicant having specified provisions in the DCD and technical report that the ECCS and CSS and their associated components will function as designed under post-LOCA fluid conditions for the required mission time.

6.2.2.4.3.10 Strainer Structural Integrity

In technical report APR1400-E-N-NR-14001-P, Appendix A, "Comparison of IRWST Sump Strainer Design to NRC RG 1.82 Requirements," the applicant presents that the strainers are installed away from the spargers to minimize the effect of hydrodynamic loads induced by the discharge of water, air, and single and two-phase steam due to the opening of the pressurizer POSRVs into the IRWST. Strainers are designed to seismic Category I and quality Class G. The staff noted that the strainers are installed away from the spargers, such that the effect of hydrodynamic loads on the strainers, if any, will be minimized. Since the strainers are submerged in the water, sump strainer submergence is adequate to preclude vortexing, sump fluid flashing, and deaeration induced by excessive differential pressure drop, and the sloshing and weight loading have a minimal effect to the strainer structure. The structural analysis of the strainer is performed to verify the structural adequacy of the sump strainer, including seismic, differential pressure, and hydrodynamic loads. This set of loading categories is consistent with those applied to ASME BPV Code Class 1, 2, and 3 components in DCD Tier 2, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Class CS Core Support Structures," and is therefore acceptable to the staff.

DCD Tier 2, Section 3.2.2, "System Quality Group Classification," Table 3.2-1, indicates that the IRWST sump strainers are designed to ANSI/AISC N690-12, "Specification for Safety-Related Steel Structures for Nuclear Facilities," seismic Category I. The staff finds that the code of construction for the strainer design is consistent with the design of other containment internal structures and is therefore acceptable.

In technical report APR1400-E-N-NR-14001-P, Section 3.5.3.3, "Total Strainer Head Loss," the applicant presents that the total strainer head loss value is the sum of the clean strainer head loss and debris head loss. According to the applicant, this total strainer head loss is conservatively calculated by double-counting the clean screen component using the analytical value and the measured value of debris head loss. Therefore, 7.62 cm-water (0.25 ft-water) of clean strainer head loss plus 24.69 cm-water (0.81 ft-water) of debris head loss equals the total assumed strainer head loss of 32.31 cm-water (1.06 ft-water) at 60 °C (140 °F). The staff noted that the design acceptance criteria for the strainer is that the total strainer head loss will be less than the as-designed maximum strainer head loss of 60.96 cm-water (2 ft-water). Accordingly, the staff finds that the total assumed strainer head loss of 32.31 cm-water (1.06 ft-water) is acceptable for purposes of the structural analysis, since the result of a total strainer head loss less than the 60.96 cm-water (2 ft-water) allowable head loss validates the strainer design with sufficient margin to provide reasonable assurance of the strainer structural integrity.

DCD Tier 2, Section 6.8.2.2.1 states that, following an accident, water introduced into containment drains to the hold-up volume tank (HVT). The debris in the containment could be transported to the HVT with this fluid and screened out using the trash rack.

RG 1.82, Revision 4, Section 1.3.9.2 states that licensees (or applicants) should compute structural loads on a strainer using the maximum pressure drop across the strainer. The evaluation addresses the limiting conditions corresponding to the break location and debris source term that induce the maximum total head loss at the ECCS strainer. In RAI 56, Question 06.02.02-11, the NRC staff requested that the applicant provide a discussion of the structural loading, analysis methodology, and total differential pressure (ΔP in psi) across the trash racks with and without latent debris greater than 3.81 cm (1.5 in) in diameter.

In its response to RAI 56-7996, Question 06.02.02-11 (ML15251A231), the applicant stated that the IRWST sump strainers are evaluated for design basis conditions, including seismic, and are capable of withstanding the force of full debris loading and hydrodynamic loads. This evaluation is described in the technical report, APR1400-E-N-NR-14002-NP, Revision 0, IRWST Sump Strainer and Trash Rack Structural Analysis. The trash rack located at the entrance to the HVT is also evaluated for structural integrity. Dead load (D), seismic load (Es), and hydro pressure load (Lh) are considered for the structural loads calculation. The structural element stresses are derived by a finite element model structural analysis which considers combined loads. Trapezoidal differential pressures of 0.41 through 3.25 psi are used to conservatively consider all screen meshes are blocked by debris. The staff finds the applicant addresses the design loading of sump strainers and trash racks, and the RAI response is acceptable. Therefore, RAI 56-7996, Question 06.02.02-11 is closed, since the applicant provided the structure loading of strainers and trash rack.

In DCD Tier 2, Section 3.9.3, Table 3.9-2 showed that the applicant used the square root of the sum of the squares (SRSS) method to combine the SSE and hydrodynamic loads of DCD Tier 2, Table 3.9-2. In reviews of Technical Report, APR1400-E-N-NR-14002-NP, Revision 0, “IRWST Sump Strainer and Trash Rack Structural Analysis,” the staff noted that Table 1-1 load combination of Service Level D does not show that the SRSS method is used consistently in combining the loads of safe shutdown earthquake (SSE) and postulated dynamic loads (i.e. Hydrodynamic Load (PDE)). Therefore, in RAI 532-8689, Question 06.02.02-45, the staff requested that the applicant clarify if the SRSS method is used to combine the SSE and PDE loads.

In its response to RAI 532-8689, Question 06.02.02-45 (ML17053B156), the applicant stated that the absolute sum method is used for the loading combination of the SSE and postulated dynamic loads in the IRWST sump structural analysis as listed in Table 1-1 of the Technical Report, APR1400-E-N-NR-14002-NP. The absolute sum method is more conservative than the SRSS method. Therefore, the IRWST sump structural analysis demonstrates that the IRWST sump strainer is designed to withstand the effects of earthquakes without loss of capability to perform the safety functions. The staff found the applicant addressed the design loading of sump strainers and trash racks, and therefore found the RAI response to be acceptable. RAI 532-8689, Question 06.02.02-45 is closed, since the applicant provided the staff requested information of the acceptable absolute sum method.

The staff conducted a regulatory audit on August 24-27, 2015, of the APR1400 component design and procurement specifications to facilitate the SE of the ASME Class 1, 2 and 3 components, component supports, and core support structures. This audit is described further in Section 3.9.3 of this SER. In addition, the staff requested the applicant to provide the design specifications and design reports (and/or stress reports) of the sump strainer for the staff’s reviews in this audit. The staff documented the results of the sump strainer design specification audit in an NRC staff memorandum dated September 1, 2015 (ADAMS Accession No. ML15219A319).

During the audit of APR1400 component design and procurement specifications, the staff found that the specified head loss of 1.5 (ft-water) in design specification number 9-447-N206 “IRWST Sump Strainer,” Section 4.05B “Operating and Design Conditions” was not consistent with the head loss of 2.0 (ft-water) of IRWST sump strainers that is specified in DCD Tier 2, Section 6.8.4.5.5 “Debris Head Loss.” During the follow-up audit of design and procurement specifications, the applicant stated that the design specification number 9-447-N206 “IRWST Sump Strainer,” Section 4.05B “Operating and Design Conditions” specified the strainer head loss of 1.5 (ft-water) for the reference OPR 1400 SKN 3 and 4. The applicant also stated that the specified head loss of 1.5 (ft-water) does not apply to the APR1400 design. The applicant provided a markup of the DCD description of a new COLA item to be added in the DCD Tier 2 Table 1.8-2 and Section 6.8.4.5.5. RAI 550-8737, Question 03.09.03-7 (ML17096A299) was issued to request the applicant to include in DCD Tier 2 that the COL applicant is to confirm that the IRWST sump strainer has a total strainer head loss less than the allowable head loss (0.61 m (2ft)). Based on the review of the DCD Tier 2, Revision 2, the staff has confirmed incorporation of the changes described above, therefore, RAI 550-8737, Question 03.09.03-7, is resolved and closed.

Based on the audit, the staff concluded that the design specifications met the provisions of IRWST sump strainer design in APR 1400 DCD Tier 1 and 2.

In Technical Report APR1400-E-N-NR-14001-P, Revision 0, Section 2.3, “Design for Prevention of Degraded Emergency Core Cooling System Performance,” the applicant stated that the HVT trash racks prevent debris particles larger than 1.5 inch from entering the HVT. However, Technical Report APR1400-E-N-NR-14002-NP, Revision 0, "IRWST Sump Strainer and Trash Rack Structural Analysis," Section 2.1, “Description of Trash Rack,” states that debris greater than 4 inch diameter is prevented from entering the HVT by a vertical trash rack and Section 2.2, “Trash Rack Analysis,” states that the trash rack is designed to prevent debris larger than 4 inch. These two technical reports appear to inconsistently describe the HVT trash rack mesh size as both 1.5 inch and 4 inch. In RAI 519-8687, Question 06-1, the staff requested the applicant to specify the mesh size of HVT trash rack and to revise the technical reports as applicable to consistently describe the HVT trash rack mesh size.

In its response to RAI 519-8687, Question 06.02.02-1 (ML16273A569), the applicant stated that the HVT trash racks prevent debris particles larger than 1.5 inch from entering the HVT because the HVT trash rack mesh size is 1.5 inch. The applicant revised Technical Report APR1400-E-N-NR-14002 NP, Revision 0, Section 2.1 and Section 2.2 to be consistent with Technical Report APR1400-E-N-NR-14001-P, Revision 0. The staff reviewed the changes in Revision 1 of APR1400-E-N-NR-14002 NP and confirmed that the changes are acceptable because they correct inconsistencies in the reported HVT trash rack mesh size. Therefore, RAI 519-8687, Question 06.02.02-1 is resolved and closed.

Additional evaluation of potential debris generation and associated effects including debris screen blockage is detailed in Section 6.2.2.4.1 of this SER.

6.2.2.4.3.11 In-Vessel Downstream Effects

The in-vessel downstream effects analysis is located in Section 15.6.5 of this SER.

6.2.2.4.3.12 Chemical Effects

6.2.2.4.3.12.1 Introduction

To determine the compliance of the APR1400 design with the requirements of 10 CFR Part 50, Appendix A, GDC 35, GDC 38, and 10 CFR 50.46(b)(5) as they relate to chemical debris (precipitates) formed in the post-loss-of-coolant accident (post-LOCA) containment pool, the staff reviewed the information in the DCD and supporting documents using the guidance listed in Section 6.2.2.3 of this SER. Chemical effects are corrosion products, gelatinous material, or other chemical reaction products that form as a result of interaction between the PWR containment environment and containment materials after a LOCA. SRP Section 6.2.2 does not provide specific guidance for chemical effects evaluations but references RG 1.82, Revision 3, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” NEI Report 04-07 guidance, "Pressurized Water Reactor Sump Performance Evaluation Methodology," and the staff's SER of NEI 04-07 for PWR sump debris evaluations. The APR1400 design conforms to Revision 4 of RG 1.82, which contains the following guidance for PWRs:

Section 2.2 Chemical Reaction Effects

a. The Westinghouse report, WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," and the limitations discussed in the associated SER provide an acceptable approach for PWRs to evaluate chemical effects that may occur in a post-accident containment sump pool (ML073521294, December 21, 2007).

b. Plant-specific information should be used to determine chemical precipitate inventory in containment. However, plant-specific chemical effect evaluations should use a conservative analytical approach. Additionally, "NRC Staff Review Guidance Regarding Generic Letter 04-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations," provides a general approach for PWR licensees to conduct plant-specific chemical effect evaluations (ML080230234, March 28, 2008).

c. WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid," is still under review by the NRC staff. When approved by the staff, it, along with the SER, will provide guidance for evaluation of chemical debris within the reactor (ML11292A021, October 12, 2011).

The staff subsequently approved WCAP-16793-NP with a SER dated April 8, 2013 (ML13084A154), including an approach for evaluating the effects of chemical debris within the reactor. Therefore, at the time the APR1400 application was submitted, RG 1.82, Revision 4, and the approved TR WCAP-16793-NP-A, contain the staff's guidance for addressing chemical effects in a post-LOCA containment pool and reactor vessel.

6.2.2.4.3.12.2 Applicant's Approach to Addressing Chemical Effects

In DCD Tier 2, Section 6.8.4.5.7, "Chemical Effects," and Technical Report (TR) APR1400-E-N-NR-140001-P, "Design Features to Address GSI-191," Section 3.8, "Chemical Effects," the applicant described the debris source materials, methodology, and analysis of potential chemical precipitation in the post-LOCA sump pool. The staff also reviewed supplemental information the applicant provided in a letter dated November 24, 2015 (ML15328A218). The applicant performed the analysis using the methodology in WCAP-16530-NP-A, which the industry developed for operating PWRs, and the staff approved with certain limitations and conditions. The methodology uses a computational spreadsheet to first calculate the "release rates" of elements from materials based on input values of sump pH, sump pool and containment temperature, and material quantities. Then the elements released into the sump pool are assumed to precipitate as certain chemical compounds based on the testing that supported the WCAP. The methodology includes procedures for preparing and using surrogate chemical precipitates in strainer and fuel assembly head-loss testing. DCD Tier 2, Section 6.8.4.5.7, stated that the chemical effects evaluation is based on the "plant specific environment and post-accident evolution." Because the DCD applies to all APR1400 plants, the applicant is removing the term "plant specific" to avoid confusion with individual operating plant conditions. Eliminating the term “plant specific” is based on Enclosure 1 to the applicant’s November 24, 2015, letter. The staff confirmed that the proposed changes were incorporated into DCD Tier 2, Revision 1, dated March 10, 2017; therefore, this issue is resolved and closed.

The staff compared the proposed materials and conditions for the APR1400 to those in WCAP-16530-NP-A to determine if the methodology is applicable to the APR1400. The staff compared the materials, peak temperature, pH range, and buffer type for the APR1400 to those evaluated to develop the WCAP-16530-NP-A methodology. The APR1400 analysis considered the materials in the containment that were evaluated for the WCAP (submerged and unsubmerged aluminum, concrete, and fiberglass insulation). The maximum temperature for the APR1400 sump pool, 118ºC (245ºF) and containment, 134ºC (273ºF) are adequately bounded by the maximum temperature evaluated for the WCAP, 132ºC (270ºF), with the only exception a period of a few hundred seconds when the containment temperature is between 132ºC and 134ºC (270 and 273ºF). The pH range of 4 to 10 for the APR1400 is also bounded by the WCAP (pH 4 to 12). Since the APR1400 materials and conditions are within the ranges used to develop the WCAP-16530-NP-A methodology (given the uncertainties in analyses and measurements), the staff finds the applicant’s use of this methodology acceptable for determining the APR1400 chemical effects.

6.2.2.4.3.12.3 Source Term for Chemical Effects

The chemical effects source term refers to the interaction of materials and environment (corrosion and dissolution) that results in dissolved species that could precipitate in the water pool. In order to use the WCAP-16530-NP-A methodology for the APR1400 chemical effects analysis, the applicant determined the temperature and pH transients, water volume, and material quantities.

The staff’s 2008 chemical effects guidance for closure of GL 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” (described in Section 6.2.2.4.3.12.1, above) states that the expectation is for the licensee (the applicant in this case) to “develop a thorough understanding of plant materials and the range of potential post-accident environments (e.g., temperature, spray and pool pH values, spray durations), and use this information to conservatively evaluate potential chemical interactions during the plant’s ECCS mission time.”

DCD Tier 2, Section 6.8.4.5.7 describes the chemical effects evaluation for the APR1400. The section begins by stating that the chemical effects evaluation was performed according to “NRC Staff Review Guidance Regarding Generic Letter 04-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations,” which is listed above in Section 6.2.2.4.3.12.1 of this SE. That statement is based on the response to RAI 391-8462, Question 06.02.02-34, dated July 1, 2016 (ML16183A314), which proposed corresponding changes to DCD Tier 2, Sections 6.8.4.5, “Performance of the IRWST Sump Strainer,” 6.8.4.5.2, “Debris Generation,” 6.8.4.5.7, 6.8.7, “References,” and to Sections 2.6, “Coatings,” 3.8, “Chemical Effects,” and 6, “References,” of technical report TR APR1400-E-N-NR-140001-P. The staff finds these changes acceptable because they are statements about the applicant using the appropriate guidance for the chemical effects evaluation. The staff confirmed that the proposed changes were incorporated into Revision 1 of the DCD and Revision 1 of APR1400-E-N-NR-140001-P. Therefore, RAI 391-84623, Question s06.02.02-34 is resolved and closed.

DCD Tier 2, Section 6.8.4.5.7 identifies the postulated chemical precipitates as aluminum oxyhydroxide, sodium aluminum silicate, and calcium phosphate formed from the aluminum, silicon, and calcium originating in metallic aluminum, concrete, and latent fiberglass insulation.

This description of the chemical effects is based on Section 3.8 of TR APR1400-E-N-NR-140001-P and revisions proposed to DCD Section 6.8.4.5.7 in the applicant's November 24, 2015 letter (ML15328A218). The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised to incorporate the proposed changes; therefore, this issue is resolved and closed.

The chemical effects evaluation included concrete areas that are in the 10-diameter (10D) zone of influence (ZOI) and exposed to the sump pool or spray. Computer-aided design was used for these calculations, with a result of 868 m^2 (9,344 ft^2) exposed concrete that would contribute to chemical effects. The sources include walls, floors, and equipment pedestals. The ZOI for concrete was clarified in the applicant's response to RAI 391-8462, Question 06.02.02-36, dated June 1, 2016 (ML16153A473), which explained that the 10D ZOI was applied to the qualified coating on the concrete despite the staff guidance accepting 4D for qualified epoxy coatings. The staff finds this acceptable because using a 10D ZOI results in more concrete in the chemical effects calculation compared to a 4D ZOI. The applicant proposed a corresponding revision to DCD Section 6.8.4.5.7. The staff confirmed that DCD Tier 2, Revision 1, dated March 10, 2017, was revised as committed in the response to RAI 391-8462, Question 06.02.02-36. Therefore, RAI 391-8462, Question 06.02.02-36 is resolved and closed.

The most significant material contributing to potential chemical effects is aluminum. The amount of aluminum is identified in TR APR1400-E-N-NR-14001-P, Revision 0, Section 3.8.3, "Evaluation Summary." The sources include heating, ventilation, and air conditioning (HVAC) equipment, ex-core detectors, refueling equipment, control element drive cooling fan, surveillance capsule handling tools, and power-operated safety relief valve actuators. In the case of the HVAC ducts, the surface area uncertainty is addressed by adding a 10 percent margin based on engineering judgment. The basis for the margin was not identified in Revision 0 of the TR. Identifying this as engineering judgement in Table 3.8-2, "Material Potentially Produced Corrosion Products," in TR APR1400-E-N-NR-14001-P, is based on Enclosure 1 to the applicant's November 24, 2015, letter (ML15328A218). The staff finds the change acceptable because it identifies the basis for the 10 percent margin. The staff confirmed that the proposed changes were incorporated into Revision 1 of APR1400-E-N-NR-14001-P. Therefore, this issue is resolved and closed.

The total aluminum surface area included in the chemical effects analysis was 216 m^2 (2,326 ft^2). This amount is identified in the TR but is less than the aluminum limit listed in DCD Section 6.1.1.2.1. Since the amount of chemical precipitate calculated in the chemical effects methodology is proportional to the aluminum surface area, assuming less surface area in the analysis is non-conservative. In RAI 520-8693, Question 06.02.02-41, dated September 13, 2016, the staff requested that the applicant address the discrepancy. The applicant responded to RAI 520-8693, Question 06.02.02-41, in a letter dated December 7, 2016 (ML16342C640). The response stated that DCD Tier 2, Section 6.1.1.2.1 would be corrected to 2,326 ft^2, and it proposed a revision to the corresponding page. The staff finds it acceptable to put the correct surface area value in the DCD. **Verification of the correction to the aluminum surface area in the applicant's next revision of the DCD, as proposed in the response to RAI 520-8693, Question 06.02.02-41 is being tracked as a confirmatory item**.

With respect to post-accident water volume, TR APR1400-E-N-NR-14001-P, Revision 0, Section 3.8.2, stated that the maximum IRWST water volume was used in the chemical effects analysis to ensure maximum material dissolution and precipitation quantity. In its letter dated November 24, 2015 (ML15328A218), the applicant proposed a reworded statement but did not provide a justification for the statement. In its response to RAI 391-8462, Question 06.02.02-37, dated July 1, 2016 (ML16183A314), the applicant explained that the conservatism from using the maximum IRWST water volume is a result of the larger water volume allowing more corrosion and dissolution before solubility limits are reached. However, since the WCAP-16530-NP-A methodology assumes all dissolved elements form a precipitate regardless of concentration, this explanation appears to be inconsistent with the methodology. The WCAP-16530-NP-A basic calculation uses pH- and temperature-dependent release rates to calculate the amount of aluminum, silicon, and calcium released into solution, without considering the concentration or solubility of those elements. As the response indicated, the WCAP-16530 methodology assumed that all of the dissolved elements form precipitates. For the WCAP-16530 methodology, parameters such as the material quantities, temperature profile, and pH profile can be used to ensure a conservative analysis, as discussed in the technical report. The design does not appear to propose a refinement to this methodology by applying a concentration or solubility limit. Therefore, in RAI 520-8693, Question 06.02.02-42, dated September 13, 2016, the staff requested that the applicant clarify if and how the water volume was used to maximize material dissolution and quantity of precipitates in its methodology.

The applicant responded to RAI 520-8693, Question 06.02.02-42 in a letter dated December 7, 2016 (ML16342C640). The response clarified that the assumption of maximum IRWST water volume does not affect the chemical effects analysis, and it proposed a corresponding revision to Subsection 3.8.2 of TR APR1400-E-N-NR-14001-P. The staff finds the response acceptable because the assumed water volume did not affect the chemical effects analysis. However, the proposed revision was not incorporated into Revision 1 of TR APR1400-E-N-NR-14001-P. **Therefore, verification of the change to APR1400-E-N-NR-14001-P proposed in the response to RAI 520-8693, Question 06.02.02-42, is being tracked as a confirmatory item.**

Temperature is one of the environmental variables in WCAP-16530-NP-A that determines the calculated release rates of aluminum, calcium, and silicon from materials in containment. On February 1, 2016, the staff issued RAI 391-8462, Question 06.02.02-35, requesting information about the source of the temperature profiles used in the chemical effects analysis, including the treatment of submerged vs. unsubmerged aluminum and the LOCADM calculation for deposition of chemical precipitates on the fuel. In its response to RAI 391-8462, Question 06.02.02-35, dated August 10, 2016 (ML16223A976), the applicant explained that the temperature profiles used in the chemical effects evaluation for TR APR1400-E-N-NR-14001-P, Revision 0, have since been updated and new chemical effects calculations were performed. The temperature profile is based on the GOTHIC code analysis shown in DCD Tier 2, Figure 6.2.1-4, "Containment Pressure and Temperature vs. Time; LOCA-DEDLSB with Minimum SI Flow." The GOTHIC analysis ended at 1,000,000 seconds (approximately 11.6 days) and was extended to 30 days for the TR using a logarithmic extrapolation. The response proposed revisions to TR Tables 3.8-4 and 3.8-5, including an increase in the amount of calculated chemical precipitate (evaluated below in the section titled "Type and Amount of Chemical Precipitates"). The staff was not able to complete its evaluation of the chemical precipitate calculation because the response did not provide the updated 30-day temperature profiles, the basis for the new extrapolation, or how the revision to the temperature profiles is documented. The staff requested this information in RAI 520-8693, Question 06.02.02-43, dated September 13, 2016, while RAI 391-8462, Question 06.02.02-35 was closed and unresolved.

In its response to RAI 520-8693, Question 06.02.02-43, dated December 7, 2016 (ML16342C640), the applicant provided a plot of the IRWST and containment air temperatures extended to 30 days. The response explains that the full profile is based on the GOTHIC calculation. The staff finds this acceptable because it clarified that the same GOTHIC analysis used to determine the temperature for the first 11.6 days (DCD Figure 6.2.1-4) was used for the

30-day profile. The response also indicated the GOTHIC model was being revised, and that updated temperature profiles and chemical effects analyses would be incorporated in TR APR1400-E-N-NR-14001-P. **Verification of these revisions to TR APR1400-E-N-NR-14001-P based on the response to RAI 520-8693, Question 06.02.02-43, is being tracked as a confirmatory item.**

The sump and spray pH is another environmental parameter that determines the material release rates. The applicant considered the bounding pH conditions for the APR1400 and calculated the chemical effects using a pH profile intended to maximize the precipitate quantity. Because of the large surface area of aluminum compared to other contributing materials, the applicant used the high end of the pH range, which corresponds to higher aluminum corrosion rates. In its response to RAI 391-8462, Questions 06.02.02-38 and 06.02.02-39, dated July 26, 2016 (ML16208A575), the applicant described the pH profile used in the chemical effects analysis. The response included revisions to the November 24, 2015 responses (ML15328A218) to KHNP Action Items 6-19.7, 6-19.8, and 6-19.9. The RAI responses and revised action item responses propose corresponding changes to Section 3.8.1, Section 3.8.2, Table 3.8-1, and Figure 3.8-1 of TR APR1400-E-N-NR-14001-P. The staff found the proposed changes acceptable because they corrected and clarified the description of the pH values used in the chemical effects analyses and how they apply to submerged and unsubmerged materials. The staff confirmed that the proposed changes were incorporated into Revision 1 of TR APR1400-E-N-NR-140001-P. Therefore, RAI 391-8462, Questions 06.02.02-38 and 06.02.02-39 are resolved and closed.

The July 26, 2016, letter (ML16208A575) identified the short-term bounding pH range as 4 to 10, and long-term bounding pH range as 7 to 8.5, with the boundary between short-term and long-term identified as four hours after the accident initiation. In the analysis, the applicant assumed that the CSS would be operated continually following accident initiation in order to maximize exposure of the unsubmerged aluminum. Based on this approach, the applicant used a pH value of 10 for the first four hours and a pH value of 8.5 for the remainder of the 30 days for the chemical effects analysis. The same pH profile was applied to the submerged and unsubmerged materials. With respect to submerged vs. unsubmerged aluminum and concrete, the July 26, 2016 letter (ML16208A575) clarified that in the analysis the aluminum was unsubmerged and the concrete was treated as submerged. The staff finds this acceptable for aluminum because the applicant determined that the aluminum sources in the plant will not be submerged. For concrete, the staff finds it acceptable to treat unsubmerged concrete as submerged because WCAP-16530-NP-A includes only submerged concrete.

In summary, the materials for the chemical precipitate source term are unsubmerged aluminum (216 m^2/2,326 ft^2), submerged concrete (868 m^2/9,344 ft^2), and submerged latent fiber (0.177 m^3/6.25 ft^3). For the chemical effects calculation, the applicant assumed high pH values, which results in the greatest quantity of chemical precipitates for these source materials. This approach is acceptable because it conforms to the March 2008 chemical effects guidance, which states that input of plant parameters should be done in a manner that results in a conservative amount of precipitate formation (Section 3.7.c.i). The 30-day post-LOCA temperature profiles for the sump and spray are acceptable because they are from the GOTHIC code containment analysis. RAI 520-8693, Questions 06.02.02-41, 06.02.02-42, and 06.02.02-43, are being tracked as confirmatory items pending revisions to the DCD and TR APR-1400-E-N-NR-140001-P. These questions are related to the aluminum quantity in containment, the assumed water volume, and the temperature profiles, respectively, as discussed above in this section.

6.2.2.4.3.12.4 Type and Amount of Chemical Precipitates

Using the material quantities, sump pool temperature, containment temperature, and pH transient as described above, as inputs to the WCAP-16530-NP-A spreadsheet, the applicant calculated the following chemical precipitate quantities:

Aluminum oxyhydroxide (AlOOH)	240.1 kg (529.3 lbm)
Sodium aluminum silicate ($NaAlSi_3O_8$)	4.3 kg (9.5 lbm)
Calcium phosphate ($Ca(PO_4)_2$)	0.7 kg (1.5 lbm)
TOTAL	245 kg (540 lbm)

These amounts are based on the applicant's response to RAI 391-8462, Question 06.02.02-35, dated August 10, 2016 (ML16223A976), and are higher than the amounts listed in TR APR1400-E-N-NR-14001-P, Revision 0, Table 3.8-4. As discussed above in the Section titled, "Source Term for Chemical Effects," the increase is a result of a change to the temperature profile for the sump pool and spray. **Verification of these changes in the TR is being tracked as part of the RAI 520-8693, Question 06.02.02-43 confirmatory item.** According to the WCAP-16530-NP-A methodology, these precipitates are generated from aluminum released by aluminum metal or alloys, aluminum released by concrete, silicon released by fiberglass insulation, and calcium released by concrete. With respect to the total amount of precipitate, WCAP-16530-NP-A assumes all dissolved calcium in the presence of phosphate, and all dissolved aluminum, form precipitates. In its SER on WCAP-16530-NP-A, the staff found this to be a reasonable assumption for calcium and a conservative assumption for aluminum. The applicant followed this approach in the APR1400 analysis and did not propose any WCAP refinements to reduce conservatism.

The staff's expectation expressed in the March 2008 guidance is that input of plant parameters should be done in a manner that results in a conservative amount of precipitate formation (Section 3.7.c.i of the guidance). In evaluating this for the APR1400, the staff performed calculations using the WCAP-16530-NP-A spreadsheet. In addition to using the same inputs (temperature, pH, materials) as the applicant, the staff varied inputs to assess the conservatism of the chemical precipitate proposed in the design. For the design basis conditions, the staff estimated the sump and spray temperature profiles based on the DCD Tier 2, Figure 6.2.1-4 and the WCAP-16530-NP-A results presented in TR APR1400-E-N-NR-140001-P. For this case the staff calculated approximately the same amount as the applicant (right-hand column). The staff also evaluated an acidic pH (4) and a pH profile that increased from 4.2 to a steady value of 7.3 (representing a realistic buffering effect for TSP in borated water). For these cases the calculated precipitate total was much lower than the applicant assumed in the APR1400 analyses. The lower precipitate quantity calculated by the staff for realistic conditions indicates that the applicant's approach led to a conservative chemical precipitate quantity in accordance with the staff guidance. **Verification of the corresponding revisions to APR1400-E-N-NR-140001-P and the DCD proposed changes in the responses to RAI 520-8693, Questions 06.02.02-41, 06.02.02-42, and 06.02.02-43 are being tracked as confirmatory items**.

Chemical Precipitate Quantity: Comparison of Applicant and Staff WCAP-16530-NP-A Calculations

Analysis performed		**Calculated Precipitate Quantity, kg**			
	Conditions	AlOOH	$NaAlSi_3O_8$	$Ca_3(PO_4)_2$	Total
Applicant design basis analysis	Post-LOCA temperature. pH from10.0 to 8.5	240.1	4.3	0.7	245.1
Staff calculation	Estimated[1] post-LOCA temperature and pH 10.0 to 8.5[2] (design basis condition)	241	7	1	249
Staff calculation	Estimated[1] post-LOCA temperature and constant low pH of 4	58	7	1	66
Staff calculation	Estimated[1] post-LOCA temperature and gradually increasing pH from 4.2 to 7.3	98	7	1	106

Note 1: Staff requested the actual temperature values in RAI 520-8693, Question 06.02.02-43.43.

Note 2: In all cases the sump and spray pH were the same.

6.2.2.4.3.12.5 Chemical Precipitates in Strainer Head-Loss Testing

This Section describes the staff's review of the chemical debris component in the design-basis strainer testing for the APR1400. The staff's review of the strainer testing is discussed mainly in Section 6.2.2.4.3.6 of this SER. The applicant performed integrated testing with fiber, particulate, and chemical precipitate, and described the use of the chemical precipitate in Appendix C of TR APR1400-E-N-NR-140001-P. The applicant performed the testing with the goal of all chemical precipitate eventually reaching the strainer (i.e., no settlement or solubility credit).

All of the postulated chemical precipitate for the plant was represented in the testing by an AlOOH surrogate precipitate. For the plant calculation using WCAP-16530-NP-A, AlOOH was about 97 percent of the total chemical precipitate. The staff's SER for WCAP-16530-NP-A finds that the AlOOH surrogate (prepared according to certain guidelines) has adequate settlement and filterability characteristics to represent post-LOCA chemical precipitates in strainer head-loss tests. Using AlOOH surrogate also avoids the chemical hazard associated with handling sodium silicate.

TR APR1400-E-N-NR-14001-P, Revision 0, Section C.5.1.3, "Aluminum Oxy-hydroxide Preparation," describes the applicant's procedures for preparing and testing the settling rate of the AlOOH surrogate according to the procedures in WCAP-16530-NP-A. The description of the procedures indicates that the chemical precipitate met the principal acceptance criterion for settlement properties. However, the description did not identify the size of the sample for settlement testing, whether it was diluted and whether it met all of the acceptance criterion in WCAP-16530-NP-A. The staff requested this information in RAI 391-8462, Question 06.02.02-

40, dated February 1, 2016. In its response to RAI 391-8462, Question 06.02.02-40, dated July 1, 2016 (ML16183A314), the applicant indicated the following:

- The surrogate chemical precipitate was prepared at a concentration of 11 grams per liter (g/L) in the mixing tank.
- The intent was to keep the surrogates suspended during head-loss testing.
- The surrogate solution was mixed continuously in the mixing tank prior to addition to the testing tank.
- When testing the settling properties of the surrogate solution, dilution of the samples to 2.2 g/L was not required based on Section 4.4 of the NRC SE for WCAP-16530-NP-A.

The staff determined that the response did not provide sufficient detail to determine if the surrogate was prepared and tested according to the staff guidance. Specifically, the staff needed confirmation or clarification of the understanding that the chemical precipitate passed the settling test at a concentration of 11 g/L according to the acceptance criteria in WCAP-16530-NP-A. Therefore, on September 13, 2016, the staff requested this information in RAI 520-8693, Question 06.02.02-44. In the response to RAI 520-8693, Question 06.02.02-44, dated December 7, 2016 (ML16342C640), the applicant stated that the surrogate chemical was prepared within 24 hours of testing and passed the WCAP-16530-NP-A settling test at a concentration of 11 g/L. The staff finds this acceptable because it clarifies that the applicant demonstrated adequate settling properties for the surrogate chemical in accordance with staff guidance. Therefore, RAI 520-8693, Question 06.02.02-44 is resolved and closed.

The chemical precipitate was added in batches by manually transferring buckets of the precipitate solution from the mixing tank into the test tank. The chemical precipitate was added after all of the particulate and fiber were in the pool. In the first test, APR-HL-0813-1, three chemical additions were made, accounting for about 20 percent of the revised design basis chemical debris proposed in the response to RAI 391-8462, Question 06.02.02-35, dated August 10, 2016 (ML16223A976). Since no head loss increase was observed after the second and third additions, no additional chemical was added. Stopping chemical additions when no additional head loss is observed conforms to the 2008 staff guidance. In the second test, APR-HL-0913-2, all of the precipitate was added, which corresponds to about 76 percent of the revised design basis amount proposed in the response to RAI 391-8462, dated August 10, 2016 (ML16223A976). The chemical addition commenced approximately two hours into the test and was completed in about 23 hours. The applicant terminated the tests upon completing at least 10 pool turnovers and reaching a head-loss stability of less than one percent change per hour. In both tests, the applicant reported bare strainer surface area due to the relatively small amount of total debris.

In the response to RAI 391-8462, Question 06.02.02-35, dated August 10, 2016 (ML16223A976), the applicant stated that the increase in the design basis chemical debris resulting from the revised temperature profiles would not affect the results of the strainer design and testing. The applicant's statement is based on the testing observation that no additional head loss occurred after the first two chemical additions. It conforms to staff guidance to allow ceasing chemical additions if chemicals reach the test strainer without causing additional head loss. **Verifications of the related revisions to APR1400-E-N-NR-140001-P and the DCD proposed in the responses to RAI 520-8693, Questions 06.02.02-41, 06.02.02-42, and 06.02.02-43 are being tracked as confirmatory items.**

6.2.2.4.3.12.6 Chemical Precipitates in Fuel Assembly Testing

This Section describes the staff's review of the chemical debris component in the design-basis fuel assembly testing for the APR1400. The applicant documented the testing in TR APR1400-K-A-NR-14001-P, Revision 1, "In-vessel Downstream Effect Tests for the APR1400." The staff's review of the fuel assembly testing is discussed mainly in Section 15.6 of this SER. The applicant performed integrated testing with fiber, particulate, and chemical precipitate and described the use of the chemical precipitate in Section 3.3, "Debris Description," of TR APR1400-K-A-NR-14001-P. The applicant performed the testing with the assumption of all chemical precipitate bypassing the strainer, which is acceptable because it is conservative in the fuel assembly testing to assume none of the chemical precipitate settles out of the flow or is captured in the strainer debris bed.

Table 3-2, "Debris Types and Amounts per Fuel Assembly (FA)," of TR APR1400-K-A-NR-14001-P, Revision 1, shows how the applicant calculated the amount of chemical precipitate to include in the three test conditions (hot leg break, cold leg break, and cold leg break after hot leg switchover). The original plant quantity of 185 kg (408 lbm), calculated for TR APR1400-K-A-NR-14001-P, Revision 0, was divided by the number of fuel assemblies (241) to determine the mass of chemical precipitate for the single fuel assembly tests (768 g, 1.69 lbm). The tests used AlOOH surrogate prepared outside the test loop to represent all of the chemical debris in the fuel assembly tests. The staff finds this acceptable for the reasons discussed above in Section 6.2.2.4.3.11.6.

Based on the revised chemical precipitate quantity proposed in the response to RAI 391-8462, Question 06.02.02-35, dated August 10, 2016 (ML16223A976), the amount of chemical precipitate added to the fuel assembly testing was approximately 76 percent of the design basis amount. The test results in TR APR1400-K-A-NR-14001-P, Revision 1, show that only the first chemical addition caused an increase in head loss. Subsequent additions had no effect, which indicates that an increase in the design basis amount does not affect the test conclusions. Since additional chemical additions had no effect after adding a large percentage of the design basis amount, the staff finds that the amount tested is acceptable. Since this depends on confirming the design basis chemical debris quantity **verifications of the related revisions to TR APR-E-N-NR-140001-P and the DCD proposed in the response to RAI 520-8693, Questions 06.02.02-41, 06.02.02-42, and 06.02.02-43 are being tracked as confirmatory items.**

6.2.2.4.3.12.7 Chemical Effects Summary

The staff evaluated the applicant's chemical effects methodology and how it was applied to determining the chemical source term and performing sump strainer and fuel assembly head-loss testing. The staff found that the applicant's methodology conforms to staff guidance. Revisions proposed to the DCD and TR APR1400-E-N-RN-140001-P in the response to RAI 520-8693, Questions 06.02.02-41, 06.02.02-42, and 06.02.02-43 are being tracked as confirmatory items. The confirmatory items are related to the amount of aluminum in containment, the temperature profile used in calculating the amount of chemical precipitate, and the role of sump water quantity in calculating the amount of chemical precipitate.

6.2.2.5 Combined License Information Items

The DCD Tier 2, Revision 1, Section 6.8.6 contains six COL information items pertaining to the IRWST. These items included: establishing a cleanliness, housekeeping, and foreign material exclusion (FME) program, implementing procedures to ensure that administrative controls for plant modifications and temporary changes evaluate the potential for sump strainer blockage, assessing and managing the risk associated with maintenance and temporary changes, establishing a containment coatings monitoring program, confirming that the RMI used in the plant is one of the tested RMIs in NUREG/CR-6808, and confirming that post-accident ECCS flow rates remain applicable to those used in the analyses for a minimum of 30 days. The acceptability of the COL items is evaluated above in Section 6.2.2.4.3.3, for COL items 6.8(1), 6.8(2), 6.8(3) and 6.8(6), and in Section 6.2.2.4.3.12 for items 6.8(4) and (5). The staff concluded that no additional COL information items were needed.

6.2.2.6 Conclusion

In conclusion, subject to the closure of the confirmatory items noted above, the staff finds that the applicant has satisfied the requirements in 10 CFR Part 50 GDCs 38, 39 and 40 and 10 CFR 50.46(b)(5) as well as the requirements associated with GDCs 4 and 35 pertinent to the containment spray system, which require that a containment heat removal system exist and that the applicant demonstrate adequate long-term cooling, including NPSH margin in the presence of LOCA-generated and latent debris. Specifically, the applicant demonstrated the CSS will act to cool containment during a DBA, that the system is inspected and tested to demonstrate operability, and that the IRWST and recirculated containment water will act as a sufficient water source for the CSS and ECCS pumps in the event of a LOCA or transient requiring long term cooling. The staff evaluation of the efficacy of the spray system at reducing containment pressure is located in Section 6.2.1.1 of this SE. Based on the aforementioned considerations, the staff concludes the design of the CSS is acceptable and meets the applicable regulations specified in Section 6.2.2 of the SRP. Additionally, as documented above, the staff finds that the design provides for a sufficient long term recirculation water source even in the presence of LOCA-generated and latent debris.

The staff also finds the applicable system performance proposed ITAAC in Table 2.11.2-4 of DCD Tier 1 are necessary and sufficient to provide reasonable assurance that, if met, the CSS will be in conformity with the certified design, the applicable regulations, in accordance with 10 CFR 52.47(b)(1). ITAAC for functional qualification of the pumps and valves, as well as the ASME qualification ITAAC are discussed in Chapter 3 of this SER. As a whole, the ITAAC ensure the CSS conforms to the design as described in the DCD by providing the classifications for components and provides the necessary controls and alarms for monitoring system operation. This conclusion is based on the inclusion of comprehensive inspections, tests, and analyses that verify the acceptability the CSS and that the system parameters for the as-built components will satisfy the assumptions used in the analyses documented in DCD Chapter 6. Specific ITAAC are discussed in further detail above.

6.2.4 Containment Isolation System

6.2.4.1 Introduction

The containment isolation system (CIS) consists of isolation barriers, such as valves, blind flanges, and closed systems, and the associated instrumentation and controls (I&Cs) required for the automatic or manual initiation of containment isolation. The purpose of the CIS is to permit the normal or post-accident passage of fluids through the containment boundary, while protecting against release of fission products to the environment that may be present in the containment atmosphere and fluids as a result of postulated accidents.

6.2.4.2 Summary of Application

DCD Tier 1: The Tier 1 information pertaining to the CIS is given in DCD Tier 1, Section 2.11, "Containment System."

DCD Tier 2: The applicant provided a Tier 2 description of the CIS in DCD Tier 2, Section 6.2.4 summarized here, in part, as follows:

The containment prevents or limits the release of fission products to the environment. The CIS allows the free flow of normal or emergency-related fluids through the containment boundary in support of reactor operations but establishes and preserves the containment boundary integrity. The CIS includes the system and components (piping, valves, and actuation logic) that establish and preserve the containment boundary integrity.

ITAAC: The ITAAC for the CIS are specified in DCD Tier 1, Table 2.11.3-2, "Containment Isolation System ITAAC (Inspections Tests, Analyses and Acceptance Criteria)."

TS: DCD Tier 2, Chapter 16, Section 3.6 provides limiting conditions for operation and SRs for the CIS.

Technical Reports: There are no technical reports for this area of review.

APR1400 Interface Issues identified in the DCD: There are no APR1400 interface issues for this area of review.

Site Interface Requirements identified in the DCD: There are no site interface requirements for this area of review.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex):

TMI Item II.B.8, "Three- Foot Diameter Containment Penetration."
TMI Item II.E.4.2, "Actuation and Control Features for Isolation Valves."
TMI Item II.E.4.4, Vent/Purge Valve Positions."

RTNSS: There is no RTNSS for this area of review.

In accordance to 10 CFR 20.1406: DCD Tier 2, Table 12.4-10, "NRC RG 4.21 Design Objectives and Applicable DCD Section Information for Minimizing Contamination and Generation of Radioactive Waste," describes the provisions related to the CIS for minimizing contamination.

CDI: There is no CDI for this area of review.

6.2.4.3 Regulatory Basis

The relevant requirements for the Commission regulations for this area of review and the associated acceptance criteria are specified in Section 6.2.4 of NUREG–0800, the SRP, and are summarized below. Review interfaces with other SRP Sections can be found in SRP Section 6.2.4.

- GDC 1, "Quality standards and records," requires that SSCs important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 2, “Design bases for protection against natural phenomena,” requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

- GDC 4, “Environmental and dynamic effects design basis,” requires that SSCs important to safety shall be designed to accommodate the effects and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 16, “Containment Design,” requires that the reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- GDC 54, “Systems penetrating containment,” requires that piping systems penetrating the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems, and as it relates to designing such piping systems with a capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

- GDC 55, “Reactor coolant pressure boundary penetrating containment,” and GDC 56, “Primary containment isolation,” require that lines penetrating the primary containment boundary as parts of the RCPB (GDC 55), or as direct connections to the containment atmosphere (GDC 56), be provided with containment isolation valves (CIVs) as follows, unless it can be demonstrated that the containment isolation provisions are acceptable on some other defined basis:

    - One locked-closed isolation valve inside and one outside containment; or

    - One automatic isolation valve inside and one locked-closed isolation valve outside containment; or

    - One locked-closed isolation valve inside and one automatic isolation valve outside containment; or

    - One automatic isolation valve inside and one outside containment.

- GDC 57, “Closed systems isolation valves,” requires that lines penetrating the primary containment boundary that are neither part of the RCPB nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment.

- The regulations in 10 CFR 50.34(f)(2)(xiv)(A) requires that CISs are provided that ensure all non-essential systems are isolated automatically by the CIS.

- The regulations in 10 CFR 50.34(f)(2)(xiv)(B) requires that CISs are provided that for each non-essential penetration (except instrument lines) have two isolation barriers in series.

- The regulations in 10 CFR 50.34(f)(2)(xiv)(C) requires that CISs are provided that do not result in the reopening of the CIVs on resetting of the isolation signal.

- The regulations in 10 CFR 50.34(f)(2)(xiv)(D) requires that CISs are provided that utilize a containment setpoint pressure for initiating containment isolation as low as is compatible with normal operation.

- The regulations in 10 CFR 50.34(f)(2)(xiv)(E) requires that CISs are provided that include automatic closing on a high radiation signal for all systems that provide a path to the environs.

- The regulations in 10 CFR 50.34(f)(2)(xv) requires that CISs are provided a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure, and that provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.

- The regulations in 10 CFR 50.34(f)(3)(iv) requires one or more dedicated containment penetrations, equivalent in size to a single 3 feet (0.9 m) diameter opening in order not to preclude future installation of systems to prevent containment failure such as a filtered, vented containment system.

- The regulations in 10 CFR 50.63(a)(2) requires, in part, that appropriate containment integrity is maintained in the event of a station blackout (SBO) for a specified duration.

- The regulations in 10 CFR 52.47(a)(8) requires, in part, that an application for a standard DC contain the information necessary to demonstrate compliance with any technical requirements set forth in 10 CFR 50.34(f).

- The regulations in 10 CFR 52.47(b)(1) requires that an application for a standard DC contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that if the inspections, tests, analyses and acceptance criteria are performed, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations.

- The regulations in 10 CFR 20.1406, "Minimization of Contamination," requires that applicants for a standard DC shall describe how facility design and procedure for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Acceptance criteria adequate to meet the above requirements are described in NUREG–0800, SRP Section 6.2.4 and RG 1.141, "Containment Isolation Provisions for Fluid Systems."

The applicant's CIS is acceptable if it meets the relevant regulatory guidance. This will ensure that the relevant requirements of 10 CFR 50.34, "Contents of Construction Permit and Operating License Applications; Technical Information," 10 CFR 50.63, "Loss of All Alternating Current Power," 10 CFR 52.47, "Contents of Applications; Technical Information," 10 CFR Part 50, Appendix A, GDC 1, GDC 2, GDC 4, GDC 16, GDC 54, GDC 55, GDC 56 and GDC 57, and 10 CFR 20.1406 are met.

6.2.4.4 Technical Evaluation

The staff reviewed the APR1400 DCD, Revision 1, Section 6.2.4 in accordance with NUREG–0800, SRP Section 6.2.4, Revision 3, March 2007, and RG 1.141. In addition, APR1400 DCD Section 6.2.8, "Combined License Information," is included in this review.

The staff's review encompassed the following areas specified by Section 6.2.4 of the SRP, 10 CFR 50.34(f)(2)(xiv) (A) through (E), 10 CFR 50.34(f)(2)(xv) and 10 CFR 50.63(a)(2):

- CIS design, including:
    - the number and location of isolation valves (e.g., the isolation valve, arrangements, location of isolation valves with respect to the containment wall, purge and vent valve conformance to BTP 6-4, and instrument line conformance to RG 1.11)
    - the actuation and control features for isolation valves
    - the normal positions of valves, and the positions valves take in the event of failures
    - the initiating variables for isolation signals, and the diversity and redundancy of isolation signals
    - the basis for selecting closure time limits for isolation valves
    - the redundancy of isolation barriers
    - the use of closed systems inside containment as isolation barriers
- the protection provided for the CIS against loss of function caused by missiles, pipe whip, and natural phenomena
- environmental conditions in the vicinity of the CIS and equipment and its potential effect
- the mechanical engineering design criteria applied to isolation barriers and equipment
- the provisions for alerting operators of the need to isolate manually controlled isolation barriers in the event of leakage or an accident
- the provisions for, and TS pertaining to, operability and leak rate testing of isolation barriers
- the calculation of containment atmosphere released prior to isolation valve closure for lines that provide a direct path to the environs
- containment purging/venting requirements of 10 CFR 50.34(f)(2)(xv)
- containment penetration requirements of 10 CFR 50.34(f)(3)(iv)

The following sections provide a discussion of the staff's findings and conclusions for each of the above review areas.

6.2.4.4.1 Number, Location, and Arrangement of Isolation Valves

The regulatory requirements relating to number, location, and arrangement of isolation valves serving containment piping penetrations are specified in GDC 55, GDC 56, and GDC 57. GDC 55 and GDC 56 require two isolation valves, one inside and one outside containment, per penetration, and the valves shall be locked closed or automatic, with the restriction that a simple check valve may not be used as an automatic valve outside containment. GDC 57, which applies to penetrations for which there is a closed system inside containment, requires one locked closed, automatic (but not simple check) or remote manual isolation valve outside containment. The staff reviewed the applicant's proposed use of CIVs, as described in DCD Table 6.2.4-1, "List of Containment Penetrations and System Isolation Positions," for conformance with these GDC. The staff reviewed the valve arrangement information for each penetration to confirm that the number, location, and arrangement conform to the acceptance criteria. The staff used acceptance criteria described in NUREG–0800, Section 6.2.4, and guidance described in RG 1.141, which endorses ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems."

SRP Section 6.2.4, Section II, Acceptance Criteria 4 and 5 provide review guidance for review of containment isolation provisions on another design basis than GDC 55 and GDC 56, for lines consisting of two valves located outside of containment or a single valve outside of containment respectively. This "other defined basis" for GDC 55 and GDC 56 is described in detail in ANSI N271-1976. The staff noted that the APR1400 design does not include other design basis for GDC 55 and GDC 56, for lines consisting of two valves located outside of containment. Therefore, the staff determined that SRP Section 6.2.4, Section II, Acceptance Criteria 4 does not apply to the APR1400 design. Each of the four SIP suction lines (Figure 6.2.4-1, "Control Isolation Valve Arrangement," Sheet 4) consists of a single remote manual motor operated valve located outside containment. The SIS is credited as a closed system outside containment. The system is protected from missiles, is classified as seismic Category 1, and Quality Group B standards. The SIS has a design temperature and pressure rating at least equal to that for the containment.

On January 5, 2016, the staff issued RAI 357-8344, Question 06.02.04-1, requesting the applicant to provide additional justification to demonstrate that the configuration meets the requirements of the ANSI N271-1976 "other defined basis" of GDC 56. The staff requested the applicant to provide a discussion that shows how a single active failure can be accommodated with only one valve in the line and a leakage detection from that portion of the SIS outside containment while maintaining system integrity during normal plant operation. Without this information, the staff cannot confirm that SRP Section 6.2.4, Section II, Acceptance Criteria 5, is met. The applicant submitted its response to RAI 357-8344, Question 06.02.04-1, on June 30, 2016 (ML16182A591). In its response, the applicant stated that each of the four SI pump suction lines, including feed to the two CS pump suction lines which provide in-containment refueling water storage tank (IRWST) water for safety injection and containment spray, contain a single motor-operated gate valve for containment isolation. Each of these four valves is locked open, fail-as-is, and remains open during shutdown and accident conditions. Main control room (MCR) operators have the ability to close any of these four valves manually to prevent loss of IRWST water due to any identified downstream leakage. For this reason, these valves are also equipped with an alternate emergency power supply to ensure a single active failure can be accommodated. These valves can be used to isolate the IRWST for the performance of pump maintenance and are located outside the containment in valve rooms inside the auxiliary building. Leakage from the valves and piping is routed to the floor drain sumps in the corresponding quadrants, where sump levels are monitored. The MCR operators can detect leakage from individual segments of the piping and valve by the indication of the high sump water level in the sump located within the quadrant in which the associated SI/CS piping and valve are located. These cubicles are also designed to facilitate periodic access for surveillance and inspection of the valves and the associated piping outside the containment for their integrity and leak tightness. In addition, these valves are included in the in-service inspection requirements of ASME XI to provide reasonable assurance of their operability, reliability, and the ability to meet the closure requirements (ANS 56.2, Section 4.4.4), as described in DCD Subsection 6.2.4.4. The SI/CS systems circulate water from inside the containment through outside containment components and back to the inside of the containment, and are therefore closed systems. Since the piping and valves in the suction lines are designed to preclude a breach of piping integrity, which is described in DCD subsection 3.6.2, and since leak detection is provided for the corresponding segments of piping and valves, protective housing is not provided for any of the four segments of piping and valves in accordance with SRP 6.2.4 SRP Acceptance Criteria 5. DCD Subsection 6.2.4.3, "Design Evaluation," provides a discussion which addresses these issues.

Based upon the safety features aforementioned along with the in-service inspection provisions, this design approach using a single valve configuration is reliable and meets the intended safety injection function requirements of SRP 6.2.4, RG 1.141, and ANSI 271-1976. The staff has reviewed the applicant response as evaluated above and finds it acceptable. Therefore, the staff considers RAI 357-8344, Question 06.02.04-1 resolved and closed.

On January 5, 2016, the staff issued RAI 357-8344, Question 06.02.04-2, requesting the applicant to clarify design requirements that prevent debris from interfering with containment valve closure of SIS pump suction line. The applicant's response (ML16182A591) to RAI 357-8344, Question 06.02.04-2 was received on June 24, 2016. In its response, the applicant stated the in-containment refueling water storage tank (IRWST) provides a source of borated water for containment spray (CS) and safety injection (SI) through penetrations numbers 24, 25, 26, and 27 and the associated containment isolation valves (CIVs) 304, 308, 305, and 309. The open end of each suction pipe in the IRWST is equipped with a debris strainer that satisfies NEI 04-07, and conforms to the guidance in NRC RG 1.82. The strainers limit debris from entering the suction piping. Debris that gets entrained in the suction, mostly very fine fibrous material, will be transported through the valves and system due to the velocity of the system flows. The isolation valves are gate valves and will not accumulate debris in the seating area due to the inherent design and flow through the valve. The seating surfaces of the safety related ECCS gate valves are hard metallic composition not prone to wear. The composition of the fluid is such that it will not have an impact on the capability of the motor operated valve closure.

A discussion regarding the debris blockage of components downstream of the IRWST sump strainers during a LOCA and long-term post-LOCA related to Generic Safety Issue (GSI)-191 is summarized in DCD subsection 6.8.4.5.9. Based on the above discussion, the staff finds the APR1400 includes the design and an evaluation pertaining to the limits of debris entering into the SI piping, thereby ensuring the closure of the CIVs is not inhibited. Additional changes to the DCD are not needed. The staff has reviewed the applicant response as evaluated above and finds it acceptable. Therefore, the staff considers RAI 357-8344, Question 06.02.04-2 resolved and closed.

Based on the review of the information provided in the DCD as described above, the staff finds that the design satisfies SRP 6.2.4, Acceptance Criteria items 4 and 5 and meets the requirements of GDCs 55 and 56 as it relates to provisions for containment isolation on other the defined bases as allowed by those GDCs.

As stated in Section SRP Section 6.2.4, Section II, Acceptance Criterion 1, RG 1.11 describes acceptable containment isolation provisions for instrumentation lines. In DCD Section 6.2.4.1, "Design Bases," the applicant has stated that I&C sensing lines that penetrate the containment are provided with containment isolation provisions that meet the intent of NRC RG 1.11 and RG 1.141.

On January 5, 2016 the staff issued RAI 357-8344, Question 06.02.04-12, the staff requested the applicant to clarify what is meant by "meet the intent of RG 1.11 and RG 1.141." If the containment isolation provision of instrument lines meets the isolation guidelines on some other defined bases not indicated in RG 1.11 and RG 1.141, then the staff requested that the applicant indicate clearly and justify the reasons in the DCD. In its response (ML16057A075), the applicant stated that the I&C sensing lines that penetrate the containment are provided with containment isolation provisions that meet the requirements of NRC RG 1.11 and NRC RG 1.41. The applicant stated that DCD Tier 2, Sections 6.2.4.1.1 and 6.2.4.1.2 will be revised accordingly as indicated in the attached markups. The staff has reviewed the applicant response and DCD markups and finds it acceptable to meet the guidance in RG 1.11 and RG 1.141. The staff confirms that the RAI response markups were incorporated into Revision 1 of the DCD. Therefore, the staff considers RAI 357-8344, Question 06.02.04-12 resolved and closed.

Per SRP Section 6.2.4, Section II, Acceptance Criterion 9, the staff has reviewed the CIS design description as it relates to design requirements for location of the outside CIV as close to containment as practical. In Section 6.2.4.1, the applicant has stated that the CIS is designed in accordance with GDC 55, GDC 56, and GDC 57.

In Section 6.2.4.3, "Design Evaluation," the applicant stated that valves outside containment in systems designed in conformance with GDC 55, GDC 56, and GDC 57 are located as close to containment as practical. Table 6.2.4-1, "List of Containment Penetrations and System Isolation Positions," which contains a value, for each penetration, of the maximum length of pipe from containment to the outermost isolation valve, which will not be exceeded in further design.

The staff also reviewed DCD Tier 1, Table 2.11.3.2, "Containment Isolation System ITAAC." The staff issued RAI 357-8344, Question 06.02.04-11 and requested the applicant to describe any inspections, tests or Acceptance Criteria that will confirm that the as built piping distance will not exceed those listed in DCD Tier 2, Table 6.2.4.1. The staff also requested that the applicant indicate the associated penetration numbers in DCD Tier 1, Table 2.11.3-1, "Containment Isolation System Component List." In the response (ML16182A591) to RAI 357-8344, Question 06.02.04-11 that was received on June 30, 2016, the applicant stated that GDCs 55, 56, and 57 require that isolation valves located outside of containment should be located as close to containment as practical. The APR1400 design has incorporated this design concept into the location of the containment isolation valves. The applicant stated that acceptable containment isolation valve location is assured through the overall design and piping analysis program. The length of pipe between containment and the isolation valve indicated in DCD Tier 2, Table 6.2.4-1 does not necessarily represent a bounding condition for each piping line listed. Therefore, the applicant stated that including verification of as-built piping distances as a prescriptive ITAAC item is not meaningful nor practical for a subjective criteria such as locating isolation valves as close as practical to containment and the graded approach for piping analysis that has been implemented for the APR1400. In a revised response (ML17171A364), the length of pipe from containment to outer isolation valves in DCD Tier 2, Table 6.2.4-1 markups was included. The applicant also provided an example from an operational plant as a reference which was subject to change during the detailed design phase. The staff's review found the values in Table 6.2.4-1 similar to those in currently operating plants. Based on the above review and engineering judgement, the staff finds these values of length in Tier 2, Table 6.2.4-1, and Tier 1, Table 2.11.3-1 are reasonable for conformance with the requirements of GDCs 55, 56 and 57. In the response to RAI 357-8344, Question 06.02.04-11, KHNP did not provide an ITAAC item to ensure that the isolation valves in the as-built structure are located as close to the containment as practical. In an email from NRC to KHNP (ML18073A391), NRC notified KHNP that an ITAAC needs to be provided to meet the requirements of GDCs 55, 56, and 57. When provided, NRC staff will review and evaluate the ITAAC in Section 14.3 of this SER.

The staff confirmed changes to the DCD, Table 6.2.4-1 made in the response to the RAI 306-8240 Question 06.02.06-9 and RAI 357- 8344 Question 06.02.04-9 were incorporated in Revision 1 of the DCD. The applicant also indicated that some incorrect information in DCD Tier 2, Table 6.2.4-1 were identified in item nos. 48, 49, 50, 55, 161, 162, 163 and will be corrected as indicated in attachment 1 markups, associated with this response. The staff reviewed the proposed markup changes to DCD Tier 2, Table 6.2.4-1 and finds them to be acceptable.

Containment penetration numbers have been added in DCD Tier 1, Revision 1, Table 2.11.3-1 according to the response to the RAI 357-8344 Question 06.02.04-11, Revision 0. The applicant indicated that some incorrect information in DCD Tier 1, Table 2.11.3-1 regarding valves FW-V132, IA-V0020, PS-V0032, PS-V0258 and WI-V0015 will be corrected and missing manual valves VQ-V2014, V016 and V2024 will be added as indicated in attachment 2 associated with this response. The staff reviewed the proposed markup changes to DCD Tier 1, Table 2.11.3-1 and finds them to be acceptable.

The staff reviewed the applicant’s response and proposed DCD markups provided as evaluated above and finds the changes to be acceptable. Therefore, **RAI 357-8344, Question 06.02.04-11 response will be tracked as a confirmatory item.**

Pending the confirmatory item, the staff concludes that the SRP Section 6.2.4, Section II, Acceptance Criterion 9 is met regarding the requirements of GDC 55, GDC 56 and GDC 57 as it relates to locating CIVs outside containment as close to containment as practical.

6.2.4.4.2 Actuation and Control Features for Isolation Valves

As stated in SRP Section 6.2.4, Section II, Acceptance Criterion 8, in accordance with 10 CFR 50.34(f)(2)(xiv)(A), (TMI Action plan item II.E.4.2) all nonessential systems shall be automatically isolated upon initiation of an appropriate containment isolation signal. Nonessential systems are generally those that are neither ESF systems nor systems which accomplish a function similar to an ESF system. However, non-ESF and nonsafety grade systems shall be classified as essential, if their continued operation under post-accident conditions will improve the reliability of a safety function.

In DCD Tier 2, Section 6.2.4.1, the applicant stated that the CIS is designed in accordance with the TMI-related requirements of 10 CFR 50.34(f)(2)(xiv)(A) through (E).

Per NUREG–0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses,” Appendix B (SRP Section 6.2.4, Reference 25), all plants should give careful consideration to the definition of essential and non-essential systems, identify each system determined to be essential, identify each system determined to be non-essential, and describe the basis for selection of each essential system.

The staff reviewed the ESF classification and the actuation and control features (e.g., automatic, manual, or remote manual) for each isolation device as listed in APR1400 DCD Table 6.2.4-1, "List of Containment Penetrations and System Isolation Positions," and has determined that the applicant has classified each isolation device as either essential or non-essential and provides for automatic isolation of nonessential systems. As stated in Section 6.2.4.2, "System Design," of the DCD Tier 2, piping systems penetrating containment are provided with leak detection, isolation and containment capabilities. Therefore, the staff finds that the design has provisions for leakage detection for remote manual controlled systems, and automatic isolation of nonessential systems

Therefore, the staff finds that SRP Section 6.2.4, Section II, Acceptance Criterion 8 is met because the applicant classified systems penetrating the containment as essential or nonessential, provided for automatic isolation of nonessential systems, and has provided for detection of leakage for lines outside of containment. Therefore, the staff finds that the applicant meets the requirements of 10 CFR 50.34(f)(2)(xiv)(A) through (E) as it relates to the classification of systems penetrating the containment as either essential or nonessential, provisions for automatic isolation of nonessential systems, and provisions for detecting leakage from lines outside containment.

In Section 6.2.4.2 of the DCD Tier 2, the applicant stated that APR1400 DCD Table 6.2.4-1 provides the design information regarding provisions for isolating containment penetrations. The staff reviewed Table 6.2.4-1 to identify those containment penetrations that rely on relief valves as CIVs. Per SRP Section 6.2.4, Section II, Acceptance Criterion 7, relief valves may be used as isolation valves provided the relief setpoint is greater than 1.5 times the containment design pressure. The two shutdown cooling system (SCS) pump suction lines (DCD Tier 2, Table 6.2.4-1 Sheet 4 and Figure 6.2.4-1 Sheet 2) and the four main steam supply lines (DCD Tier 2, Table 6.2.4-1, Sheet 1-3) rely on relief valves for such purpose. Since the four main steam supply line relief valves remain closed at main steam pressure, their relief setpoint is much greater than 1.5 times containment design pressure, and the staff therefore finds them to be acceptable.

Per ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems," Section 4.7.4, "Relief Valves," when relief valves that discharge into containment are also used for containment isolation barriers, the discharge side of the valve shall be designed to withstand and be tested at the containment design pressure. The two SCS pump suction lines rely on relief valves for such purpose.

On January 5, 2016, the staff issued RAI 357-8344, Question 06.02.04-3, to request the applicant to clarify design requirements for relief valves used as CIVs. As per guidance in Sections 3.6.6 and 4.74 of ANSI N271-1976, for relief valves, the staff requested the applicant to provide in the DCD that the discharge of the relief valves in SCS suction lines are designed to withstand and be tested at the containment design pressure. In its response to RAI 357-8344, Questions 06.02.04-3, dated January 27, 2016 (ML16027A209), the applicant stated that the relief valves in the two shutdown cooling (SC) pump suction lines (DCD Table 6.2.4-1, Sheet 4, Item Nos. 18 and 19) protect the SCS from over-pressurization and also serve as CIVs in the backflow direction. The normal relief valve flow is discharged to the IRWST. The discharge side of the relief valves in SC pump suction lines is designed to be 200 psig, which is greater than the containment design pressure, and will be tested at a pressure greater than the containment design pressure. DCD Tier 2, Section 6.2.4 is revised to reflect the requirements in Sections 3.6.6 and 4.74 of ANSI N271-1976 and provided markups of the DCD Section associated with this response. The staff reviewed the applicant's response and finds it acceptable to meet the SRP Acceptance Criterion 7. The staff confirmed that the RAI response markups were incorporated into Revision 1 of the DCD. Therefore, the staff considers RAI 357-8344, Question 06.02.04-3 resolved and closed.

Based on the above, the staff finds that the SRP 6.2.4, Subsection II, Acceptance Criterion 7 is met as it relates to usage of relief valves as containment isolation barriers.

The staff reviewed provisions in the APR1400 design to address the requirements of GDC 54 as they relate to provisions in the design of the CIS to reduce the possibility of unintended isolation valve reopening following isolation.

The staff evaluated these provisions described in APR1400 DCD Section 6.2.4.1 against the acceptance criteria for those provisions contained in Section 6.2.4, Section II of the SRP. 10 CFR 50.34(f)(2)(xiv) requires control systems for automatic CIVs be designed for resetting the isolation signal without automatically reopening the valves.

In Section 6.2.4.1 the applicant stated that the CIS is designed in accordance with the TMI-related requirements of 10 CFR 50.34(f)(2)(xiv)(A) through (E). In Section 7.3.1.3, "Actuation Logic," the applicant stated that all actuation signals are latched at the train level and require manual reset.

Based on review of this design information in the DCD, the staff finds that the APR1400 design addresses the requirements of GDC 54, as they relate to provisions in the design in the CIS to reduce the possibility of unintended isolation valve reopening after valve reset. Therefore, the staff concludes that the SRP Section 6.2.4, Section II, Acceptance Criterion 19 is met.

The staff has reviewed provisions in the APR1400 design to address RG 1.155, "Station Blackout," Regulatory Position C.3.2.7 which states:

"The ability to maintain appropriate containment integrity" should be addressed. "Appropriate containment integrity" for SBO means that adequate containment integrity is ensured by providing the capability, independent of the preferred and blacked out unit's onsite emergency ac power supplies, for valve position indication and closure for CIVs that may be in the open position at the onset of an SBO. The following valves are excluded from consideration:

- Valves normally locked closed during operation,
- Valves that fail closed on a loss of power,
- Check valves,
- Valves in nonradioactive closed-loop systems not expected to be breached in a station blackout (this does not include lines that communicate directly with containment atmosphere), and
- Valves of less than three-inch nominal diameter.

The staff reviewed the design requirements of the CIS as described in APR1400 DCD Table 6.2.4-1 and Figure 6.2.4-1 against the acceptance criteria for those provisions contained in SRP Section 6.2.4 and, Section II of the SRP. The staff confirmed that, as described in DCD Sections 8.3, "Onsite Power Systems," and 8.4, "Station Blackout," there is an alternate ac power supply available within 10 minutes in accordance with Position C.3.2.5 of NRC RG 1.155, which will allow closure of CIVs which may be open at the onset of a SBO.

On January 5, 2016, the staff issued RAI 357-8344, Question 06.02.04-4, to request the applicant to clarify if all power operated CIVs have position indication in MCR, and SBO considerations for indication and closure. In its response to RAI 357-8344, Question 06.02.04-4 (M16103A519), dated April 12, 2016, the applicant stated that all power operated CIVs in Table 6.2.4-1 are provided with open/close position indication on the operator console and the safety console in the MCR. These consoles and safety-related component control cabinets are continuously powered from 120 Vac supplies, which are backed up by the station batteries or alternate AC power supply. Therefore, the MCR operator can monitor the CIVs position even if the onsite emergency ac power is not available. The closure for CIVs at the onset of a SBO is described in the attached table. DCD, Tier 2 Section 6.2.4.1.2, "Design Features," will be revised accordingly. In addition, DCD Tier 2, Table 9.2.2-6, "Component Cooling Water System Emergency Power Requirements," and Figure 9.2.2-1, "Essential Service Water System Flow Diagram," will be revised to correct the power segregation for Valves CC-231, 249, and 250. The staff has reviewed the applicant response and DCD markups and finds it acceptable because applicant provided information to clarify that all power operated CIVs have position indication in MCR, and SBO considerations for indication and closure. Therefore, the staff considers RAI 357-8344 Question 06.02.04-4 resolved and closed.

6.2.4.4.3 Normal and Fail Positions of Isolation Valves

Per SRP Section 6.2.4, Section II, Acceptance Criterion 10, the staff reviewed the CIS design as it relates to meeting the following criteria:

1. Upon loss of actuating power, automatic isolation valves should take the position of greatest safety, usually the post-accident position, or the usage of redundant isolation barriers ensure that the isolation function for the line is satisfied.

2. If a fluid system has no post-accident function, the isolation valves in the lines should be closed automatically.

3. For ESF or ESF-related systems, isolation valves in the lines may remain open or can be opened.

4. All power operated isolation valves should have position indication in the MCR.

The staff reviewed APR1400 DCD Table 6.2.4-1 and DCD Figure 6.2.4-1 to determine the position of automatic isolation valves upon loss of power in order to ensure the valve takes the position of greatest safety.

On January 5, 2016, the staff issued RAI 357-8344, Question 06.02.04-5 (Item Nos.31, 32, 33, 34, 69, and 70) and Question 06.02.04-6 (Item Nos. 35, 68 and 77) to request the applicant to justify fail-as-is open position of motor-operated isolation valves (MOVs), upon loss of power, when their post-accident position is closed in Table 6.2.4-1. Pursuant to the requirement of GDC 56, as it relates to the criteria that upon loss of actuating power, the position of automatic isolation valves should take the position of greater safety. The applicant was requested to explain how a failed open position of MOVs is the position of greater safety upon loss of power.

In its response to RAI 357-8344, Question 06.02.04-5, dated April 16, 2016 (ML16107001), the applicant stated the following:

(1) For Item Nos. 31 and 32, the two motor operated CIVs (296 and #297) for CCW supply to the letdown heat exchanger are powered from two separate Class 1E trains to ensure continuous and uninterrupted letdown cooling in the event of a single active failure in the electrical system. The Class 1E power train designation for the CIVs is summarized in Table 9.2.2-6. The backup power for these CIVs is the EDGs, which is to restart in accordance with the EDG load sequencing. Hence the current valve arrangement (i.e., fail-as-is position) is the position of greater safety upon loss of power only. If and when a loss of power is concurrent with a DBA, the CIVs are designed to close upon a CIAS. The two motored operated CIVs (#301 and #302) for CCW, which return from the letdown heat exchanger, follow the same design approach and are consistent with the operation of the CCW supply valves.

(2) For Item Nos. 33 and 34, the applicant stated that the motor operated CIV (#231) provides CCW supply to the RCP motor air cooler, RCP motor oil cooler, RCP oil cooler, and RCP high pressure cooler for RCP 1A, 1B, 2A and 2B. The two motor operated CIVs (#249 and #250) provide CCW return back to the CCW Quadrant A header, and are powered from two separate Class 1E trains to ensure continuous and uninterrupted RCP cooling in the event of a single active failure in the electrical system. The backup power for these CIVs is the EDGs, which is to restart in accordance with the EDG load sequencing. Hence the current valve arrangement (i.e., fail-as-is position) is the position of greater safety upon loss of power only. If and when a loss of power is concurrent with a DBA, the CIVs are designed to close upon an ESF-CSAS.

(3) For item Nos. 69 and 70, the applicant stated that the two motor operated CIVs (#431 and #432) are each designed to isolate the containment air sample line to one of two containment air radiation monitors. The radiation monitors detect the radioactivity of air particulate, gas, and iodine of the containment atmosphere. The particulate detector channel serves as the RCPB leak detection in accordance with NRC RG 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage.” The monitors continuously draw containment air in a closed piping system. The containment air passes through a separate sample line and is returned to the containment atmosphere. These CIVs are designed to fail-as-is (open position) in order to ensure that the continuous monitoring function is not interrupted due to power failure. Hence, the current valve arrangement (i.e., fail-as-is position) is the position of greater safety upon loss of power only. However, these valves are actuated to close upon a containment isolation actuation signal (CIAS) to ensure containment isolation. Containment isolation is confirmed and monitored after the CIAS occurs. These CIVs can be closed even if the normal power supplies fail because the valves are powered from 480 VAC, which is backed up by emergency power supplies and AAC power supplies.

The staff reviewed the applicant response and finds it acceptable because the applicant has justified fail-as-is open position of motor-operated isolation valves (MOVs), upon loss of power. RAI 357-8344, Question 06.02.04-5, is considered resolved and closed.

In its response to RAI 357-8344, Question 06.02.04-6, dated June 10, 2016 (ML16162A775), the applicant stated that the following:

(1) For Item No. 35, the CIV (CV-509) on the IRWST makeup line is normally closed and used infrequently during normal plant and shutdown operation to provide makeup to the IRWST when necessary. The containment isolation configuration for this penetration is

shown in Figure 6.2.4-1 Valve Arrangement 4 with a closed motor operated valve outside containment and a check valve inside containment. Table 6.2.4-1 Item 35 correctly depicts that the normal position for this valve as O/C meaning that it is infrequently opened during normal power and shutdown operation, the fail-safe position is as-is since it is a motor operated valve, and the accident position is closed for containment isolation. The valve does not have a safety function (e.g., there is no credit taken in the accident analysis) to open post-accident to fill the IRWST. If a LOCA occurs while CV-509 is in the open position filling the IRWST, CV-509 will be closed on a CIAS. If a loss-of-motive power to the CV-509 valve in the open position (e.g., loss of normal and emergency power), the inboard check valve CV-189 would provide the containment isolation function required for this penetration. Therefore, the specified configuration is in accordance with GDC 56 requirements.

(2) For Item No. 68, the applicant stated that, the motor operated CIV (WI-0015) is normally open to return plant chilled water (PCW) from containment ventilation units and it is automatically closed upon receipt of a CIAS when its power is available. The CIV (WI-0015) is designed to fail as-is (open position) upon loss of its power. It is noted that a pneumatically operated CIV (WI-0012), located downstream and outside the containment in the PCW return line from the containment ventilation units, is designed to fail in the closed position and is used as backup of the CIV (WI-0015). The CIV (WI-0012) is also automatically closed upon receipt of CIAS and it can provide containment isolation even if the CIV (WI-0015) fails-as-is (open) upon loss of its power. From the overall safety point of view, the use of different types of CIVs in the PCW return line, in which the motor operated CIV (WI-0015) is located inside the containment and the pneumatically operated CIV (WI-0012) is located outside the containment, provides greater safety in the event of an accident.

(3) For Item No. 77, the applicant stated that the motor operated CIV (GW-001) is normally open to receive radioactive gas vented from the reactor drain tank and is automatically closed upon the receipt of a CIAS when its power is available. The CIV (GW-001) is designed to fail-as-is (open position) during the loss of its power. It is noted that a solenoid operated CIV (GW-002), which has a fail closed position, is located downstream of the valve GW-001 and outside the containment. The CIV (GW-002) will be closed upon the receipt of CIAS in order to provide the containment isolation function even if the CIV GW-001 is in the open position due to a loss of its power. From the overall safety point of view, the use of different types of CIVs in the radioactive gas vented line, in which the motor operated CIV (GW-001) is located inside the containment and the pneumatically operated CIV (GW-002) is located outside the containment, provides greater safety in the event of an accident.

The staff reviewed the applicant response and finds it acceptable because the applicant has justified fail-as-is open position of above motor-operated isolation valves (MOVs), upon loss of power, when their post-accident position is closed in Table 6.2.4-1. RAI 357-8344, Question 06.02.04-6 is considered resolved and closed

Based on above review, the staff determines that the position of automatic isolation valves upon loss of power takes the position of greatest safety and because the staff's review confirmed that non motor-operated automatic isolation devices fail in the closed position upon loss of power source (air or electrical power).

MOVs are powered by Class 1E DC power. A single power system failure will not prevent closure of both isolation valves in a containment penetration. These features ensure single-failure-proof isolation capability for all penetrations that might be opened during operation.

The staff reviewed APR1400 DCD Table 6.2.4-1 and DCD Figure 6.2.4-1 to determine the post-accident automatic isolation of fluid systems that do not have a post- accident function. Using this table, the staff compared the ESF/ESF support designation for each penetration to the post-accident position of the valves. The staff determined that the APR1400 design ensures that fluid systems that do not have a post-accident function are automatically isolated.

The staff reviewed APR1400 DCD Table 6.2.4-1 and DCD Figure 6.2.4-1 to determine the post-accident position of ESF or ESF-related systems isolation valves upon loss of power. Using this table, the staff determined that engineered safety feature or engineered safety feature related system isolation valves remain open or can be opened upon loss of actuating power.

Based on review of APR1400 DCD Section 6.2.4.1, the staff finds that electrical redundancy via two independent power sources is provided on those lines that utilize two power operated isolation valves in series on the same penetration, and that all power-operated isolation valves have position indications in the MCR.

6.2.4.4.4 Initiating Variables for Isolation, Diversity, and Redundancy of Isolation Signals

Per SRP Section 6.2.4, Section II, Acceptance Criterion 12, there should be diversity in the parameters sensed for the initiation of containment isolation to satisfy the GDC 54 requirement for reliable isolation capability. The staff reviewed APR1400 DCD Section 7.3 in order to evaluate the requirements of GDC 54, as they relate to diversity in the parameters sensed for the initiation of containment isolation. The staff also reviewed APR1400 DCD Table 6.2.4-1 in order to verify that there was diversity in the means of isolation of lines that penetrate containment.

The staff found that the following signals initiate closure of CIVs, as indicated in APR1400 DCD, Tier 2, Section 7.3.1, “System Description,” and Table 7.3-3 “Monitored Variables for ESFAS Signals.”

- Low pressurizer pressure
- High containment pressure
- Manual initiation

RG 1.141 states that CIS designs should have diversity in the parameters sensed for the initiation of containment isolation, in accordance with SRP Section 6.2.4. Based on the above design information, the staff determined that SRP Section 6.2.4, Section II, Acceptance Criterion 12 is met because the staff finds that the design meets the GDC 54 requirements for reliable isolation capability as they relate to diversity in the parameters sensed for the initiation of containment isolation.

Per SRP Section 6.2.4, Section II, Acceptance Criterion 11, the staff reviewed DCD Tier 2, Section 7.3 and Tables 7.3-3 and 7.3-5A “NSSS ESFAS Setpoints and Margins to Actuations.” The staff reviewed the above in order to evaluate them against the requirements of 10 CFR 50.34(f)(2)(xiv) as they relate to the reduction of the containment set point pressure that initiates containment isolation for nonessential penetrations to the minimum value compatible with normal operating conditions. Containment isolation of nonessential penetrations is initiated via the Containment Isolation Phase A function upon receipt of a valid ECCS actuation signal, which can be generated by receipt of a valid high containment pressure signal.

The staff reviewed the set point, the range, accuracy response time as listed in APR1400 DCD Tier 2, Tables 7.3-3, and 7.3-5A, and compared this with the guidance for TMI Action Item II.E 4.2, contained in NUREG–0737, "Clarification of TMI Action Plan Requirements." TMI Item II.E.4.2 states that the containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions. It further states that the pressure setpoint selected should be far enough above the maximum expected pressure inside the containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification. Applicants for an operating license should use pressure history data from similar plants that have operated more than one year, if possible, to arrive at a minimum containment setpoint-pressure.

In APR1400 DCD Tier 2, Table 7.3-5A, the applicant indicated that an ESF actuation signal for ECCS function, and consequently, a Containment Isolation Phase A isolation signal is generated at a high containment setpoint pressure of 134 cm H_20 (1.9 psig) which is 1 psi above maximum expected containment pressure.

Per SRP Section 6.2.4, Section II, Acceptance Criterion 13, GDC 56 system lines which provide open paths from the containment to the environs (e.g. purge and vent lines) should be equipped with radiation monitors capable of isolating these lines upon a high-radiation signal, which should not be considered a diverse containment isolation parameter. The staff reviewed APR1400 DCD Sections 9.4.6, "Containment Ventilation System," 7.3, Table 6.2.4-1, and Figure 6.2.4-1 in order to evaluate if the APR1400 design meets this criterion.

The CIVs for the containment purge system will close within 5 seconds upon initiation of a valid containment purge isolation signal. The containment purge isolation signal is generated on receipt of two-out-of-four High Containment High Range area radiation signals. The signals have a response time of 15 seconds. The containment purge isolation signal is not considered and is not listed as a diverse containment isolation parameter. Based on this design information, the staff has determined that the design features of the purging/venting system minimize purging time and that there is a high degree of assurance that the purge system will isolate reliably under accident conditions. The system is equipped with radiation monitors capable of isolating the lines on a high radiation signal. Therefore, SRP Section 6.2.4, Section II, Acceptance Criterion 13 is met. Consequently, the staff finds the design meets the requirements of GDC 56, as they relate to the reliability of containment isolation of system lines which provide open paths from the containment to the environs.

6.2.4.4.5 Basis for Selection of Closure Time Limits

Per SRP Section 6.2.4, Section II, Acceptance Criterion 14, the staff has reviewed the APR1400 CIS design as it relates meeting the following criteria:

- Containment isolation closure times should be selected for rapid isolation of containment following postulated accidents.

- Isolation valve closure times should be five seconds or less for lines providing open paths from the containment to the environs. Radiological consequences and the effect on the containment back-pressure on the release of containment atmosphere should justify the selected valve closure time.

- BTP 6-4 provides additional guidance on the design and use of containment purge systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby, and hot shutdown).

- Containment purge valves that do not satisfy the operability criteria of BTP 6-4 must be sealed closed as defined in Section II.6 of this SRP Section during operational Conditions 1, 2, 3, and 4. Furthermore, closure of these valves must be verified at least every 31 days. These requirements should be incorporated into the TS for plant operation.

The staff reviewed the basis of the APR1400 CIV closure times stated in APR1400 DCD Section 6.2.4.2 and the values of the closure time for each valve listed in Table 6.2.4-1, against guidance contained in Section 6.2.4, Section III of the SRP and the guidance in paragraph 4.4.4, "Valve Closure Time," of ANSI N271-1976, endorsed by RG 1.141.

Based on review of information provided in the DCD Table 6.2.4-1, the staff has determined that the applicant's stated CIV closure time design basis and valve closure times are in accordance with paragraph 4.4.4, of ANSI N271-1976 and thus RG 1.141 for CIVs other than purge, vent or other valves which may be open during operation and provide an open path from the containment atmosphere to the environs.

The piping penetration that provides a direct path to the atmosphere consists of the two 8 in (20 cm) low volume purge system valves. The isolation valves in these lines are specified as having a five-second closure time. The staff noted that this closure time is consistent with the assumptions and criteria for radiological dose analyses used in APR1400 DCD Chapter 15 analyses.

The containment high volume purge supply and exhaust system consists of 48 inch isolation valves which are sealed closed in plant Modes 1, 2, 3, and 4 and are verified sealed closed every 31 days as required by proposed TS in Chapter 16 of the DCD. These valves are also specified as having a five second closure time. Based on review of information provided in the DCD, the staff determined that containment purge system is in compliance with BTP 6-4.

Based on compliance with RG 1.141, as described above, the staff has found that SRP Section 6.2.4, Section II, Acceptance Criterion 14 is met. Consequently, the staff finds that the APR1400 containment isolation closure times are in accordance with GDC 54 requirements as they relate to the rapid isolation of containment following postulated accidents.

6.2.4.4.6 Sealed Closed Barriers

Per SRP Section 6.2.4, Section II, Acceptance Criterion 6, the staff reviewed provisions for systems which utilize blank flanges as containment isolation barriers. In APR1400 DCD Tier 2, Section 6.2.4.2, the applicant stated that Table 6.2.4-1 lists GDC 55 and GDC 56 systems which utilize blank flanges as containment isolation barriers and provides justification for their use. The staff has reviewed this listing and finds the following flanged or sealed penetrations:

- The personnel airlock (APR1400 DCD Tier 2, Figure 6.2.4-1, “Containment Isolation Valve Arrangement,” Sheet 11) consists of two flanged doors in series. The doors are interlocked to ensure that both doors are not opened simultaneously. The airlock is designed to seismic Category 1 and Quality Group B standards. Each door is provided with a testable seal. The airlock has a design temperature and pressure rating at least equal to that for the containment.

- The equipment hatch (APR1400 DCD Tier 2, Figure 6.2.4-1, Sheet 12) consists of a single flanged door. The door is not opened during normal or accident conditions. The hatch is designed to seismic Category 1 and Quality Group B standards. The hatch is provided with a testable seal. The hatch has a design temperature and pressure rating at least equal to that for the containment.

- The fuel transfer tube (APR 1400 DCD Tier 2, Figure 6.2.4-1, Sheet 10) consists of a single flange. The penetration is not opened during normal or accident conditions. The tube is designed to seismic Category 1 and Quality Group B standards. The tube is a flanged closure supplied with a testable seal. The tube has a design temperature and pressure rating at least equal to that for the containment.

- There are 66 electrical penetrations (APR1400 DCD Tier 2, Figure 6.2.4-1, Sheet 13). The penetrations are designed to seismic Category 1 and Quality Group B standards. The penetrations have a design temperature and pressure rating at least equal to that for the containment.

The applicant has not shown any spare penetrations. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-9, to show all the penetrations in the DCD. The response to RAI 306.8240, Question 06.02.06-9 is discussed in SER Section 6.2.6. On January 5, 2016, the staff also issued RAI 357-8344, Question 06.02.04-7, to clarify administrative controls and leak testing provisions for flanged closures, personnel airlock, equipment hatch, and fuel transfer tube. In the response to RAI 357-8344, Question 06.02.04-7, on June 30, 2016 (ML16182A591), the applicant stated that DCD Subsection 6.2.4 will be revised to clarify the administrative controls applicable to the flanged closures for personnel airlock, equipment hatch and fuel transfer tube. The administrative controls procedures provide assurance that these components will operate as designed during normal and post-DBA conditions.

For Leak Testing Provisions, the applicant clarified that DCD Subsection 6.2.4.4 describes the testing and inspection requirements for containment leakage testing. The leak test for the fuel transfer tube, equipment hatch, and personnel airlock are described in DCD subsection 14.2.12.1.120 through 14.2.12.1.122. The pressure test taps are provided for testing of the annulus between the seals. Air or nitrogen is used as a pressure medium for the pressure test. The fuel transfer tube blind flange, equipment hatch and personnel airlock are Type-B leak rate tested in accordance with ANSI/ANS 56.8-1994

The staff reviewed the applicant’s response as discussed above and the corresponding changes to Revision 1 of the DCD, specifically Subsections 6.2.4 and 6.2.4.2 and Figure 6.2.4-1, which describe administrative controls and leak testing provisions for the personnel airlock, equipment hatch, and fuel transfer tube and finds it acceptable to meet the SRP acceptance criterion 6. Therefore, the staff considers **RAI 357-8344, Question 06.02.04-7 resolved and closed.**

Based on the above review, the staff concludes that the sealed closed barriers meet SRP acceptance criterion 6 and conform to ANSI N271-1976, Paragraph 4.10, endorsed by RG 1.141.

6.2.4.4.7 Use of Closed Systems as Isolation Barriers

Per SRP Section 6.2.4, Section II, Acceptance Criterion 15, the staff has reviewed the use of closed systems inside containment as one of the isolation barriers as they relate to the following criteria:

1. The system does not connect with either the RCS or the containment atmosphere.

2. The system is protected against missiles and pipe whip.

3. The system is designated seismic Category I.

4. The system is classified Quality Group B.

5. The system is designed to withstand temperatures equal to at least that of the containment design.

6. The system is designed to withstand the external pressure from the containment structure acceptance test.

7. The system is designed to withstand the LOCA transient and environment.

APR1400 DCD Table 6.2.4-1 lists those systems (or portions of systems) that meet the criteria of GDC 57. These are as follows:

1. Main steam system (MSS) steam lines from SG

2. Steam generator blow-down system (SGBDS) SG blowdown lines

3. Main feedwater system (MFWS) water lines to SG

The staff reviewed the design of those systems, as described in Chapters 3, “Design of Structures, Systems, Components and Equipment,” Chapter 9, “Auxiliary Systems,” and Chapter 10, “Steam and Power Conversion System,” of the APR1400 DCD, with acceptance criteria for those systems contained in Section 6.2.4, Section II of the SRP.

DCD Tier 2, Section 3.2.2, “System Quality Group Classification,” lists the classification of mechanical fluid systems, components, and equipment. The staff reviewed this list and confirmed that the specified portions of the above systems are designated seismic Category 1 and classified Quality Group B. The staff determined that the systems do not connect with either the RCS or the containment atmosphere. The staff noted that Section 3.1.4 of the DCD stated that the safety-related SSCs are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the normal operation, maintenance, testing, and postulated accidents, including LOCAs. DCD Tier 2, Section 3.5, “Missile Protection,” states that safety-related SSCs are identified in DCD Tier 2, Section 3.2 and Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” and are protected from missiles. The staff noted that DCD Tier 2, Section 10.3.1.1, “Safety Design Bases,” specified the scope of the safety-related portion of the MSS, and compliance with GDC 4. The staff has concluded that the safety-related portion of the MSS includes that portion of the system that penetrates containment. The staff noted that DCD Tier 2, Section 10.4.7.1.1, “Safety Design Basis,” specifies the scope of the safety-related portion of the MFWS, and compliance with GDC 4. The staff has concluded that the safety-related portion of the MFWS includes that portion of the system that penetrates containment. The staff noted that DCD Tier 2, Section 10.4.8.1.1, “Safety Design Bases,” specifies the scope of the safety-related portion of the SGBDS, and compliance with GDC 4.

The staff has concluded that the safety-related portion of the SGBDS includes that portion of the system that penetrates containment. The staff noted that DCD Tier 2, Section 9.2.2.1.1, “Safety Design Basis,” specifies the scope of the safety-related portion of the CCWS, and compliance with GDC 4. The staff has concluded that the safety-related portion of the CCWS includes that portion of the system that penetrates containment. The staff noted that DCD Section 6.2.4.1 specifies that the containment isolation barriers are required to be protected from missiles.

The staff has concluded that isolation barriers are located behind missile barriers and that pipe whip was considered in the design of pipe restraints. The staff also concluded that the Sections of piping penetrating the containment to the isolation barriers, including the piping between isolation valves, are designated seismic Category I, and are therefore adequately protected from missiles and pipe whip.

Based on review of the above design information contained in the DCD, the staff has concluded that SRP Section 6.2.4, Section II, Acceptance Criterion 15, is met. Therefore the use of closed systems inside containment as one of the isolation barriers in the APR1400 CIS design, meets the requirements of GDC 1, GDC 2, GDC 4, and GDC 16 as these GDC apply to this purpose. Additional staff findings regarding compliance of these systems to these GDC are in Chapters 3, 9, and 10 of this SE.

6.2.4.4.8 Protection of CISs against Loss of Function as a Result of Missiles, Pipe Whip, and Natural Phenomena

As mentioned in Section 6.2.4.4.7 of this SER, “Use of Closed Systems as Isolation Barriers,” the staff confirmed that the CIS design bases include protection from missiles, pipe breaks, earthquakes, fire, internal and external flooding, ice, wind, and tornados. Other sections of this SE discuss specific features and design criteria for the protection of systems, structures, and equipment from these phenomena.

6.2.4.4.9 Environmental Conditions in the Vicinity of Containment Isolation Components

Containment isolation equipment may be subject to potentially harsh conditions resulting from pressure, temperature, flooding, jet impingement, radiation, missile impact, and seismic response. The staff review confirmed that the APR1400 CIS has been properly classified to ensure that protection from these environmental hazards is encompassed by the mechanical and electrical design bases and quality standards of the isolation system. Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” of this SE discusses the staff’s review of the environmental qualification of the APR1400 SSCs, including containment isolation equipment.

6.2.4.4.10 Mechanical Engineering Design Criteria Applied to the Containment Isolation System, Structure, and Components

Per SRP Section 6.2.4, Section II, Acceptance Criterion 16, the staff has reviewed the reliability and performance considerations used in the design of the APR1400 isolation barriers as they relate to the requirements of GDC 1, GDC 2, GDC 4, and GDC 54. The design criteria for components performing a containment isolation function, including the isolation barriers and the piping between them or the piping between the containment and the outermost isolation barrier was found acceptable if:

1. Group B quality standards, as defined in RG 1.26, apply to the components, unless the service function dictates that Group A quality standards apply.

2. The components are designated seismic Category I in accordance with RG 1.29, "Seismic Design Classification."

The staff reviewed the design of those systems and portions of those systems that serve as containment isolation barriers as described in APR1400 DCD Section 3.2 and Table 3.2-1.

The staff has determined that:

1. Group B quality standards, as defined in RG 1.26, apply to the components, unless the service function dictates that Group A quality standards apply.

2. The components are designated seismic Category I in accordance with RG 1.29.

Therefore, the staff determined that SRP Section 6.2.4, Section II, Acceptance Criterion 16, is met. Consequently, the staff found that the requirements of GDC 1, GDC 2, GDC 4, and GDC 54, have been met as they relate to the inclusion of appropriate reliability and performance considerations in the APR1400 design that reflect the safety importance of containment capability under accident conditions because it meets RG 1.26 for quality Groups A and B standards, and RG 1.29 for seismic Category I design, for components.

6.2.4.4.11 Provisions for Alerting Operators of the Need to Actuate Manual Isolation Devices in the Event of Leakage or an Accident

Per SRP Section 6.2.4, Section II, Acceptance Criteria 2 and 3, the staff reviewed provisions to detect leakage outside containment for essential system lines that penetrate containment. In Table 6.2.4-1, "List of Containment Penetrations and System Isolation Positions," in the DCD application, the applicant has provided a listing of containment penetrations and system isolation positions. This listing includes provisions for containment isolation by means of remote-manual valves in engineered safety feature related systems including some valves in the SIS, SCS, MSS, CSS, and CCWS. This listing also includes provisions for containment isolation by means of remote-manual valves in lines in systems needed for safe shutdown of the plant including some valves in the CVCS.

As discussed in APR1400 DCD Section 6.2.4.2, "System Design," the APR1400 design provides means of detection of possible leakage from these lines.

Per SRP Section 6.2.4, Section II, Acceptance Criterion 17, the staff has reviewed provisions in the APR1400 design to address the requirements of GDC 54, as they relate to reliable isolation capability. For remote manual isolation valves, the design of the CIS was found acceptable if there are provisions to allow the operator in the MCR to know when to isolate fluid systems equipped with remote manual isolation valves.

On January 5, 2016, the staff issued RAI 357-8344, Question 06.02.04-8, to request the applicant to provide information on the provisions to alert the operator to isolate remote manual containment isolation systems. For each containment penetration listed in DCD Tier 2, Table 6.2.4-1 that is equipped with remote manual CIVs, the applicant was requested to provide details as to what provisions are provided to alert the operator of the need to isolate fluid systems equipped with these remote manual isolation valves.

In its response to RAI 357-8344, Question 06.02.4-8, dated June 10, 2016 (ML16162A775), the applicant indicated the remote manual CIVs in DCD Table 6.2.4-1 and their valve position status

The applicant stated that the systems including remote manual valve for containment isolation are the MSS, SIS, chemical and volume control system (CVCS), containment monitoring system (CMS), and IWSS.

The applicant stated that remote manual CIVs in the MSS are located on the main steam line. The main steam atmospheric dump valves (MSADVs) (MS-101 through MS-104) are closed during normal operation. These valves are provided to allow cooldown of the SG by discharging steam to the atmosphere when the MSIVs are closed or when the main condenser is not available as a heat sink. Therefore, there are no remote manual CIVs in the MS that need to be manually closed during normal operation or accident condition. Remote manual CIVs are provided in SIS, SCS, and CVCS. Those CIVs are located on the SIP suction lines, SIP discharge lines, SI hot leg lines, and SIP mini-flow lines of the SIS; SCP suction lines of the SCS; and charging and RCP seal injection lines of the CVCS.

The applicant stated that some remote manual CIVs in the SIP miniflow lines and the SIP suction lines are maintained in the locked open (LO) position (SI-302, SI-303, SI-304, SI-305) and some in the closed position (SI-300, SI-301) during normal operation to be in standby for the SI injection, as well as during accident conditions, in order to inject coolant into the RCS. The operator may close these valves with the CIAS after deciding that SI is no longer required.

On the other hand, those lines in the SIS and SCS, except for the SIP miniflow and SIP suction lines, are normally isolated from the RCS by maintaining the remote manual CIVs closed during normal operation. However, those CIVs can be opened by the operator during an accident, if necessary. Therefore, there are no CIVs in the SIS and SCS that need to be closed by the operator during normal operation or accidents.

The applicant stated that valves CV-524 in the CVCS charging line and CV-255 in the RCP seal injection line are in the open position during normal operation. These valves are also maintained in the open position during an accident condition to help maintain RCS coolant inventory and RCP seal integrity by establishing flow through these valves with an available charging pump or auxiliary charging pump. However, the operator can manually close these valves (as needed) in the MCR if the charging pump and auxiliary charging pump become unavailable by checking the indications for charging flow, charging line pressure, and seal injection flow.

In addition, valves CM-017 through CM-022 are solenoid-operated instrument isolation valves located outside of the containment building. They isolate the instrument sensing lines that contain the containment air. The instruments monitor the containment air pressure to generate engineered safety features actuation signals (ESFAS) such as CIAS, CSAS, MSIS, and SIAS if the containment air pressure exceeds the ESFAS actuation setpoint. Therefore, these CIVs are open during normal operation and during design basis accident DBA conditions. These valves are designed to the fail-open position so that the instruments can continuously monitor the containment air pressure. These valves are closed during instrument maintenance. The CIVs are manually opened and closed only by the MCR operator. No field instruments are used to provide interlock signals to these CIVs.

The applicant also stated that valves IW-010 through 035 are solenoid-operated instrument isolation valves located outside of the containment building. They isolate the instrument sensing lines that contain containment air or the IRWST borated water. The instruments connected to valves IW-010, 011, and 022 through 027 are to monitor the IRWST water level during normal operation and DBA conditions. The instruments connected to valves IW-016 and 017 are to monitor containment leakage during normal operation. The instruments connected to valves IW-012 through 015 and 028 through 031 are to monitor the HVT water level during DBAs. The instruments connected to valves IW-018 through 021 and IW-032 through 035 are to monitor the reactor cavity water level during DBAs. Therefore, these CIVs are in the open position during normal operation and under DBA conditions. These valves are designed to the fail-open position in order to continuously monitor the IRWST water level or containment leakage. These valves are closed during instrument maintenance. The CIVs are manually opened and closed only by the MCR operator. No field instruments are used to provide interlock signals to these CIVs.

In addition, valves MS-016 and 017 are not only remotely and manually operated, but they are also automatically actuated by the main steam isolation signal (MSIS). Table 6.2.4-1 will be revised to correct the actuation signal type for valves MS-016 and 017.

The applicant stated that DCD Tier 2, Section 6.2.4.2 and Table 6.2.4-1, will be revised to incorporate the above discussion, as indicated in the attachment associated with this response.

The staff reviewed the applicant's response and DCD markups, and finds it acceptable because applicant provided details on the provisions that are provided to alert the operator of the need to isolate fluid systems equipped with these remote manual isolation valves as per SRP 6.2.4, Section II, Criterion 17. Therefore, the staff considers RAI 357-8344, Question 06.02.04-8 resolved and closed.

The staff evaluated these provisions as described in DCD Section 6.2.4.2 against the acceptance criteria for those provisions contained in Section 6.2.4, Subsection II of the SRP. In this Section of the DCD, the applicant indicated that detection of possible leakage is provided for those systems where remote-manual isolation valves are employed.

6.2.4.4.12 Provisions for, and Technical Specifications Pertaining to Operability and Leakage Rate Testing of Isolation Barriers

Per SRP Section 6.2.4, Section II, Acceptance Criterion 18, the staff has reviewed provisions in the APR1400 design to address the requirements of GDC 54, as they relate to provisions for operability testing of the CIVs and leakage rate testing of the isolation barriers. In order to permit periodic Type A, Type B, and Type C testing of the containment and its piping penetrations, test connections should be provided on the containment and on penetrations in order to permit application and measurement of test air pressure and venting of leakage air.

The staff evaluated these provisions as described in APR1400 DCD Section 6.2.4.4 and in Figure 6.2.4-1. In the DCD Section 6.2.4.4, the applicant stated that test connections are provided for 10 CFR Part 50, Appendix J, Type C leakage rate testing. APR1400 DCD Tier 2, Chapter 16, Section 3.6 specifies periodic Type B and C leakage rate testing. DCD Tier 1, Section 2.11.3, "Containment Isolation System," contains inspection, tests and analysis and acceptance criteria to ensure that as-built containment isolation leakage is within design limits.

Based on the review of information provided in the DCD sections listed above, the staff found that additional information is needed in order to evaluate if requirements of GDC 54, as they relate to the ability to test the operability of isolation barriers are met, and to determine if the valve leakage is within limits. The staff requested additional information to confirm that test, vent, and drain connections are provided at suitable locations.

On January 5, 2016, the staff issued RAI 357-8344, Question 06.02.04-9 and Question 06.02.04-10 to request the applicant (1) to provide a table describing the provisions for individual leakage testing of the isolation barrier, (2) to clarify use of containment vent and purge valves and accommodations for seal replacement if supplied. The applicant provided a response (ML16182A591) to RAI 357-8344, Question 06.02.04-9 describing the provisions for individual leakage rate testing of the isolation barrier. In Revision 1 to the DCD Tier 2, Figure 6.2.4-1 was revised to include the leak rate test connections and locations for associated valves, vents, and drain connections along with the locations of the applicable CIVs. Table 6.2.4-1 contains a list of all containment isolation valves including a designation of the test type for each individual valve. Table 6.2.4-1 also includes reference to the valve arrangements depicted in Figure 6.2.4-1. A detailed discussion regarding the Type C test is presented in DCD Tier 2, Subsection 6.2.6.3. The staff reviewed the applicant response and revised DCD Figure 6.2.4-1 and Figure 2.11.3-1 for individual CIVs leakage measurement and finds it acceptable for providing provisions describing individual leakage testing of isolation barriers. Therefore, the staff considers RAI 357-8344, Question 06.02.04-9 resolved and closed.

The applicant provided its response (ML16057A075) to RAI 357-8344, Question 06.02.04-10, on February 26, 2016, to clarify use of containment vent and purge valves and accommodations for seal replacement. The applicant stated in its response that DCD Tier 2, Section 9.4.6.4.2, "Reactor Containment Building Purge System" will be revised to describe the use of resilient seals on CIVs in the RCP purge system and provisions for seal replacement. The applicant provided DCD Tier 2, Section 9.4.6.4.2 markups, which stated that resilient seals of CIVs are replaced according to the manufacturer's recommendations and when required by leakage rate testing. Testing of CIVs to verify operability and ability to meet closing requirements is described in Section 6.2.4. The staff reviewed the applicant response and revised markups and finds it acceptable to describe the use of resilient seals on CIVs in the RCP purge system and provisions for seal replacement. The staff confirmed that the RAI response markups has been incorporated into Revision 1 of the DCD. Therefore, the staff considers RAI 357-8344, Question 06.02.04-10 resolved and closed.

Based on the above, the staff finds that SRP Section 6.2.4, Section II, Acceptance Criteria 18, is met to address the requirements of GDC 54, as they relate to provisions for operability testing of CIVs and leakage rate testing of the isolation barriers.

6.2.4.4.13 Calculation of Containment Atmosphere Released before Isolation Valve Closure for Lines that Provide a Direct Path to Environs

Per SRP Section 6.2.4, Section II, Acceptance Criterion 22, the staff reviewed the APR1400 design to address the extent to which 10 CFR Part 50, Appendix K is used for the determination of the extent of fuel failure (source term) in the radiological calculations.

The staff reviewed APR1400 DCD Tier 2, Section 11.1.2, "Expected Source Term," which states that the realistic reactor coolant source term represents the expected average concentrations of radionuclides contained in the reactor coolant and the secondary coolant. These concentrations are calculated according to the modeling procedures in ANSI/ANS-18.1, "Radioactive Source Term for Normal Operation for Light Water Reactors." The reference plant values provided in

ANSI/ANS-18.1 were adjusted to be consistent with the APR1400 plant values listed in Table 11.1-9, “Expected Specific Activities of Reactor Coolant during Normal Operation (Core Power: 3,983 MWt, No Gas Stripping),” by using adjustment factors. The staff found this calculation method conforms to the guidance in RG 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors.”

The staff reviewed DCD Tier 2, Chapter 15, “Transient and Accident Analyses.” Per Table 15.6.5-13, “Major Input Parameters Used in Radiological Consequences Analysis for Large Break LOCA,” the assumed radionuclide inventory in the reactor coolant is based on an assumed one percent fuel defect and is listed in Table 11.1-2, “Maximum Reactor Coolant Fission Product Source Term,” of Tier 2 of the DCD.

The piping penetration that provides a direct path to the atmosphere is the two 20 cm (8 inch) low volume purge system valves. The isolation valves in these lines are specified as having a 5 second closure time. This closure time is consistent with the assumptions and criteria for radiological dose analysis used in the DCD Tier 2, Chapter 15.

Based on this review, the staff finds that the APR1400 DCD utilizes 10 CFR Part 50, Appendix K, for the determination of the extent of fuel failure (source term) in the radiological calculations and therefore finds that SRP Section 6.2.4, Section II, Acceptance Criterion 22 is met.

6.2.4.4.14 TMI Item II.E.4.4 Vent/Purge Valve Positions

Per SRP Section 6.2.4, Section II, Acceptance Criterion 20, the staff has reviewed provisions in the APR1400 design to address the requirements of 10 CFR 50.34(f)(2)(xv) purging requirements as they relate to the regulatory guidance of BTP 6-4, “Containment Purging During Normal Plant Operations.”

In APR1400 DCD Tier 2, Section 1.9.2, “Conformance with Standard Review Plan,” the applicant has stated that the APR1400 design complies with SRP Section 6.2.4 Acceptance Criterion 14 (BTP 6-4, “Containment Purging during Normal Plant Operations”). The staff reviewed the design requirements of the containment purge system as described in Chapter 9, “Auxiliary Systems,” Section 9.4.6, “Reactor Containment Building HVAC System and Purge System,” of the DCD and has confirmed that:

- The performance and operability assurance program for the containment purge system is consistent with SRP Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.”
- The number of supply and exhaust lines is limited to one supply and one exhaust line.
- The size of the lines do not exceed 20 cm (8 inches) in diameter.
- The containment isolation provisions for the purge system meet the standards appropriate for engineered safety features.
- II&C systems isolating the purge system lines are independent and actuated by diverse parameters.
- Purge system isolation valve closure times do not exceed 5 seconds.

- Isolation valve closure is not prevented by debris, which could become entrained in the escaping air and steam.

- The purge system is not designed for temperature and humidity control within the containment.

- The need for purging of the containment is minimized by the presence of other containment atmospheric cleanup systems within containment.

- The containment purge valves isolation function and leakage rate are periodically tested during reactor operation.

The bases for TS 3.6.3 indicate that the 20 cm (8 inch) low volume purge system valves will be opened as needed in Modes 1, 2, 3, and 4. BTP 6-4 guidance states that the opening of valves that provide a direct path from the containment atmosphere to the environs should be minimized during power operation. The staff also notes that the plant design has very few safety-related items in containment that would require containment entry while at power.

Therefore, use of venting or purging should occur infrequently. As a result, the containment vent/purge system should only be used for containment pressure control, or air quality considerations for personnel entry, or TS surveillances. TS SR 3.6.3.2 includes this restriction.

As discussed above, the staff found that the APR1400 design utilizes the regulatory guidance of BTP 6-4 and therefore SRP Section 6.2.4, Section II, Acceptance Criterion 20 is met and complies with 10 CFR 50.34(f)(2)(xv) purging requirements.

6.2.4.4.15 TMI Item II.B.8 Three-Foot Diameter Containment Penetration

The staff has reviewed provisions in the APR1400 design to address the requirements of 10 CFR 50.34(f)(3)(iv) for provisions for a dedicated 0.9 meter (3 foot) diameter containment penetration so as not to preclude later installation of a ventilated containment system. The staff evaluated these provisions as described in APR1400 DCD Tier 2, Section 19.2.3.3.8, “Other Severe Accident Mitigation Features,” with the requirements as stated in the regulation. In APR1400 DCD Tier 2, Section 1.9, Table 1.9.4, the applicant has stated that the requirements of TMI Item II.B.8 are addressed in Section 19.2.3.3.8. In DCD Tier 2, Section 19.2.3.3.8, the applicant stated that the containment high volume purge opening of 48 inch satisfies the requirement of the regulation. The staff reviewed the design description of the high volume purge opening and has concluded that the containment high volume purge opening of 48 inches satisfies the 36 inch regulatory requirement in 10 CFR 50.34(f)(3)(IV).

6.2.4.4.16 Minimization of Contamination

In consideration of 10 CFR 20.1406, “Minimization of Contamination,” the staff reviewed the APR1400 design in order to determine how the design will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize to the extent practicable the generation of radioactive waste. APR1400 DCD Tier 2, Table 12.4-10, “Regulatory Guide 4.21 Design Objective and Applicable DCD Section Information for minimizing Contamination and Generation of Radioactive Waste,” describes the provisions related to the CIS for providing for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage. The piping systems penetrating the containment are provided with leak detection, isolation, and containment capabilities. These piping systems are designed with the capability to test, periodically, the operability of the isolation valves and associated apparatus and determine if valve leakage is within acceptable limits.

The staff finds that these design provisions meet the requirement of 10 CFR 20.1406 and are consistent with guidelines of RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” since the existence of leak detection systems ensure that operating personnel will be alerted in a timely manner. Prompt action by plant personnel to address detected leakage will minimize radioactive contamination of spaces containing CIS components. Section 12.3-12.4, “Radiation Protection Design Features,” of this SE further addresses the APR1400 design in accordance with 10 CFR 20.1406.

6.2.4.5 Inspection, Tests, Analysis, and Acceptance Criteria (ITAAC)

DCD Tier 1, Section 2.11.3, “Containment Isolation System,” describes the design description and safety-related functional requirements and location of all the components in the CIS in Table 2.11.3-1, “Containment Isolation System Components List,” and as shown in Figure 2.11.3-1. The ITAAC associated with the design commitment are provided in Table 2.11.3-2, “Containment Isolation System ITAAC.” The staff evaluation of the ITAAC will be addressed in SER Section 14.3.11, “Containment Systems ITAAC.”

6.2.4.6 Combined License Information

There are no COL item numbers in DCD Tier 2, Table 1.8-2, related to this review area. The staff finds that there are no COL information items that are needed for this review area.

6.2.4.7 Conclusion

The staff has reviewed the CIS in accordance with SRP Section 6.2.4 and BTP 6-4. Pending confirmatory item Question 06.02.04-11, the staff concludes that the CIS complies with the requirements of GDC 1, GDC 2, GDC 4, GDC 16, GDC 54, GDC 55, GDC 56 and GDC 57, 10 CFR 50.63, and 10 CFR 20.1406 as related to containment isolation.

6.2.5 Combustible Gas Control in Containment

6.2.5.1 Introduction

Control of combustible gases in containment is described in DCD Tier 2, Section 6.2.5, “Containment Hydrogen Control System (CHCS).” The CHCS mitigates the consequences of postulated accidents by mixing, monitoring, preventing, or removing combustible gas concentrations that may be released into the containment atmosphere in the event of a significant beyond DBA. Following a postulated accident, hydrogen and oxygen may accumulate inside the containment. Combustible gas is predominantly generated within the containment as a result of reactions between the fuel-clad and reactor coolant and also by interactions between the molten core and concrete. Since significant amounts of combustible gas can be generated, these gases shall be mixed and removed to prevent the uncontrolled hydrogen/oxygen recombination (detonation) and possible damage to the containment structure.

6.2.5.2 Summary of Application

DCD Tier 1: Information for the CHCS is provided in DCD Tier 1, Section 2.11.4, “Containment Hydrogen Control System.” CHCS relies on nonsafety-related hydrogen igniters (HIs) as well as nonsafety-related passive autocatalytic recombiners (PARs) for the control of hydrogen in containment. The major equipment and the associated locations for the CHCS are shown in DCD Tier 1, Table 2.11.4-1, “Containment Hydrogen Control System Components List” and Table 2.11.4-2 “Containment Hydrogen Control System.” ITAAC are provided in DCD Tier 1, Table 2.11.4-3 “Containment Hydrogen Control System ITAAC.” The hydrogen monitors measure the hydrogen concentration in containment both for DBAs and for beyond DBAs.

DCD Tier 2: DCD Tier 2, Section 6.2.5 information is summarized as follows:

The CHCS and the Containment Hydrogen Monitoring System (CHMS) collectively are designed to mix, monitor, prevent, or remove combustible gases from the containment atmosphere, thereby preserving containment integrity and mitigating the consequences of core damage accidents, including severe accidents. The CHCS consists of PARs, and HIs. The CHCS component locations and sizes are given in DCD Tier 2, Table 6.2.5-1, “CHCS Location of PARs and HIs.” The CHMS provides continuous hydrogen concentrations in containment and in the IRWST during severe accidents. The monitors are designed to measure 0-15 percent volume of hydrogen in the pressure range (-) 5 psig to the maximum containment design pressure. An analysis of the design of the CHMS is described in DCD Tier 2, Table 6.2.5-2, “Hydrogen Monitoring System Failure Modes and Effects Analysis.” The system arrangement for CHMS is shown on DCD Tier 2, Figure 6.2.5-2, “Containment Monitoring System Diagram.”

ITAAC: The ITAAC associated with combustible gas control in containment are given in DCD Tier 1, Section 2.11.4, “Containment Hydrogen Control System,” and Table 2.11.4-3, “Containment Hydrogen Control System ITAAC.”

6.2.5.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG–0800, Section 6.2.5 and are summarized below. Review interfaces with other SRP Sections can also be found in NUREG–0800, Section 6.2.5.

1. The regulations in 10 CFR 50.44(c), “Requirements for Future Water-Cooled Reactor Applicants and Licensees,” as it relates to PWR plants being designed to accommodate hydrogen generation equivalent to 100 percent fuel clad-coolant reaction while limiting containment hydrogen to less than 10 percent and maintain containment structural integrity and appropriate accident mitigating features; and the capability to ensure a mixed atmosphere during design-basis and significant beyond DBAs.

2. The regulations in 10 CFR 52.47(b)(1), “Contents of applications; technical information,” which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the certified design is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations.

6.2.5.4 Technical Evaluation

The main regulatory requirements for combustible gas control of future nuclear power reactors are specified in 10 CFR 50.44(c), namely:

- A mixed containment atmosphere shall be ensured; this applies to both design-basis and significant beyond DBAs. A mixed atmosphere means that the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity.

- The concentration of hydrogen shall be limited, both globally and locally, to less than 10 percent.

- Equipment and systems needed to maintain containment integrity shall be able to perform their functions during and after a hydrogen burn; detonations of hydrogen shall also be included unless it can be shown that such detonations are unlikely to occur.

- Equipment shall be provided for continuously measuring hydrogen concentration inside containment following a significant beyond design basis accident.

- A structural analysis shall be completed that demonstrates containment integrity will be maintained during and after a hydrogen burn that ignites all of the hydrogen that is released by the fuel clad-coolant reaction.

The latter four criteria are based on the limiting conditions that are created by significant beyond DBAs, which bound the conditions that are generated during a design basis accident. The significant beyond design basis analyses shall consider the amount of hydrogen that is equivalent to that generated from a 100 percent fuel clad-coolant reaction.

6.2.5.4.1 Containment mixing for beyond design basis accidents

In accordance with the requirements stated in 10 CFR 50.44(c)(1), all containments shall have the capability to ensure a mixed atmosphere during significant beyond DBAs. A mixed atmosphere means that the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity.

The acceptance criterion for the CHCS, which provides for a mixed and homogeneous gas atmosphere in the containment, is to maintain the hydrogen concentration in containment, both globally and locally below 10 percent by volume. For PWRs, if the hydrogen concentration exceeds 10 percent, the acceptance criterion can be met if the atmosphere is inerted, either by oxygen concentration too low to support combustion or by high steam concentration.

The applicant used the MAAP4 code to demonstrate the extent of containment mixing. The applicant's "Severe Accident Analysis Report," APR1400-E-P-NR-14003-P, Revision 0, provides the hydrogen concentration versus time for 24 hours for selected scenarios and for all containment nodes modeled in MAAP4 containment combustion analysis.

APR1400-E-P-NR-14003-P is based on the analysis documented in the applicant's calculation note 1-035-N389-101, Revision 3, identified in the Chapter 19 audit report (ML13249A244). The applicant's calculation note selected five initiating events whose accident sequences represent the spectrum of severe accidents important to hydrogen accumulation and distribution in the containment. These are the most probable core damage sequences from the Level 1 PRA, plus representative LOCA sequences. They are: large, medium and small break LOCAs (LBLOCA, MBLOCA, and SBLOCA, respectively); station blackout (SBO); and total loss of feed water (TLOFW). The combustion analysis for all of these base case scenarios results in hydrogen concentration in containment, both locally and globally, maintained below 10 percent, when all the severe accident mitigating systems are credited. These mitigating systems include: auxiliary feedwater, SITs, PARs, HIs, reactor cavity flooding, containment spray, rapid depressurization utilizing the POSRVs, and operation of the three-way valves, as appropriate for each scenario.

The applicant also performed sensitivity analyses on the base case scenarios, by crediting some but not all mitigation systems.

The applicant evaluated the global and local hydrogen concentration and found that:

- the hydrogen concentration does not exceed 10 percent anywhere in the containment except in the IRWST and in the SG compartment for high pressure sequences such as SBO and TLOFW, as long as the POSRV via the three-way valve is available, and

- the hydrogen concentration for all LOCA sequences does not exceed 10 percent anywhere in the containment, as long as the PARs and HIs are available and no containment sprays are actuated.

Detonations are a concern both for the potential of large detonations to challenge containment integrity and local detonations to affect equipment survivability. Since extremely high-energy ignition sources are not present inside containment, the analysis of detonations is limited to those arising from flame acceleration during a deflagration, i.e., deflagration to detonation transitions (DDT). The potential for DDT exists for any node where the hydrogen concentration exceeds 10 percent. The applicant evaluated the potential for DDT for the sensitivity scenarios identified above, utilizing the method described in the reference document NEA/CSNI/R(2000)7, “Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety,” OECD/NEA, October 2000. According to the generally accepted methodology in this document, the DDT condition is considered possible if the conditions for flame acceleration and the detonation cell size allow DDT to develop within a compartment. In any control volume where the hydrogen concentration exceeded 10 percent under flammable conditions, this calculation then evaluated the potential for DDT by estimating the detonation cell width, λ. The cell width is a function of the atmospheric composition as well as temperature and pressure, with small cell widths corresponding to a more easily detonated mixture.

Using these criteria, the applicant found that:

- there is no DDT potential in the containment as long as the POSRV discharge via the three-way valve is available;

- if the POSRV discharge via the three-way valve is not available in high pressure sequences, the IRWST air space and the areas above it have the potential for DDT;

- there is DDT potential in the lower containment areas for any sequences with igniter failure and cavity flooding failure.

The applicant concluded that with all severe accident mitigation features available, there is no potential for DDT anywhere in containment. The staff performed confirmatory calculations on mixing in containment for the same five base cases, using MELCOR with a containment nodalization similar to the applicant’s nodalization. These confirmatory calculations are documented in staff’s contractor report, ERI/NRC 16-208, Revision 2 (ML16314E431). The calculations confirm that in all five base cases, when crediting all severe accident mitigating systems, the containment is well-mixed and does not support hydrogen combustion either by having a hydrogen concentration less than 10 percent both globally and locally or a steam-inerted atmosphere.

The staff selected scenarios that could produce local hydrogen concentrations greater than 10 percent for further analysis. If the analysis resulted in local hydrogen concentrations greater than 10 percent, an analysis for potential DDT was performed, using the same methodology as the applicant.

The staff performed:

- five additional sensitivity calculations on the high pressure scenarios for SBO, varying the availability of each of the mitigating systems;
- two sensitivity calculations for TLOFW, one with depressurization but no three-way valve operation and the other with delayed operation of the three-way valve, i.e., at 60 minutes after core damage vs. 30 minutes used for the base case calculations;
- four sensitivity calculations for LBLOCA, varying the availability of the mitigating systems.

These confirmatory calculations agree with the applicant's results that:

- there is a potential for DDT in the reactor pressure vessel annulus for LBLOCA without PAR or HI operation;
- the potential for DDT in the IRWST and the area above it exists for either SBO or TLOFW with no operation of the three-way valve;
- an additional sensitivity calculation for the TLOFW with depressurization and three-way valve operation delayed until 60 minutes identified a potential DDT condition in the IRWST. This result confirms the importance of changing the position of the three-way valve position within 30 minutes after core damage, for TLOFW.

The staff confirmed that, with all severe accident mitigation features available, there is no potential for DDT anywhere in containment.

The staff also confirmed that for the high pressure TLOFW sequence, depressurization is accomplished via the POSRV and the three-way valves, in a timely manner, i.e., within 30 minutes of core damage, there is no potential for DDT in the IRWST. The importance of the applicant's recommended opening time of the three-way valve has been discussed in other sections of the DCD, such as Section 19.2.5.1, Severe Accident Management Framework, as well as in the KHNP Severe Accident Analysis Report, APR1400-E-P-NR-14003-P, Revision 0.

The staff also confirmed the importance of the mitigating system of the PARs and HIs for the LBLOCA scenarios.

The staff found that, with all severe accident mitigation features available, the containment provided a mixed atmosphere during design-basis and significant beyond design-basis accidents and therefore meets 10 CFR 50.44(c)(1).

6.2.5.4.2 Equipment Survivability

The regulations in 10 CFR 50.44(c)(3) requires that, in a containment that does not rely on an inerted atmosphere to control combustible gases, all equipment and instrumentation in containment needed to establish and maintain safe shutdown and containment structural integrity must also be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, in an amount equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding.

The equipment survivability assessment performed by the applicant to meet the particular requirements of 10 CFR 50.44(c)(3) first identified all the equipment and instrumentation required to mitigate a severe accident and then determined the harsh environmental conditions of pressure and temperature caused by the burning of hydrogen in each location in containment.

Equipment and instrumentation in containment associated with the CHCS and the CHMS that are required to mitigate a severe accident must withstand the conditions expected to occur during a severe accident, including hydrogen burning, are the PARs and the HIs.

DCD Tier 2, Section 19.2.3.3.7., “Equipment Survivability” identifies the mitigation equipment and instrumentation in containment which must withstand the conditions expected to occur during a severe accident. This equipment is also identified in DCD Tier 2, Table 19.2.3-4, “Systems and Equipment/Instrumentation Required for Equipment Survivability Assessments.” On October 22, 2015, the staff issued RAI 264-8243, Question 06.02.05-6, to request the addition of containment piping penetrations and the emergency containment spray backup system (ECSBS) check valve, ECSBS-V1014, be added to this list. In its response to RAI 264-8243, Question 06.02.05-6, dated November 18, 2016, the applicant agreed to add both ECSBS-V1014 and containment piping penetrations to Table 19.2.3-4 (ML16323A488). The staff confirmed that APR1400 DCD, Revision 1 contains this update. RAI 264-8243, Question 06.02.05-6 is resolved and closed.

The applicant using MAAP4 evaluated representative scenarios to determine the harsh temperatures and pressures the selected equipment would be subject to. The applicant then created a histogram of a composite of the temperatures for each node in the MAAP4 model, reordering the maximum temperatures together, to obtain a longer duration of peak temperature. The applicant classified the temperature environments as challenging - severely, highly, quite, moderately or nominally – depending on the magnitude and duration of the temperatures. Figures depicting these temperature curves are found in the DCD, Tier 2, Section 19.2.3.3, “Severe Accident Mitigation Features,” in Figures 19.2.3-16, “ES Curve for Nominally Challenging Environments,” through 19.2.3-20, “ES Curves for Challenging Environments.” The results range from 1200 K short term and 600 K long term for the severely challenging environments to 460 K for the nominally challenging environments. The applicant then assigned a temperature curve to each piece of equipment selected for equipment survivability, as is shown in DCD Tier 2, Table 19.2.3-5, “Summary of Temperature Envelopes for Equipment Survivability Assessment.”

The staff performed two confirmatory calculations using MELCOR for the AICC temperatures in all the containment nodes: one for a LBLOCA scenario, which resulted in higher short term peaks (1340 K) and much lower long term temperatures (400 K); and, the second for a TLOFW scenario which resulted in much lower peaks (500 K) of longer duration and similar long term temperatures. The staff finds that the confirmatory calculation temperature results are bounded by the applicant’s temperature results, except for the short term peak temperatures. Since these peak temperatures last less than 100 seconds, the long term temperature is expected to pose a greater challenge to the equipment. These confirmatory calculations are documented in staff's contractor report ERI/NRC 16-208, Revision 2.

The applicant determined the bounding pressure in containment for the representative scenarios to be 110 psia. The MAAP4 analysis and the results are found in the applicant's report APR1400-E-P-NR-14003-P, Revision 0, "Severe Accident Analysis Technical Report," December 2014 (ML15009A225). The staff performed a confirmatory calculation for a LBLOCA scenario, resulting in a peak containment pressure of 68 psia. This confirmatory calculation is documented in staff's contractor report ERI/NRC 16-208, Revision 2. The staff finds the applicant's selection of the bounding peak pressure of 110 psia acceptable for the equipment survivability assessment.

The staff has evaluated the equipment survivability compliance with 10 CFR 50.44(c)(3). A discussion of the site-specific equipment survivability assessment is found in SE Section 19.2.3.3, "Severe Accident Mitigation Features" under "Equipment Survivability."

6.2.5.4.3 Hydrogen Monitoring

In accordance with 10 CFR 50.44(c)(4)(ii), equipment shall be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen shall be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond DBA for accident management, including emergency planning.

The CHMS performs continuous hydrogen monitoring inside containment for a severe accident. Continuous post-accident indication of containment hydrogen concentration is provided in the control room through redundant safety indicators. The CHMS measures the hydrogen concentration in containment during and after the accident and remains functional. The CHMS system consists of two redundant trains, each with its own gas sampler. Each sampler analyzes two air samples from the containment atmosphere, and one gas sample from the air space in the IRWST. The measurement range of the gas samplers is 0 - 15 percent hydrogen. This information is used for accident management, to assess the efficiency of the CHCS and for estimating the risk of deflagrations in containment.

In order to confirm that the CHCS and the CHMS meet the requirements of 10 CFR 50.44(c)(4), staff issued RAI 155-8167, Question 06.02.05-5, dated August 18, 2015, requesting additional design information about the CHMS, including setpoints, alarms, and readouts in the MCR. Since the applicant provided the additional design information, the staff finds that the response to RAI 155-8167, Question 06.02.05-5, dated November 18, 2015 (ML15322A028) meets the requirements to 10 CFR 50.44(c)(4) by providing equipment for monitoring hydrogen in containment. DCD Tier 1, Section 2.11.4.1, "Design Description," Table 2.11.4-2, "Containment Hydrogen Control System," and Table 2.11.4-3, "Containment Hydrogen Control System ITAAC," will be revised to include detailed information regarding the identity, measuring range, capability of monitors, and installed location of hydrogen monitoring instrumentation. DCD Tier 2, Section 6.2.5.2.2, "Containment Hydrogen Monitoring System," and 6.2.5.2.3, "Alarm and Indication," will also be revised to include specific information regarding the capability of the monitor including measuring range, containment pressure range, power supply, and operation mode change. The staff confirmed that APR1400 DCD, Revision 1 contains this update. RAI 155-8167, Question 06.02.05-5 is resolved and closed.

6.2.5.4.4 Containment Integrity

In order to meet 10 CFR 50.44(c)(5) requirements, a structural analysis must be completed to demonstrate containment integrity. The structural analysis must include loads due to the global deflagration of hydrogen and must include hydrogen detonation loads, unless they can be shown to be highly unlikely.

In DCD Tier 2, Section 19.2.4.2.1, "Combustible Gas Control inside Containment," the applicant evaluated the results of a global burn produced from the hydrogen generated from the complete reaction of 100 percent of the active fuel cladding with steam, performing an adiabatic isochoric with complete combustion (AICC) analysis. The applicant found that the pressure resulting from the AICC analysis is below the factored load requirement of ASME Section III, Division 2, Subarticle CC-3720. The staff's review of the applicant's analysis for internal pressure loading is based upon the applicant's calculation of AICC pressure, considering a safety margin, of 123.7 psia. The staff evaluation of containment structural integrity under this hydrogen combustion load is found in Section 3.8.1, "Concrete Containment," of this SER.

The staff's contractor calculation ERI/NRC 16-208, Revision 2, has confirmed that the AICC pressure calculated by the applicant of 99.8 psia bounds the confirmatory value of 65 psia. The staff finds that the containment integrity under this hydrogen combustion load meets the requirements of 10 CFR 50.44(c)(5).

6.2.5.4.5 Inspection, Tests, Analysis, Acceptance Criteria

DCD Tier 1, Section 2.11.4, "Containment Hydrogen Control System and Monitoring System," provides the design description and non-safety-related functional requirements and location of all the components in the CHCS in Table 2.11.4-1, "Containment Hydrogen Control System Components List." The ITAAC associated with the design commitment are provided in Table 2.11.4-2, "Containment Hydrogen Control System ITAAC." The staff evaluation of the ITAAC will be addressed in SER Section 14.3.11, "Containment Systems ITAAC."

6.2.5.5 Combined License Information Items

There is one proposed COL information item related to this area of review. The staff determined that for COL Information item 19.2(1), until the COL applicant performs and submits the site-specific equipment survivability assessment in accordance with 10 CFR 50.34(f) and 10 CFR 50.44, it shall continue to be included in DCD Tier 2, Table 1.8-2, "APR1400 Combined License Information Items." The staff believes the COL item should be expanded to refer to the equipment required for severe accident mitigation, identified in DCD, Tier 2, Table 19.2.3-4, and to the figures identified in Table 19.2.3-5 which correlate the equipment with the temperature curves, which represent the containment atmospheric temperatures during a severe accident. The staff requested this expanded description in RAI 264-8243, Question 06.02.05-6. The applicant has agreed to add expanded language (ML16323A490). The staff confirmed that APR1400 DCD, Revision 1 contains this update. RAI 264-8243, Question 06.02.05-6 is resolved and closed.

6.2.5.6 Conclusion

Combustible gas control inside containment is provided by the CHCS and the CHMS. The staff evaluated the design of the CHCS and CHMS for the APR1400 described primarily in DCD Tier 2, Section 6.2.5, DCD Tier 2, Section 19.2, "Severe Accident Evaluations," and DCD Tier 1, Section 2.11.4, "Containment Hydrogen Control System." The staff's evaluation to confirm compliance with the provisions of 10 CFR 50.44(c) and 10 CFR 52.47(b)(1) was performed in accordance with the guidance provided in SRP Section 6.2.5.

In order to satisfy the requirement in 10 CFR 50.44(c)(2) to provide adequate mixing of the containment atmosphere, the concentration of hydrogen, both globally and locally shall be limited to 10 percent for significant beyond DBAs. If the hydrogen concentration cannot be limited to 10 percent, the containment atmosphere shall not support detonation, which could cause loss of containment integrity. The applicant has demonstrated that for a wide spectrum of severe accident sequences there is no potential for DDT anywhere in containment when all the mitigating systems are available. Of particular importance in maintaining containment integrity is the timely manual opening of the POSRVs and the operation of the three-way valves. Based on the staff's review and confirmatory calculations, the staff finds that applicant's combustible gas control design meets the requirements of 10 CFR 50.44(c)(1) and (c)(2).

The staff's contractor calculations in ERI/NRC 16-208, Revision 2, have confirmed that the AICC pressure calculated by the applicant of 99.8 psia bounds the confirmatory value of 65 psia. The staff finds that the containment integrity under this hydrogen combustion load is maintained and that this aspect of the APR1400 design meets the applicable requirements of 10 CFR 50.44(c)(5). The staffs has found that the applicant's containment pressure of 123.7 psia, which represents the AICC calculated pressure with a safety factor added, is bounded by the factored load category value, and therefore acceptable. Based on its review, the staff finds that the APR1400 design to protect containment integrity following a global hydrogen burn meets the applicable requirements of 10 CFR 50.44(c)(5).

The staff's review and conclusion of the applicant's proposed ITAAC for CHCS and the CHMS is addressed in SER Section 14.3.11.

6.2.5.7 References

1. SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," U.S. Nuclear Regulatory Commission, June 1990.

2. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," U.S. Nuclear Regulatory Commission, April 1993.

3. NEA/CSNI/R(2000)7, "Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety," OECD/NEA, October 2000.

4. APR1400-E-P-NR-14003-P, Revision 0, "Severe Accident Analysis Technical Report," KHNP, December 2014. (ML15009A225).

5. ERI/NRC 16-208, Revision 2, October 2016, "Assessment of Combustible Gas Control during Severe Accidents in APR 1400," ML16314E431

6. APR 1400 PRA and SA Audit Report, ML13249A244

7. KHNP Calculation Note 1-035-N389-101, Revision 3, November 2015, "Hydrogen Generation and Control during Severe Accidents"

8. KHNP Calculation Note 1-035-N389-102, Revision 1, April 2013, "Assessment of AICC Pressure Load Due to Hydrogen Combustion in Containment"

9. KHNP Calculation Note 1-035-N389-103, Revision 2, September 2014, "Analysis of Local DDT Potential in the APR 1400 Containment"

6.2.6 Containment Leakage Testing

6.2.6.1 Introduction

Section 6.2.6 of the APR1400 DCD, Tier 2, addresses the leakage rate testing program for the reactor containment. Testing requirements provide assurance that the containment leak tight integrity can be verified throughout the service lifetime. Additionally, periodic containment overall integrated leakage rate (Type A) testing, local containment penetration leakage rate (Type B) testing, and local CIV leakage rate (Type C) testing shall be performed to assure that leakage through the containment systems and components that penetrate the primary containment do not exceed allowable leakage rate values of the TS.

6.2.6.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 description of containment leak testing in Section 6.2.6, summarized here, in part, as follows:

> Section 6.2.6 of the APR1400 DCD states that the containment leakage testing program and limits implement the performance-based leakage testing requirements of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," (Option B) using the specific methods and guidance provided in NEI 94-01, "Industry Guideline for implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements," as modified and endorsed by the NRC in RG 1.163, "Performance-Based Containment Leak-Test Program," and is designed to comply with the requirements of GDC 52, GDC 53, and GDC 54, of 10 CFR Part 50, Appendix A.
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> The program elements and limits are identified in Chapter 16 in DCD Tier 2, which is based on NUREG–1432, Revision 4.0 "Standard Technical Specifications, Combustion Engineering Plants," Section 5.5, and Appendix J, Option B.
>
> DCD Tier 2, Sections 6.2.6.1, "Containment Integrated Leakage Rate Test (Type A)," through 6.2.6.5, "Special Testing Requirements," discuss the main aspects of the leakage rate testing program including integrated leak rate testing, penetration testing, valve testing, scheduling and reporting of periodic tests, and special testing requirements. DCD Tier 2, Figure 6.2.4-1, "Containment Isolation Valve Arrangement," illustrate the provisions for containment penetration testing. DCD Tier 2, Table 6.2.4-1, "List of Containment Penetrations and System Isolation Positions," include the leak rate test types to be performed for each penetration/valve.
>
> The applicant indicated that the proposed schedule and test report content requirements associated with performing pre-operational and periodic leakage rate testing is in accordance with the guidance provided in, NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and

ANSI/ANS 56.8-1994, “American National Standard for Containment System Leakage Testing Requirements,” as modified and endorsed by the NRC in RG 1.163.

ITAAC: There are no ITAAC associated with DCD Tier 2, Section 6.2.6.

TS: The TSs associated with DCD Tier 2, Section 6.2.6 are given in DCD Tier 2, Chapter 16, Sections 3.6.1, B3.6.1, 3.6.2, “Containment Air Locks,” B3.6.2, and 5.5.16. TS 5.5.16, “Containment Leakage Rate Testing Program,” provides the key TS parameters and the bases on which the staff accepts these parameters are summarized in Table 6.2.6-1 in Section 6.2.6.4.5 of this SE.

Technical Reports: There are no technical reports for this area of review.

APR1400 Interface Issues identified in the DCD: There are no APR1400 interface issues for this area of review.

Site Interface Requirements identified in the DCD:

In APR1400 DCD Tier 2, Section 13.4, “Operational Program Implementation,” the applicant provided the following item associated with DCD 6.2.6 that will be provided by the COL applicant:

COL 13.4 (1): The COL applicant is to develop operational programs and provide schedules for implementation of the programs, as defined in SECY-05-0197 (CILT). The COL applicant is to provide commitments for the implementation of operational programs that are required by regulation.

Cross-cutting Requirements (TMI, USI/GSI, Op Ex): None for this area of review.

RTNSS: There is no RTNSS for this area of review.

In accordance to 10 CFR 20.1406: There are no 10 CFR 20.1406 requirements for this area of review.

CDI: There is no CDI for this area of review.

6.2.6.3 Regulatory Basis

The relevant requirements of the Commission’s regulations and the associated acceptance criteria are given in Section 6.2.6 of NUREG–0800, the SRP, and are summarized below. Review interfaces with other SRP Sections can be found in Section 6.2.6 of NUREG–0800.

- The regulations in 10 CFR 50.54(o), “Conditions of licenses,” requires the primary reactor containment to meet the leakage-rate test requirements specified in 10 CFR Part 50, Appendix J, Option A or B “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.”

- GDC 52, requires that the reactor containment and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

- GDC 53, requires that the reactor containment be designed to permit (1) periodic inspection of all important areas such as penetrations, (2) an appropriate surveillance program and (3) periodic testing at containment design pressure of the leak tightness of penetrations which have resilient seals and expansion bellows.

- GDC 54, requires that piping systems penetrating primary containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

- The regulations in 10 CFR 52.47(a)(2) requires that in evaluating a proposed nuclear power plant design and the safety features that are engineered into the facility, an applicant should assume the expected demonstrable leakage rate from the containment. As stated in SRP Section 6.2.6, Section II, nuclear power plant leakage rate testing experience shows that a design leakage rate of 0.1 percent per day provides adequate margin above typically measured containment leakage rates and is compatible with current leakage rate test methods and test acceptance criteria. Therefore, the minimum acceptable design containment leakage rate may not be less than 0.1 percent per day.

- The regulations in 10 CFR Part 50, Appendix J, requires that the primary reactor containment shall be designed such that the maximum allowable leakage rate, La, as specified in the TS or associated bases when tested at the calculated peak internal containment pressure, Pa, related to the DBA is not exceeded, with margin, as measured with containment overall integrated leakage rate (Type A) tests, local leakage rate (Type B) tests across pressure retaining, leakage limiting boundaries, and CIV leakage rate (Type C) tests.

- The regulations in 10 CFR 52.47(b)(1) requires that an application for a standard DC contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that if the inspections, tests, analyses and acceptance criteria are performed, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the Commission’s rules and regulations.

Acceptance criteria adequate to meet the above requirements include:

- Appendix J, Option B, RG 1.163 endorses NEI 94-01, Revision 0 (with certain exceptions), which provides similar guidance in Sections 8.0 and 9.0. Instrumentation lines that penetrate containment however, are sometimes isolated for the containment integrated leakage testing (CILRT). To ensure that they are included in the test, the following should be done. Leakage rate testing of instrumentation lines that penetrate containment may be done in conjunction with either the local leak rate tests or the CILRT. Instrumentation lines that are not locally leakage rate tested should not be isolated from the containment atmosphere during the performance of the CILRT. The measured leakage rates from instrumentation lines that are locally leakage rate tested, and also isolated during CILRTs, should be added to the CILRT result. Provisions should be made to ensure that instrumentation lines isolated during the CILRT are restored to their operable status following the test.

- The reactor containment leakage rate testing program, as described in the DCD is acceptable if, under Option B, it meets the requirements stated in Option B of Appendix J to 10 CFR Part 50 and, under V.B.2 and V.B.3 of Option B, either complies with methods approved by the Commission and endorsed in RG 1.163 and includes a requirement to do so in the TS, or complies with the provisions of some other implementation document which has been adequately justified by the staff, with supporting analyses, and is cited as a requirement in the TS.

- All leakage rate tests, performed by either pneumatic or hydrostatic means, should have the capability to quantify the leakage rates either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the TS.

- Appendix J, Option B, RG 1.163 endorses NEI 94-01, Revision 0 (with certain exceptions), which provides similar guidance in Sections 8.0 and 10.0. At the CP or standard DC stage, the applicant should identify all CIVs that will be locally (Type C) leakage rate tested with the test pressure applied in a direction opposite to that which would occur under accident conditions and should commit to justify, at the operating license or COL stage, that such testing will result in equivalent or more conservative results.

- NEI 94-01, Revision 0 (Section 6.0, “General Requirements”), and ANSI/ANS 56.8 1994 (Section 3.3.1, “General”) state that Type B or Type C tests are not required for the following cases:

    - Containment boundaries that do not constitute potential containment atmospheric leakage pathways during and following a design-basis (DB) LOCA;

    - Containment boundaries sealed with a qualified seal system;

    - Test connections, vents, and drains between CIVs, which:

        - are one inch or less in size,

        - administratively secured closed, and

        - consist of a double barrier (e.g., two valves in series, one valve with a nipple and cap, one valve and a blind flange).

This guidance may be applied to either Option A or Option B of Appendix J. Examples of Case (a) are lines that terminate below the minimum post-accident water level of the suppression pool in a BWR or the recirculation sump in a PWR.

For Case (b), a qualified seal system is defined in ANSI/ANS 56.8 1994 as a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 Pa, for at least 30 days following the DB LOCA. The staff’s position is that the analysis of the sealing capability includes the assumption of the most limiting SF of any active component. Also, unless there is a virtually unlimited supply of sealing liquid (such as from a suppression pool or recirculation sump), limits for liquid leakage rate should be assigned to these valves based on analysis and included in the plant TSs. Periodic leakage rate testing, using the sealing liquid as the test medium, is then needed to ensure that the TS limits are maintained.

For Case (c), to ensure that containment integrity is restored following testing, the test, vent, and drain connections that are used to facilitate local leakage rate testing and the performance of the CILRT should be under administrative control and should be subject to periodic surveillance, to ensure their integrity and to verify the effectiveness of administrative controls.

6.2.6.4 Technical Evaluation

The staff's review of this section is primarily focused on assuring that the APR1400 design provisions will permit containment leakage rate testing to be done in accordance with the requirements of Appendix J, 10 CFR Part 50.

The staff reviewed the APR1400 DCD Tier 2, Revision 0, Section 6.2.6 and the proposed TS of DCD Tier 2, Chapter 16, Section 5.5.16 in accordance with SRP Section 6.2.6 and RG 1.163, "Performance-Based Containment Leak-Test Program." In addition, review of applicable portions of APR1400 DCD Tier 2, Section 6.2.8 "Combined License Information," is included in this review. The applicant's containment leakage testing program is acceptable if it meets the regulatory guidance commensurate with the safety function to be performed. The staff review will ensure that the relevant requirements of 10 CFR 50.54(o) Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," 10 CFR 52.47, 10 CFR Part 50, Appendix A GDC 52, GDC 53, and GDC 54 are met. The staff's evaluation of how these requirements have been met are discussed below.

The staff review of the APR1400 containment leakage rate testing program encompassed the following review areas, as identified in SRP Section 6.2.6 and SRP Chapter 16, TS 5.5.16:

- CILRT (Type A tests as defined by Appendix J), including pretest requirements, general test methods, acceptance criteria for preoperational and periodic leakage rate tests, provisions for additional testing in the event of failure to meet acceptance criteria and scheduling of tests,
- Containment penetration leakage rate tests (Type B tests as defined by Appendix J), including identification of containment penetrations general test methods, test pressures and acceptance criteria,
- CIV local (Type C tests as defined by Appendix J), including identification of isolation valves, design provisions for such testing, general test methods, test pressures, and acceptance criteria,
- Scheduling and reporting of periodic tests,
- Special testing requirements,
- Proposed TS requirements pertaining to containment leakage rate testing,
- COL Action Items.

The staff's findings for each of the above areas are discussed below.

6.2.6.4.1 Containment Integrated Leakage Rate (Type A) Test

CILRTs serve to provide assurance that the containment leakage rate, in the event of an accident, will not exceed the values assumed in the analyses of the radiological consequences of DBAs.

6.2.6.4.1.1 Pretest Requirements and General Test Method for Type A Tests

DCD Tier 2, Section 6.2.6.1 states that the CILRT is conducted in accordance with ANSI/ANS-56.8 and NEI 94-01. The containment is pressurized with clean air using temporary air compressors and dryers. DCD Tier 2, Table 6.2.4-1 and Figure 6.2.4-1, show the containment penetrations and isolation valve arrangement. Based on review of this table and figure, the staff found that permanently installed penetrations to facilitate controlled pressurization and depressurization of the containment are not shown in DCD Table 6.2.4-1. The Table 6.2.4-1 did not show all the spare penetrations or penetration numbers associated with CIVs and electrical penetrations. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-9 requesting the applicant to show all the penetrations in the DCD. In its response to RAI 306-8240, Question 06.02.06-9, dated June 26, 2016, (ML16181A349), the applicant stated that penetration numbers associated with CIVs and electrical penetrations will be added to DCD Tier 2, Table 6.2.4-1. Spare penetrations (Item Nos.165 to 177) and penetrations that will be used in the CILRT (item nos. 162 - 164) will be also added to DCD Table 6.2.4-1. Changes to DCD Table 6.2.4-1 were also requested in RAI 357-8344, Question 06.02.04-11. To be inclusive of all changes to Table 6.2.4-1, the applicant provided the associated penetration numbers in revised markups of DCD Table 6.2.4-1 in response to RAI 357-8344, Question 06-02.04-11 (ML16182A591). The staff has reviewed the applicant response to RAI 306-8240, Question 06.02.06-9, and revised markup of DCD Table 6.2.4-1 regarding showing all containment penetrations in DCD and finds it acceptable. The staff finds it acceptable because the applicant showed all the spare penetrations in the markups to DCD Tier 2, Table 6.2.4-1 and therefore meets the SRP Section 6.2.6, Section II, Acceptance Criteria 2 requirements. The staff confirmed that the RAI markups incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-9 resolved and closed.

In DCD Tier 2, Section 6.2.6.1, the applicant provided information on the test method for Type A tests. Normally, the CILRT is performed following the completion of a series of local leakage rate tests, and inspection of containment structures and components. During the preoperational testing, a structural integrity test is performed prior to the CILRT. SRP Section 6.2.6 acceptance criteria state that during the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments shall be made so that the containment can be tested as close to the “as-is” condition as practical per 10 CFR 50, Appendix J. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-10, to request the applicant to include this additional information for meeting of SRP 6.2.6 acceptance criteria in the DCD. In its response to RAI 306-8240, Question 06.02.06-10, dated January 28, 2016 (ML16028A185), the applicant provided revised markup of DCD Tier 2, Section 6.2.6.1 to include the above requirements of SRP Section 6.2.6 acceptance criteria. The staff reviewed the applicant response as indicated above for meeting SRP 6.2.6 acceptance criteria for compliance with the requirements of 10 CFR Part 50, Appendix J and therefore, finds it acceptable. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-10 resolved and closed.

DCD Tier 2, Section 6.2.6, specifies that the test pressure for the Type A, B and C tests will be at the maximum calculated peak containment pressure under DBA. DCD Tier 2, Table 6.2.1-2, lists the maximum calculated peak containment pressure under DBA as 3.59 kg/cm^{2}G (51.09 psig). This pressure value of 51.09 psig is used in DCD Chapter 16, Bases Section B.3.6.1 for containment leak rate test. But in TS Section 5.5.16 "Containment Leakage Rate Testing Program" under item b, the value of peak containment pressure for leak rate test is listed as 51.77 psig. The staff issued RAI 306-8240, Question 06.02.06-1, asking the applicant to justify using a lower value of peak containment pressure under DBA than calculated value in TS Section 5.5.16. In its response (ML16036A101), the applicant stated that the pressure value of 51.77 psig listed in TS 5.5.16 is an incorrect value. The value of 51.77 psig will be revised to the correct value listed in Table 6.2.1-2 of DCD Tier 2, 51.09 psig and provided revised markup of associated TS Section 5.5.16 with correct value of 51.09 psig. The staff has reviewed the applicant response and finds it acceptable for revising the TS Section 5.5.16 with the correct value listed in Table 6.2.1-2 of 51.09 psig. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-1 resolved and closed.

Based on review of DCD Tier 2, Section 6.2.6.1 the staff found that Type A tests will be performed at a test pressure equal to the calculated accident peak containment pressure (Pa). The staff found that this conforms to the requirements of 10 CFR Part 50, Appendix J and guidance in RG 1.163 and NEI 94-01 for leakage testing and is therefore acceptable.

The staff finds that the Type A test description conforms to the SRP Section 6.2.6 acceptance criteria because the applicant has specified that no repairs or adjustments be made to the containment prior to the performance of the CILRT. In addition, the staff found that the applicant has described CILRT test prerequisites that conform to the guidance in RG 1.163 and NEI 94-01, to the level of detail described in RG 1.206, Part III, Section C.I.6.2.6.1. Therefore, the staff found that the Type A test has the capability to quantify the leakage rate either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the TS. The staff found that the requirements of GDC 52 and GDC 53, Appendix J to 10 CFR Part 50, and 10 CFR 52.47(a)(2) have been met as these regulations apply to the test method for type A tests.

6.2.6.4.1.2 Acceptance Criteria for Type A Tests; Provisions for Additional Testing

The staff has reviewed the Type A leakage rate testing acceptance criteria provided in DCD Tier 2, Section 6.2.6, and Chapter 16. In DCD Tier 2, Chapter 16, TS B 3.6.4-1, the applicant stated that the peak containment internal pressure for the DB LOCA, Pa is equal to 3.59 kg/cm^2 (51.09 psig). In DCD Tier 2 Chapter 16, Section 5.5.16, the applicant stated that the maximum allowable leakage rate (La) is 0.10 percent of the containment air weight per day at Pa. During the first startup following testing, the leakage rate acceptance criterion will be less than or equal to 0.75 La, the staff found that this is in accordance with the provisions of Appendix J to 10 CFR Part 50, SRP Section 6.2.6, and RG 1.163. The staff found that the allowable leakage rate of 0.10 percent per day is equal or conservative with respect to the value used in analyses of the radiological consequences of a LOCA, as cited in DCD Tier 2, Table 15.6.5-13, "Major Input Parameters Used in Radiological Consequences Analysis for Large Break LOCA," (i.e., 0.10 percent of the containment air weight per day at Pa.) and is consistent with the provisions of Section 6.2.6 of the SRP. As stated above, the Type A leakage rate testing acceptance criteria provided in DCD Tier 2, Section 6.2.6, and Chapter 16 is an acceptable leakage rate acceptance criterion for Type A tests.

DCD Tier 2, Section 6.2.6, specifies the use of 10 CFR Part 50, Appendix J, Option B for Type A, B and C containment leakage rate testing. RG 1.163, "Performance-Based Containment Leak-Test Program," endorses NEI 94-01, Revision 0 for an acceptable method for complying with Option B, NEI 94-01 references ANSI/ANS 56.8-1994. DCD Tier 2, Section 6.2.4.2 and Section 14.2.1.120 state that Type B and C leak rate testing is done in accordance with ANSI/ANS 56.8-1994 However, DCD Tier 2, Sections 6.2.4.4, "Testing and Inspection," and 6.2.6, "Containment Leakage Testing," state that testing is done in accordance with ANSI/ANS 56.8 (Reference 31 which indicates 2002 version). The NRC has not yet reviewed and accepted the 2002 version. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-2, asking the applicant to submit ANSI/ANS 56.8-2002 for formal NRC review and approval and provide an explanation of how it comports with RG 1.163, NEI 94-01, and ANS 56.8-1994, or modify the DCD to reference ANSI/ANS 56.8-1994 throughout DCD. In its response to RAI 306-8240, Question 06.02.06-2, dated January 28, 2016 (ML16028A185), the applicant stated that the CILRT discussed in DCD Tier 2, Sections 6.2.4.4 and 6.2.6 conforms with 10 CFR Part 50, Appendix J, Option B (Reference 37), and follows the guidance of NRC RG 1.163. The test is performed in accordance with NEI 94-01 and ANSI/ANS 56.8-1994. The "November 2002" date of Reference 31 will be changed to "August 1994" in DCD Tier 2, Section 6.2.9, "References," and provided revised markups of DCD Tier 2, Section 6.2.9. The staff reviewed the applicant's response and finds it acceptable because the applicant corrected the version of ANSI/ANS referenced in DCD Tier 2, Section 6.2.9. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-2 resolved and closed.

The regulations in10 CFR Part 50, Appendix J, Option B, Section V.B.3 requires justification; including supporting analyses if a licensee (applicant) chooses to deviate from methods endorsed in the RG. DCD Tier 2, Chapter 16 TS Bases Section B 3.6.1 states that "comply with 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions." DCD Tier 2, Chapter 16 TS, Section 5.5.16 has similar words related to RG 1.163. The staff issued RAI 306-8240, Question 06.02.06-4, asking the applicant to provide a list of any requested deviations or exemptions from Appendix J or RG 1.163 along with justification and supporting analyses in the DCD.

In its response to RAI 306-8240, Question 06.02.06-4, dated January 28, 2016 (ML16028A185), the applicant stated that the DCD Tier 2, Chapter 16, Section 5.5.16 and TS B 3.6.1 do not have any deviations or exemptions from 10 CFR Part 50, Appendix J or RG 1.163. DCD Tier 2, Chapter 16, Section 5.5.16 and TS B 3.1.6 will be revised to clarify that there are no deviations or exemptions from 10 CFR Part 50, Appendix J or RG 1.163. The applicant provided the revised markups of DCD Tier 2, Chapter 16, Section 5.5.16 and TS B.3.6.1 associated with its response. The staff has reviewed the applicant response and finds it acceptable. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-4 resolved and closed.

The staff found that the provisions for additional testing, in the event of failure to meet acceptance criteria, conform to the guidance in RG 1.163 and NEI 94-01 Section 9.2.6, "Corrective Action," to the level of detail described in RG 1.206, Part III, Section C.I.6.2.6.1. Consequently, the staff found that with exception of the above confirmatory items, the requirements of GDC 52 and GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 52.47(a)(2) have been met as these regulations apply to the acceptance criteria and provisions for additional testing for Type A tests.

6.2.6.4.1.3 Scheduling and Reporting of Type A Tests

The staff reviewed the DCD Section 6.2.6.4, DCD COL 6.2(1) and DCD Table 1.9-1 for proposed scheduling and reporting requirements associated with performing pre-operational and leakage rate testing. As stated in Section 6.2.6.4 of the DCD, the proposed schedule and test report content requirements associated with performing pre-operational and periodic leakage rate testing are in accordance with NEI 94-01, as modified and endorsed by the NRC in RG 1.163. NEI 94-01 in turn references ANSI/ANS-56.8-1994 for the technical contents of the reports. Per DCD COL 6.2(1), the implementation milestones for pre-operational and periodic CILRT leak rate tests will be developed by each COL applicant. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-3, to clarify what aspects of the containment leak rate testing program are to be certified as part of the DC of the APR1400 and what are to be left for the COL Applicant. DCD Tier 2, Table 1.9-1, "APR1400 Conformance with Regulatory Guides," states that the APR1400 conforms with RG 1.163 (thus, by reference to NEI 94.01 and ANSI/ANS 56.8) and DCD Tier 2, Chapter 16, TS Section 3.6.1 and 5.5.16 which describes a CILRT program. If there are exceptions to the standards and regulatory guidance, the applicant was requested to specifically identify those in the DCD.

In its response to RAI 306-8240, Question 06.02.06-3, dated March 3, 2016 (ML16063A399), the applicant indicated that DCD Section 6.2.6 states that the reactor containment, containment penetrations, and containment isolation barriers are designed to permit periodic leakage rate testing, and that the CILRT program of the APR1400 conforms with the requirements of 10 CFR Part 50, Appendix J, Option B and follows the guidance of RG 1.163. The Section also states that program tests are performed in accordance with NEI 94-01 and ANSI/ANS 56.8. Sub-Sections 6.2.6.1 through 6.2.6.3 describe the Type A, B, and C testing which is to be performed, respectively. Sub-Section 6.2.6.4 describes the scheduling and reporting of the periodic tests, and Sub-Section 6.2.6.5 describes special testing requirements.

The applicant stated that the aforementioned information is provided in DCD Tier 2, Section 6.2.6, in order to describe, at a high level, the CILRT program which shall be developed by the COL applicant who selects the APR1400 standard design. DCD Tier 2, Section 13.4, "Operational Program Implementation," COL 13.4(1) requires the development of the CILT program by stating, "[t]he COL applicant is to develop operational programs and provide schedules for implementation of the programs, as defined in SECY-05-0197. The COL applicant is to provide commitments for the implementation of operational programs that are required by regulation. In some instances, the programs may be implemented in phases, where practical, and the applicant is to include the phased implementation milestones." COL 6.2(1) will be deleted in order to avoid redundancy with COL 13.4(1) and provide clarity with regard to the COL's responsibility in developing the CILRT program.

DCD Tier 2, Chapter 16, Section 3.6.1, SR Section 3.6.1.1, requires that visual examination and leakage rate testing be performed in accordance with the CILRT program. TS Section 5.5.16 describes the CILRT program to which SR 3.6.1 refers, and has been modified in the response to RAI 306-8240, Question 06.02.06-4, dated January 28, 2016, to clarify that there are no deviations or exemptions from 10 CFR Part 50, Appendix J or RG 1.163. As required by COL 13.4(1), the program is to be developed by the COL applicant.

DCD Tier 2, Chapter 1, Table 1.9-7, Item II.H "Containment Leak Rate Testing," will be revised to refer to Sections 6.2.6 and TS Section 5.5.16, and to indicate that containment leak rate testing is to be performed in accordance with RG 1.163. The applicant provided markups of DCD Tier 2, Table 1.8-2, Table 1.9-7, Section 6.2.6.1, and Section 6.2.8, "Combined License

Information," associated with above response. The staff has reviewed the applicant response and finds it acceptable because it will be performed in accordance with RG 1.163. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-3 resolved and closed.

The staff finds that DCD description of scheduling and reporting of periodic tests conform to the guidance of RG 1.163 and NEI 94-01 Section 12, to the level of detail described in RG 1.206, Part III, Section C.1.6.2.6.1. Therefore, the staff found the requirements of GDC 52 and GDC 53, Appendix J to 10 CFR Part 50, and 10 CFR 52.47(a)(2) have been met as these regulations apply to the scheduling and reporting of type A tests because it conforms to the guidance of RG 1.163 and NEI 94-01 as indicated above.

6.2.6.4.2 Containment Penetration (Type B) Leakage Rate Testing

Type B tests are intended to detect or measure the leakage rate across pressure-retaining or leakage-limiting boundaries other than CIVs.

6.2.6.4.2.1 Identification of Containment Penetrations

The staff reviewed the DCD Tier 2, Section 6.2.6.2 "Containment Penetration Leakage Rate Test -Type B" and associated DCD Tables and Figures for identification of Type B containment penetrations. The staff found that DCD Tier 2, Revision 0, Table 6.2.4-1, "List of Containment Penetrations and System Isolation Positions," does not include the complete list of penetrations that will receive preoperational and periodic Type B tests. DCD Tier 2, Table 6.2.4-1 does not list spare penetrations or the penetrations that will be used in the CILRT for temporary air compressors to facilitate controlled pressurization and depressurization. On November 16, 2016, the staff issued RAI 306-8240, Question 06.02.06-9, to request the applicant to include all the penetrations in the DCD. In its response, the applicant stated that all containment penetrations including spare will be added to DCD Tier 2, Table 6.2.4-1. Changes to DCD Table 6.2.4-1 were also requested in RAI 357-8344, Question 06.02.04-11, to be inclusive of all changes to Table 6.2.4-1, the applicant provided the associated markups in response to RAI 357-8344, Question 06.02.04-11. The staff has reviewed applicant response to RAI 306-8240, Question 06.02.06-9 and RAI 357-8344, Question 06.02.04 and revised markup of DCD Table 6.2.4-1 and finds it acceptable because it included a complete list of penetrations. The staff confirmed that the RAIs markups were incorporated into Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-09 resolved and closed. The penetrations subject to Type B testing consist of equipment hatch, personnel air lock, fuel transfer tube, and 66 electrical penetrations. The staff finds that the design conforms to SRP Section 6.2.6, Section II acceptance criteria as it applies to the identification of penetration that are subject to Type B tests and the identification and justification of those penetration for which B testing is not required. The staff found that the design conforms to the guidance in RG 1.206, Part III, Section C.1.6.2.6.2. Consequently, the staff found that the requirements of GDC 52 and GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 52.47(a)(2) have been met as these regulations apply to the identification of containment penetrations.

6.2.6.4.2.2 Design Provisions for Type B Testing

DCD Tier 2, Revision 0, Section 6.2.4.4, "Testing and Inspection," and DCD Figure 6.2.4-1, "Containment Isolation Valve Arrangement," and found that they do not show test connections for the Type B penetrations, including personnel airlocks, equipment hatch, fuel transfer tube, and electrical penetrations. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-8, to request the applicant to include this information in the DCD. In its response to RAI 306-8240, Question 06.02.06-8, dated June 29, 2016 (ADAMS Accession No. ML16161A349), the applicant provided revised DCD Tier 2, Figure 6.2.4-1, Sheets 11 and 12 to show the revised leak rate test connections in the personnel airlock and equipment hatch, respectively. The applicant provided revised Figure 6.2.4-1 Sheet 13 to show the aperture seals and leak test connections in the typical electrical penetration assembly. Welding and aperture seals are used for sealing of electrical penetrations and test connection will be used for leakage test of electrical penetrations. The applicant provided revised DCD Tier 2, Figure 6.2.4-1, Sheet 10 in its response to RAI 357-8344, Question 06.02.04-9, dated June 30, 2016 (ML16182A591), to show the leak test connections of fuel transfer tube. The staff reviewed the applicant's response showing test connections for Type B penetration and finds it acceptable. The staff confirmed that the RAI markups were incorporated into Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-8 resolved and closed.

The staff concludes that the design conforms to the guidance in RG 1.163 and NEI 94-01, Section 6 as it applies to design provisions for Type B testing that enable application of test pressure in the same direction that would occur during a DB LOCA. Therefore, the staff finds that the design meets SRP Section 6.2.6, Section II acceptance criteria as it applies to design provisions for Type B testing. Consequently, the staff found that the requirements of GDC 52 and GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 52.47(a)(2) have been met as these regulations apply to the design provisions for Type B testing because the design conforms to the guidance in RG 1.163 and NEI 94-01 as indicated above.

6.2.6.4.2.3 Test Method for Type B Tests

Based on review of DCD Tier 2, Revision 0, Section 6.2.6.2, the staff found that Type B tests will be performed at a test pressure equal to or greater than Pa (the calculated accident peak containment pressure) using either a pressure-decay method or a flowmeter method. The staff found that it conforms to the guidance in RG 1.163 and NEI 94-01 Section 6.4, and is therefore acceptable.

The staff has reviewed the Type B test method as described in the DCD Tier 2, Section 6.2.6.2, "Containment Penetration Leakage Rate Test (Type B)," and found that because the description conforms to the guidance in RG 1.163 and NEI 94-01, to the level of detail described in RG 1.206, Part III, Section C. I.6.2.6.2, the staff finds that the Type B test description conforms to the SRP Section 6.2.6 acceptance criteria. Therefore, the staff found that the Type B test has the capability to quantify the leakage rate either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the TS. Consequently, the staff found that the requirements of GDC 52 and GDC 53, Appendix J to 10 CFR Part 50, and 10 CFR 52.47(a)(2) have been met as these regulations apply to the test method for Type B tests.

6.2.6.4.2.4 Acceptance Criteria for Type B Tests

The staff reviewed the Type B leakage rate test acceptance criteria contained in the APR1400 DCD Tier 2 Revision 0, Section 6.2.6 and Chapter 16, TS Section 5.5.16. The staff found that the Type B leakage rate test results will be combined with the Type C results, in accordance with Appendix J to 10 CFR Part 50. During the first startup following testing, the leakage rate acceptance criterion will be less than 0.6 La. The staff found that this acceptance criterion for combined Type B and C leakage is acceptable because it conforms to the guidance of NEI 94-01 Paragraph 10.2, which is endorsed by RG 1.163. In addition the staff found that this criterion is consistent with the CE Plants, Revision 4.0 Standard TS (NUREG– 1432), Section 5.5, Appendix J Option B. Consequently, the staff found that the criterion is in accordance SRP Section 6.2.6, and thus is in accordance with the provisions of Appendix J to 10 CFR Part 50.

In addition, air lock chambers and individual door shall meet the specific leakage rate acceptance criteria identified in APR1400 DCD Tier 2, Revision 0, Chapter 16, TS Section 5.5.16. These are:

1. Overall air lock leakage rate is less than or equal to 0.05 La when tested at greater than or equal to Pa.

2. For each door, leakage rate is less than or equal to 0.01 La when tested at greater than or equal to 10 psig.

The staff found the above acceptance criteria acceptable because they are consistent with air lock leakage acceptance criteria in the CE Plants, Revision 4.0 Standard TS (NUREG–1432), Section 5.5, Appendix J, Option B. The staff found that APR1400 plant design is no different from current designs in which NUREG–1432 is applicable, with regard to containment airlock design. Consequently, the staff found that airlock leakage test acceptance criteria are in accordance SRP Section 6.2.6, and thus is in accordance with the provisions of Appendix J to 10 CFR Part 50.

6.2.6.4.2.5 Scheduling and Reporting of Type B Tests

The staff reviewed the DCD Section 6.2.6.4, DCD COL 6.2(1) and DCD Table 1.9-1 for proposed scheduling and reporting requirements associated with performing pre-operational and leakage rate testing As stated in Section 6.2.6.4, "Scheduling and Reporting of Periodic Tests," of the DCD Tier 2, the proposed schedule and test report content requirements associated with performing pre-operational and periodic leakage rate testing are in accordance with NEI 94-01, as modified and endorsed by the NRC in RG 1.163. NEI 94-01 in turn references ANSI/ANS-56.8-1994 for the technical contents of the reports. Per DCD COL 6.2(1), the implementation milestones for pre-operational and periodic CILRT leak rate tests will be developed by each COL applicant. On November 16, 2016, the staff issued RAI 306-8240, Question 06.02.06-3, to clarify aspects of the containment leak rate testing program to be certified as part of the DC of the APR1400 and those left for the COL applicant. DCD Tier 2, Table 1.9-1 states that the APR1400 conforms with RG 1.163 (thus, by reference to NEI 94.01 and ANSI/ANS 56.8) and DCD Tier 2, Chapter 16, TS Section 3.6.1 and 5.5.16 which describes a CILRT program. If there are exceptions to the standards and regulatory guidance, they would be specifically identified in the DCD. In its response to RAI 306-8240, Question 06.02.06-3, dated March 3, 2016, the applicant stated that the containment leak rate testing program of the APR1400 conforms to the requirements of 10 CFR Part 50, Option B and follows the guidance of RG 1.163. There are no deviations or exemptions from 10 CFR Part 50, Appendix J or RG 1.163. As required by COL 13.4(1), the program is to be developed by COL applicant. COL Item 6.2(1) will be deleted in order to avoid redundancy with COL 13.4(1). The applicant provided revised markups of associated DCD Tier 2, Table 1.8-2, Table 1.9-7, Section 6.2.6.1 and Section 6.2.8. The staff has reviewed the applicant response and finds it acceptable because it will be performed in accordance with RG 1.163. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-3 resolved and closed.

The staff finds that the DCD description of scheduling and reporting of periodic tests conform to the guidance in RG 1.163 and NEI 94-01 Section 12, "Recordkeeping," to the level of detail described in RG 1.206, Part III, Section C.1.6.2.6.1. Therefore, the staff finds the requirements of GDC 52 and GDC 53, Appendix J to 10 CFR Part 50, and 10 CFR 52.47(a)(2) have been met as these regulations apply to the scheduling and reporting of Type B tests because it conform to the guidance of RG 1.163 and NEI 94-01 as indicated above.

6.2.6.4.3 Containment Isolation Valve Local (Type C) Leakage Rate Testing

Type C tests measure CIV leakage rates.

6.2.6.4.3.1 Identification of Isolation Valves Subject to Type C Testing

DCD Tier 2, Revision 0, Table 6.2.4-1, includes the complete list of isolation valves that will receive preoperational and periodic Type C tests. In addition, this table identifies the isolation valves that do not require leakage rate testing and states the reason for their not requiring this testing. DCD Tier 2, Revision 0, Figure 6.2.4-1 illustrates the containment valves that will receive preoperational and periodic Type C tests.

DCD Tier 2, Section 6.2.6, specifies the use of Appendix J, Option B for Type A, B, and C containment leakage rate testing. RG 1.163, "Performance-Based Containment Leak-Test Program," endorses NEI-01, Revision 0 for an acceptable method for complying with Option B, NEI 94-01 references ANSI/ANS 56.8-1994. ANSI/ANS 56.8 Section 6.3 discusses draining water from CIVs for Type C testing and Section 3.2.5 discusses venting the downstream side of the valve for testing. These vent and drain connections, which are important to ensure accurate test results are not shown on DCD Tier 2, Figure 6.2.4-1. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-5, to provide figures showing the appropriate vent and drain connections for all CIVs that shall be Type C tested in the DCD.

In its response to RAI 306-8240, Question 06.02.06-5, dated April 7, 2016 (ADAMS Accession No. ML16181A349), the applicant stated that DCD Tier 2 Figure 6.2.4-1 will be revised to show the appropriate vent and drain connections for all CIVs that shall be Type C tested. RAI 357-8344 Question 06.02.04-9, dated January 5, 2016, requested similar information, and therefore, the revised DCD Tier 2, Figure 6.2.4-1 will be provided as an attachment to that response (ML16182A591). The staff has reviewed the applicant's response and attachments to Questions 06.02.06-5 and 06.02.04-9, and finds it acceptable because the proposed revised Figure 6.2.4-1 show the appropriate vent and drain connection per guidance of NEI-01 and ANSI/ANS 56.8-1994. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-5 resolved and closed.

DCD Tier 2, Section 6.2.6.3 states that Appendix J, Option B, Type C leakage tests are conducted for all CIVs as specified in DCD Tier 2, Table 6.2.4-1. Section 6.2.6.3, "Containment Isolation Valve Leakage Rate Test (Type C)," also provides three criteria which are used by the applicant to determine which CIV will be Type C tested. However, the applicant's criteria are different from the three cases specified in the NRC staff's guidance. In addition, DCD Tier 2, Table 6.2.4-1 lists all of the CIVs along with a column indicating whether or not Type C leakage testing will be performed, and many of the CIVs listed in table will not have Type C tests performed and many of those do not provide a justification. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-6, to request the applicant to either revise the three criteria in the DCD to be consistent with the three cases in the NRC staff's guidance, or provide justification for using different criteria. Also, the applicant is requested to revise DCD Tier 2, Table 6.2.4-1 to apply the new correct criteria and to include a justification for each valve that is proposed to be excluded from the Type C leak rate test program. In its response to RAI 06-8240, Question 06.02.06-6, dated June 10, 2016 (ML16064A073), the applicant stated that the three criteria for Type C testing of valves in DCD Tier 2, Section 6.2.6.3, will be revised to be consistent with NRC staff's guidance, and provided in associated markups which states that Type C tests of CIVs as specified in Table 6.2.4-1 are performed in accordance with NEI 94-01 and ANS-56.8. The Type C valves which are excluded from Type C leak rate testing in Table 6.2.4-1 were selected in accordance with the NRC staff's guidance. Justifications for each of the valves excluded from Type C testing will be revised in Table 6.2.4-1 on the DCD markup associated with the response to RAI 306-8240, Question 06.02.06-9, dated June 30, 2016. The appropriate vent and drain connections for testing the CIVs are illustrated in Figure 6.2.4-1 and applicable system piping and instrument diagrams. The staff has reviewed the applicant's response and finds it acceptable because the proposed revised Section 6.2.6.3 and Table 6.2.4-1 are consistent with NRC guidance for Type C tests of CIVs in accordance with NEI 94-01 and ANSI/ANS 56.8-1994. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-6 resolved and closed.

The staff finds that all containment valves in lines that constitute potential containment atmospheric leakage pathways during and following a DB LOCA are Type C tested. The staff found the listing of valves acceptable because, the listing follows the guidance of NEI 94-01, Section 6.0, and ANSI/ANS-56.8-1994, Section 3.3.1, which states that Type C testing is not required on test connections, vents and drains between CIVs which are one inch size or less, and are administratively secured closed and consist of a double barrier. Therefore the staff finds that the design conforms to SRP Section 6.2.6, Section II acceptance criteria as it applies to the identification of penetrations that are subject to Type C tests and the identification and justification of those penetrations for which C testing is not required. Consequently, the staff found that the requirements of GDC 52 and GDC 53, Appendix J to 10 CFR Part 50 and 10 CFR 52.47(a)(2) have been met as these regulations apply to the design provisions for Type C testing.

6.2.6.4.3.2 Design Provisions for Type C Testing and Test Methods

DCD Tier 2, Section 6.2.6.3 indicates that Type C tests are performed for CIVs as specified in Table 6.2.4-1. DCD Tier 2, Figure 6.2.4-1, shows the CIV arrangement. The staff has reviewed the DCD description of these test methods and connections in DCD Tier 2, Section 6.2.6.3, Table 6.2.4-1 and Figure 6.2.4-1 and determined that additional information is required. SRP Section 6.2.6 and ANS-56.8 (Section 6.2) specify that all CIVs are to be tested so that pressure is applied in the same direction that would occur in a design basis accident, unless such testing would give equivalent or more conservative results. In order to ensure compliance with the guidance, the staff issued RAI 306-8240, Question 06.02.06-7, dated November 16, 2015, to request the applicant to provide (or indicate where in the DCD application it is provided):

1. A list of those CIVs that will be locally (Type C) leakage tested with test pressure applied in a direction opposite to that would occur in a DB accident.

2. For each CIV identified above, justification that any Type C containment leakage test results conducted in such manner will result in equivalent or more conservative test results.

3. DCD figures that are complete and meet the Type C test requirements and guidance related to test direction or provide the required exemption requests and justification.

In its response to RAI 306-8240, Question 06.02.06-7, dated January 28, 2016 (ML16028A185), the applicant stated that the CIV listed in Table 6.2.4-1 will be tested with test pressure applied in the same direction as that which would result from a DBA, except for valves VQ-V0012, V0013, V0032 and V0033 of Reactor Containment These four valves are a type of butterfly valve with concentric stem and non-tapered seat, and can be tested in the reverse direction according to ANS/ANS 56.8-1994, 6.2, (1). The test direction of these CIVs will be shown on Figure 6.2.4-1 on the markups associated with the response to RAI 306-8240, Question 06.02.06-5 (ML16181A349), and individual system piping and instrumentation diagrams. The staff has reviewed the applicant's response and finds it acceptable because it conforms to the guidance in RG 1.163 and ANSI/ANS 56.8-1994. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-7 resolved and closed.

The staff concludes that the design conforms to the guideline in RG 1.163 and NEI 94-01 Section 6 as it applies to design provisions for Type C testing and test methods. Therefore, the staff finds that the design meets SRP Section 6.2.6, Section II acceptance criteria as it applies to design and test provisions for Type C testing. Consequently, the staff found that the requirements of GDC 52 and GDC53, Appendix J to 10 CFR Part 50 and 10 CFR 52.47(a)(2) have been met as these regulations apply to the design provisions and test methods for Type C testing in accordance with guidance in RG 1.163 and ANSI/ANS 56.8-1994.

6.2.6.4.3.3 Acceptance Criteria for Type C Tests

The staff has reviewed the Type C leakage rate testing acceptance criteria contained in APR1400 DCD Tier 2, Revision 0, Section 6.2.6 and Chapter 16, TS Section 5.5.16. The staff found that the Type C leakage rate test results will be combined with the Type B results, in accordance with Appendix J to 10 CFR Part 50. During the first startup following testing, the leakage rate acceptance criterion will be less than 0.6 La. The staff found that this acceptance criterion for combined Type B and C leakage is acceptable because it conforms to the guidance of NEI 94-01 paragraph 10.2, which is endorsed by RG 1.163. In addition the staff found that this criterion is consistent with the CECE Plants, Revision 4.0 Standard TS (NUREG–1432), Section 5.5, Appendix J, Option B. Consequently, the staff found that this value is in accordance with SRP Section 6.2.6, and thus is in accordance with the provisions of Appendix J to 10 CFR Part 50.

6.2.6.4.3.4 Scheduling and Reporting of Type C Tests

The staff reviewed the DCD Section 6.2.6.4, DCD COL 6.2(1) and DCD Table 1.9-1 for proposed scheduling and reporting requirements associated with performing pre-operational and leakage rate testing. As stated in Section 6.2.6.4 of the DCD, the proposed schedule and test report content requirements associated with performing pre-operational and periodic leakage rate testing are in accordance with NEI 94-01, as modified and endorsed by the NRC in RG 1.163. NEI 94-01 in turn references ANSI/ANS-56.8-1994 for the technical contents of the reports. Per DCD COL 6.2(1), the implementation milestones for pre-operational and periodic CILRT will be developed by each COL applicant. On November 16, 2015, the staff issued RAI 306-8240, Question 06.02.06-3, to clarify in the DCD what aspects of the containment leak rate testing program are to be certified as part of the DC of the APR1400 and what are to be left for the COL Applicant. DCD Tier 2, Table 1.9-1 states that the APR1400 conforms with RG 1.163 (thus, by reference to NEI 94.01 and ANSI/ANS 56.8) and DCD Tier 2, Chapter 16, TS Section 3.6.1 and 5.5.16 which describes a CILRT program. If there are exceptions to the standards and regulatory guidance, the applicant was requested to specifically identify them in the DCD. In its response to RAI 306-8240, Question 06.02.06-3, the applicant stated that the containment leak rate testing program of the APR1400 conforms to the requirements of 10 CFR

Part 50, Option B and follows the guidance of RG 1.163. There are no deviations or exemptions from 10 CFR Part 50, Appendix J or RG 1.163. As required by COL 13.4(1), the program is to be developed by COL applicant. COL Item 6.2(1) will be deleted in order to avoid redundancy with COL 13.4(1). The applicant provided revised markups of associated DCD Tier 2, Table 1.8-2, Table 1.9-7, "Conformance with SECY-93-087," Section 6.2.6.1 and Section 6.2.8. The staff has reviewed the applicant response and finds it acceptable because the proposed scheduling and reporting requirements associated with performing pre-operational and leakage rate testing will be performed in accordance with RG 1.163. The staff confirmed that the RAI markups were incorporated into the Revision 1 of the DCD. Therefore, the staff considers RAI 306-8240, Question 06.02.06-3 resolved and closed.

The staff finds that the DCD description of scheduling and reporting of periodic tests conforms to the guidance of RG 1.163 and NEI 94-01, Section 12, to the level of detail described in RG 1.206, Part III, Section C.1.6.2.6.1. Therefore, the staff found the requirements of GDC 52 and GDC53, Appendix J to 10 CFR Part 50, and 10 CFR 52.47(a)(2) have been met as these regulations apply to the scheduling and reporting of Type C tests.

6.2.6.4.4 Special Testing Requirements

Based on review of DCD Tier 2, Revision 1, Section 6.2.6, the staff finds that the APR1400 design does not rely on containment boundaries sealed with qualified seal system.

Consequently the staff found that there are no special testing requirements that are associated with a seal system. In DCD Tier 2, Section 6.2.6.5, "Special Testing Requirements," the applicant stated that because the APR1400 design does not have a secondary containment, or sub-atmospheric primary containment, or isolation valve seal systems and fluid-filled systems, there are no special testing requirements for these design features. Based on the above review, the staff found that the requirements of 10 CFR 52.47(a)(2), as it relates to the evaluation of the ESF of secondary containment, or a qualified seal system, have been met.

6.2.6.4.5 Technical Specifications

TS 5.5.16, "Containment Leakage Rate Testing Program," states that a testing program shall be established as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B; and in accordance with the guidelines specified in RG 1.163. Key TS parameters and the bases on which the staff accepts these parameters are summarized in DCD Tier 2, Table 6.2.6-1 below:

Table 6.2.6-1 - Key TS Parameters and Bases

Technical Specification	TS Value	Acceptable	Acceptance Basis
Test Pressure (Pa)	3.59 kg/cm^2 (51.09 psig)	Yes	Pa is the peak calculated accident pressure of 3.59 kg/cm^2 (51.09 psig). (DCD Tier 2, Table 6.2.1-2).
Test Pressure (Pa)	3.59 kg/cm^2 (51.09 psig)	Yes	Pa is less than containment design pressure of 4.22kg/cm^{2}Pa (60 psig) (DCD Tier 2, Table 6.2.1-3).

Technical Specification	TS Value	Acceptable	Acceptance Basis
Maximum allowable leakage rate (L_a) at P_a	0.1% per day by weight	Yes	DCD Tier 2, Table 15.6.5-13 shows primary containment leakage rate assumption at 0.1% percent per day starting at time zero to 24 hours into DBA-LOCA. The TS acceptance criterion is equal or conservative with respect to the radiological consequence analysis assumption and therefore is acceptable.
Leakage rate acceptance criteria [as-found] condition	≤1.0 L_a	Yes	ANSI 56.8-1994 and NEI 94-01 (Paragraph 9.2.5) identify 1.0 La acceptable for “as found.”
During first startup following testing per this program [as-left condition]	≤0.75 L_a	Yes	ANSI 56.8-1994 and NEI 94-01 (Paragraph 9.2.5) say 0.75 L_a is acceptable for “as left.”
Type B/Type C Testing	≤ 0.6 La total for Type B and Type C combined	Yes	Appendix J requires combined Type B and C leakage results shall be less than L_a with margin.

As described in Chapter 16 of this SE, the proposed TS for the APR1400 conform to NUREG–0800, Section 16 and NUREG–1432, “Standard Technical Specifications (STS) for Combustion Engineering Plants.” Based on the staff review as discussed above, the staff finds that the CILRT Program described in DCD Tier 2, Section 6.2.6, and the requirements stated in TS Section 5.5.16, satisfy the provisions of RG 1.163. Therefore, the staff finds that the APR1400 TS meets SRP Section 6.2.6 Section II acceptance criteria as it applies to the review of TS related to the CILRT. Consequently, the staff finds that the containment leakage testing program satisfies the requirements of 10 CFR Part 50, Appendix J, Option B and, 10 CFR 50.54(o), as these regulations apply to the TS requirements for the CILRT Program.

6.2.6.5 Combined License Information Items

The CILT program shall be developed by the COL applicant who selects the APR1400 standard design. DCD Tier 2, Section 13.4, “Operational Program Implementation,” COL 13.4(1) requires the development of the CILT program by stating, “The COL applicant is to develop operational programs and provide schedules for implementation of the programs, as defined in SECY-05-0197. The COL applicant is to provide commitments for the implementation of operational programs that are required by regulation. In some instances, the programs may be implemented in phases, where practical, and the applicant is to include the phased implementation milestones.” The staff finds the above COL item adequately describes the action necessary for the COL applicant.

6.2.6.6 Conclusion

The staff finds that the APR1400 CLRT program as described in the APR1400 DCD, Revision 1, along with the COL action item identified by the applicant, conform to RG 1.163 and the applicable provisions of NEI 94-01 and ANSI/ANS 56.8, which are endorsed by the staff; and that proposed TS 5.5.16 includes a requirement for the CLRT program to satisfy the provisions of RG 1.163. The staff finds that the APR1400 CLRT meets the acceptance criteria in SRP Section 6.2.6 in the areas of review discussed above.

Conformance with the criteria in Section 6.2.6 of the SRP, as described in this section, constitutes an acceptable basis for satisfying the requirements specified by 10 CFR Part 50, Appendix J, Option B. Therefore, the staff finds that the requirements specified in GDC 52, GDC 53, and GDC 54 and 10 CFR 50.54(o) are also satisfied.

Such compliance assures that leak tight integrity of the containment can be verified periodically throughout its service life to ensure that leakage rates are maintained within the limits of the TS. Maintaining containment leakage rates within such limits assures that, the containment design reflects those assumptions used for the design and performance of ESF in site selection as required by 10 CFR Part 50.47(a)(2), and that, in the event of any radioactivity releases within the containment, the radiological release through containment leak paths will not be in excess of the acceptable limits for the site.

6.2.7 Fracture Prevention of Containment Pressure Boundary

6.2.7.1 Introduction

6.2.7.2 Summary of Application

The containment vessel of the APR1400 is a prestressed concrete structure with ferritic parts, such as a liner. The ferritic parts are made of materials that have a nil ductility transition temperature sufficiently below the minimum service temperature to ensure that, under operating, maintenance, testing, and postulated accident conditions, the ferritic materials behave in a non-brittle manner, considering the uncertainties in determining the material properties, stresses, and size of flaws. In DCD Tier 2, Table 6.1-1, the applicant identified the containment vessel liner materials. The applicant further identified all ferritic attachment and appurtenance materials as conforming to ASME, Code Section III, Division 1, Subarticle NE-2300. The applicant required that COL applicants are to provide identification of all weld filler metals to be used with the above and that these materials are to conform to ASME Code, Section III and ASME Code, Section II, as applicable, and/or ASME Code Cases approved by the NRC.

6.2.7.3 Regulatory Basis

The staff reviewed APR1400 DCD Tier 2, Revision 0, Section 6.2.7 in accordance with SRP Section 6.2.7, "Fracture Prevention of Containment Pressure Boundary," Revision 1, issued March 2007.

The reactor containment system includes the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products. Fracture of the containment pressure boundary should be prevented for it to fulfill its design function. The APR1400 design should address the following regulations:

- GDC 1, requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Section 6.1.1 addresses the applicant's discussion and the staff's evaluation.

- GDC 16, requires that the reactor containment and associated systems establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Section 6.2.4 addresses the applicant's discussion and the staff's evaluation.

- GDC 51, "Fracture prevention of containment pressure boundary," requires that the reactor containment boundary be designed with sufficient margins to assure that, under operating, maintenance, testing, and postulated accident conditions, (1) its ferritic materials behave in a non-brittle manner, and (2) the probability of a rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) flaw size.

The staff reviewed the DCD to ascertain whether the containment pressure boundary materials meet the requirements of GDC 51.

6.2.7.4 Technical Evaluation

The staff reviewed the information included in DCD Tier 2, Section 6.2.7, in accordance with the guidance provided in SRP Section 6.2.7. These ferritic materials are acceptable if they meet the requirements of GDC 51, as it relates to the reactor containment pressure boundary being designed with sufficient margins to ensure that, under operating, maintenance, testing, and postulated accident conditions, the ferritic materials will behave in a non-brittle manner and the probability of a rapidly propagating fracture will be minimized.

The applicant stated that the containment liner plate; all ferritic attachment and appurtenances, and associated weld materials will conform to ASME Code, Section III, Subarticle CC-2520, "Fracture Toughness Requirements for Materials," and ASME Code, Section II material specifications. Based on its review, the staff finds that the ferritic materials of the reactor containment pressure boundary will (or will be where appropriate) acceptably tested and demonstrated to meet the fracture toughness requirements for Class MC components as specified in Article NE-2300 of ASME Code Section III, Division 1 or Class CC components as specified in Article CC-2520 of ASME Code Section III, Division 2. A full evaluation of the use of NE and CC is provided in DCD Tier 2, Sections 3.8.2, and 3.8.1, respectively. The DCD states that COL applicant are to provide the weld filler material in the supplier specifications via COL 6.2(1). The staff finds this acceptable as it improves assurance of traceability of the weld filler material.

6.2.7.5 Combined License Information Items

Table 6.2.7-1 lists the item numbers and descriptions from Section 6.2.8 of the DCD.

Table 6.2.7-1 Combined License Information Items

Item no.	Description
6.2(1)	The COL applicant is to provide the weld filler material in the supplier specification.

6.2.7.6 Conclusion

Based on the review of the information included in the APR1400 DCD, the staff finds that the fracture toughness of the materials used in the reactor containment pressure boundary meets the fracture toughness requirements specified in GDC 51.

The staff concludes that, under operating, maintenance, testing, and postulated accident conditions, the APR1400 provides reasonable assurance that the materials used in the reactor containment pressure boundary will not undergo brittle fracture and that the probability of a rapidly propagating fracture will be minimized, thereby meeting the requirements of GDC 51.

6.3 Emergency Core Cooling System/Safety Injection System

6.3.1 *Introduction*

The ECCS, otherwise known as the SIS, functions to supply an active and passive safety-related source of borated water as emergency core cooling for the APR1400. The SIS design is intended to limit fuel assembly damage in order to maintain a coolable core geometry, and supply emergency short- and long-term borated water for core cooling following a LOCA. Other accidents for which the SIS is designed to mitigate include the steam generator tube rupture (SGTR), the steam line break (SLB), and the control elements assembly ejection (CEAE) accidents. The active components of the SIS are found in four mechanically separated trains, each consisting of a SIP and associated valves. Each SIP takes its own direct suction from the IRWST and discharges to its own DVI nozzle on the reactor vessel. The passive components of the SIS are four identical pressurized SITs each containing borated water pressurized with nitrogen gas. A fluidic device (FD) at the tank exit inside the SIT is designed to passively control the flow rate based on water level inside the SIT. The SI water from each SIT is routed to its own respective DVI line where it combines with the SIP flow ultimately discharging into the reactor vessel via the four separate DVI nozzles.

6.3.2 *Summary of Application*

DCD Tier 1: Tier 1 information associated with the SIS is found in DCD Tier 1, Sections 2.4.2, “In-Containment Water Storage System,” and 2.4.3, “Safety Injection System.”

DCD Tier 2: Information provided by the applicant in DCD Tier 2, Section 6.3, “Safety Injection System,” is summarized here, in part as follows:

The DCD provides information regarding the SIS design as a whole and on a component basis. The DCD details the functional, reliability, protection, and environmental design bases, including design requirements, power source requirements, SF requirements, design basis environment requirements, missile protection, seismic design, and water hammer requirements, testing and inspection requirements, instrumentation and system actuation signal requirements, as well as requirements for minimizing contamination.

The applicant designed the APR1400 SIS based on the SIS of the System 80+ Certified Design, Appendix B to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

The staff reviewed the application in accordance with the NUREG–0800, SRP Section 6.3, "Emergency Core Cooling System," the guidance provided in applicable RGs, and the NRC's regulations.

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.3 are given in DCD Tier 1, Section 2.4.2, Table 2.4.2-4 and in Section 2.4.3, Table 2.4.3-4, "Safety Injection System ITAAC."

TS: The TS applicable to the SIS can be found in DCD Tier 2, Chapter 16, Section 3.5, "Emergency Core Cooling System (ECCS)."

Initial Plant Testing: Initial plant testing of the SIS is discussed in DCD Tier 2, Section 14.2, "Initial Plant Testing Program," which specifies testing that is applicable to the SIS in DCD Tier 2, Section 14.2.12.1.21, "Safety Injection System Test," DCD Tier 2, Section 14.2.12.1.22 and DCD Tier 2, Section 14.2.12.1.59, "Pre-Core Safety Injection Check Valve Test."

6.3.3 *Regulatory Basis*

The relevant requirements of Commission regulations for this area of review, and the associated acceptance criteria, are given in NUREG–0800, Section 6.3, "Emergency Core Cooling System," and are summarized below. Review interfaces with other SRP sections can also be found in NUREG–0800 Section 6.3.

- GDC 2, as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of an SSC important to safety, including the ECCS, to perform its safety function.

- GDC 4, as it relates to dynamic effects associated with the environmental conditions associated with normal operation and accident conditions, among other things, and that include flow instabilities and loads (e.g., water hammer).

- GDC 5, "Sharing of Structures, Systems, and Components," as it relates to SSCs important to safety shall not be shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.

- GDC 17, as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded during anticipated operational occurrences and that the core is cooled during accident conditions.

- GDC 27, "Combined Reactivity Control Systems Capability," as it relates to the reactivity control system design having the capability in conjunction with poison addition by the ECCS, to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

- GDC 35, as it relates to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling.

- GDC 36, "Inspection of Emergency Core Cooling System," as it relates to the ECCS being designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

- GDC 37, "Testing of Emergency Core Cooling System," as it relates to the ECCS being designed to permit appropriate periodic pressure and functional testing.

- The regulations in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," in regard to the ECCS being designed so that its cooling performance, as predicted by an acceptable evaluation model, is in accordance with specified acceptance criteria.

- The regulations in 10 CFR 50.34(f)(2)(xxvi), with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the ECCS outside of the containment that contain or may contain radioactive material following an accident.

- The regulations in 10 CFR 50.34(f)(2)(xi), as it relates to the requirements that direct indication of RCS relief and safety valve position be provided in the control room.

- The regulations in 10 CFR 50.34(f)(2)(xviii), with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling.

- The regulations in 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations.

Acceptance criteria adequate to meet the above requirements can be found in SRP 6.3, and include, but are not limited to, the following:

- RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)," November 1970.

- RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident," March 2012.

- RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," February 2010.

- RG 1.29, "Seismic Design Classification," March 2007.

- RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989.

- Design certification/Combined license-Interim Staff Guidance (DCD/COL-ISG)-019, "Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems," July 2010.

6.3.4 Technical Evaluation

6.3.4.1 Functional Design Bases

The staff noted that the applicant designed the SIS to achieve two major functions: SI and safe shutdown. The staff confirmed that the SIS consists of active and passive injection components for use in mitigating LOCAs, SGTRs, SLBs, and CEAE accidents. The staff noted that the following functions should be performed by the SIS:

- Core cooling to prevent fuel damage
- Post-accident decay heat removal
- Control of reactivity changes to maintain core subcriticality

The staff also noted that the design provides for the active portion of the SIS to be mechanically separated amongst four trains, where each train consists of a SIP and associated piping and valves. Each SIP has its own suction line from the IRWST and its own discharge line to a DVI nozzle on the reactor vessel. The DVI nozzle directs borated injection water directly into the reactor vessel downcomer region. The staff further noted that the passive portion of the SIS consists of four identical pressurized SITs with FDs.

6.3.4.1.1 Loss-of-Coolant Accident

As the applicant described in Section 15.6.5, "Loss-of-Coolant Accidents Resulting from the Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," of the DCD Tier 2, during a LOCA, the RCS pressure decreases due to a breach in the system. Once the pressurizer pressure drops to the low-pressurizer pressure setpoint, the safety injection actuation signal (SIAS) is generated and SI flow is initiated by the SIPs. When the RCS pressure decreases below the SIT pressure setpoint, the SIT's borated water starts to discharge into the DVI lines where it is routed directly to the reactor vessel downcomer. The applicant designed the SIS to provide the required minimum injection flow rate for large break LOCAs and small break LOCAs assuming the most limiting SF.

For a large break LOCA (LBLOCA) where the RCS depressurization is quick relative to a small break LOCA (SBLOCA), the applicant designed the SITs to provide a means of rapid reflooding of the core until flow from the SIPs becomes available. The FD in the SITs passively controls SIT injection flow rate during operation as discussed in the applicant's TR, APR1400-Z-M-TR-12003-P. The FD maintains a low flow rate from the SIT after the initial refill phase to ensure the core remains covered and ensure minimum spillage from the break during the reflood phase. The SIPs begin injecting borated water from the IRWST following the SI actuation signal. The operators of the plant accomplish long term cooling following a LBLOCA by manually realigning the SIS for simultaneous hot leg and DVI nozzle injection. The operators align two of the four SIPs to discharge to the RCS hot legs. The applicant intends for the hot leg realignment to provide flushing flow to preclude boron precipitation and the ultimate subcooling of the core when shutdown cooling cannot be used. The staff noted that this is a widely-accepted industry practice.

For a SBLOCA, the RCS pressure remains relatively high for a longer period of time following the event initiator. The applicant designed the SIPs to provide reasonable assurance that the injected flow is sufficient to meet the design bases for the SIS. If necessary, the operators can throttle the SIP flow to assist in RCS depressurization to conditions that allow for the initiation of shutdown cooling for the long-term cooling mode. The staff noted that this too is a widely-accepted industry practice.

The applicant provided a safety analysis for LOCAs in DCD Tier 2, Section 15.6.5, “Loss-of-Coolant Accidents resulting from the Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary.” The staff reviewed this safety analysis as part of Chapter 15 of this SE.

6.3.4.1.2 Steam Line Break Accident

The staff noted that a steam line break causes a cooldown event due to an increase in the steam flow rate, which causes excessive energy removal from the RCS. The excessive energy removal causes a decrease in temperature and pressure in the RCS. Cooling down the RCS causes a positive insertion of reactivity, thus requiring a need to counteract with negative reactivity. Following the decrease of RCS pressure to the low pressurizer pressure setpoint, the engineered safety features actuation system (ESFAS) generates the SIAS causing initiation of the SIS. The SIPs introduce negative reactivity via borated water from the IRWST. In some SLB accidents (e.g. small breaks) the RCS pressure will not decrease below the SIT pressure; therefore, SIT injection is precluded for these types of SLBs (see Chapter 15.1.5). The staff confirmed that the SIS is designed to mitigate against steam line breaks in the event the SIS is called into action (e.g. when pressurizer pressure reaches the SIAS setpoint).

The applicant provided a safety analysis of the steam line break accidents in DCD Tier 2, Section 15.1.5, “Steam System Piping Failure Inside and Outside the Containment.” The staff reviewed this safety analysis as part of Chapter 15 of this SE.

6.3.4.1.3 Steam Generator Tube Rupture Accident

The staff noted that a SGTR accident occurs due to the rupture of a SG U-tube and consists of an RCS barrier breach and a potential for containment bypass. Radiological release is the greatest concern for this accident due to primary system coolant mixing in the shell side of the SG and being transported through the turbine to the main condenser where the non-condensable gas air ejectors evacuate the radioactive non-condensable gases to the environment. The accident progression ends when the operators are able to cooldown and depressurize the primary system to a pressure below that of the ruptured SG.

The staff noted that SI during a SGTR causes the primary system depressurization rate to slow down and increases the primary-to-secondary leakage flow. Maximum SI flow has a greater influence on radiological release; however, maximum SI flow also prevents void formation in the upper head during a SGTR concurrent with a LOOP. The applicant provided a safety analysis of the SGTR accident in DCD Tier 2, Section 15.6.3, “Steam Generator Tube Failure.” The staff reviewed this safety analysis as part of Chapter 15 of this SER.

6.3.4.1.4 Control Elements Assembly Ejection Accident

The staff noted that a control element assembly (CEA) ejection accident occurs when a mechanical failure causes an instantaneous rupture of the control element drive mechanism housing or associated nozzle. The RCS pressure causes the CEA and drive shaft to be ejected to the fully withdrawn position. Since the control element drive mechanism housing is a part of the RCPB, a LOCA ensues after the ejection. The staff noted that the SIS, thus, actuates to maintain coolant inventory and shutdown margin. The applicant provided a safety analysis of the control element assembly ejection accident in DCD Tier 2, Section 15.4.8, “Spectrum of Control Element Assembly Ejection Accidents.” The staff reviewed this safety analysis as part of Chapter 15 of this SE.

6.3.4.1.5 Summary of SIS Design for Accident Mitigation

The staff considered the information provided regarding loss of coolant, steam line break, SGTR and control elements assembly ejection accidents and finds that the description provided in DCD Tier 2, Section 6.3 includes a complete discussion of the SIS functional design bases for the APR1400. Evaluation of the SIS performance is included in Chapter 15 of this SE. The staff further concludes that the applicant considered all postulated accident scenarios where SIS involvement in accident mitigation is credited; and thus, designed the SIS utilizing the appropriate functional design bases.

6.3.4.2 System Design

The staff reviewed the SIS design and confirmed that the SIS has four redundant, independent SI trains. Each train consists of an accumulator, otherwise known as a SIT with a passive flow controlling FD, a SIP, and related piping and valves. The SIS takes suction from the IRWST and delivers it to the RCS via the DVI nozzles. A single suction line connects each train to the IRWST. Furthermore, the SIP discharge is directly routed to the DVI nozzles located on the RPV. Each SIT discharges to its own DVI line shared by one of the SIPs. A check valve exists on each DVI line upstream of the SIT discharge line connection to prevent SIT injection flow from flowing away from the RPV after it enters the DVI line. In typical PWRs, the low-head SIPs also act as part of the residual heat removal system. The APR1400 has an equivalent residual heat removal system, known as the shutdown cooling system; the applicant described its design in DCD Tier 2, Section 5.4.7, “Shutdown Cooling System,” and the staff reviews that as part of Chapter 5 of this SE. The staff noted that the APR1400 SIS does not consist of low-head SI; instead, the applicant designed the SIS such that the SIPs have sufficient capacity to satisfy the functional requirements for decay heat removal through feed and bleed operations (e.g. when decay heat removal by the SCS cannot be attained). The staff noted that the operators can align two SIPs to discharge into the hot leg to provide a flushing flow in the core in order to preclude boron precipitation during long-term cooling.

The staff confirmed that the applicant designed each train of SI to be capable of being powered from the onsite power source, offsite power source, and emergency power source through four independent electrical trains with each train providing power to each bus. The applicant also designed each EDG and its associated automatic sequencer to ensure SI flow is delivered to the reactor vessel within a maximum of 40 seconds after a SIAS. The applicant described the EDG design requirements in DCD Tier 2, Section 8.3.1, “AC Power Systems,” the staff reviewed these requirements as part of Chapter 8 of this SE. The staff confirmed that the SIS delay times are appropriate and are used in the applicant’s accident analyses. Furthermore, the staff confirmed that the SIS is designed to seismic Category I requirements and designed to withstand design bases environments, including protection from RCS generated missiles, to the extent practical.

The staff confirmed that the SIAS automatically initiates the functions of the SIS and that no manual operator action is necessary to initiate SI. The water source is the IRWST, which is located at the bottom of containment and is continually replenished with water spilled from the break and steam condensed in the containment. The staff also confirmed that no manual realignment of piping nor valves is necessary to initiate SI. Later in the accident, to prevent boron precipitation in the reactor vessel and to mitigate steaming from the break, the staff noted that the applicant designed the SIS with the capability of hot leg injection mode. This occurs two to three hours into the event and is done manually. The operators control the SIS from the MCR for all operating conditions; however, the remote shutdown room also provides capability for SIS operation in the event that the MCR is uninhabitable.

The applicant provided an ITAAC in DCD Tier 1, Table 2.4.3-4, "Safety Injection System ITAAC," which indicates that SIS controls required by the design are provided in the remote shutdown room (RSR). In Revision 0 of the DCD, the applicant provided no discussion in Tier 2, Section 6.3 about SIS controls being provided in the RSR. The staff could not conclude that the Tier 1 ITAAC item in question was derived from the Tier 2 information. Following the guidance provided in SRP Section 14.3.4, "Reactor Systems – Inspections, Tests, Analyses, and Acceptance Criteria," the staff observed that all Tier 1 information should be derived from Tier 2 information. Therefore, on August 20, 2015, the staff issued RAI 158-7997, Question 06.03-1 to address this issue (ML15295A497). On September 17, 2015, the applicant provided a response to RAI 158-7997, Question 06.03-1 (ML15260B338). In its response, the applicant stated that DCD Tier 2, Section 6.3 will be revised to include discussion and description of the SIS controls which are provided in the RSR, thus enabling the Tier 1 information to be derived from the Tier 2 information. The applicant also presented in its response to the staff a proposed revision to the DCD. Based upon the staff's review of the applicant's Tier 2, Section 6.3 draft revisions, the staff finds that the applicant's Tier 2 SIS information supports the applicant's Tier 1 information, including the associated ITAAC. Furthermore, the staff concludes that the Tier 1 SIS information is sufficient to meet the intent of 10 CFR 52.47(b)(1) because it is derived from the Tier 2 information. On March 10, 2017, the applicant submitted Revision 1 of the DCD. The staff confirmed that Revision 1 of the DCD contained the appropriate revisions, which were proposed as part of the response to RAI 158-7997, Question 06.03-1. Therefore, RAI 158-7997, Question 06.03-1 is resolved and closed.

NRC staff identified that Safety Injection ITAAC 9.d, of Tier 1, Table 2.4.3-4 contains acceptance criteria that is not adequate to demonstrate compliance with the associated design commitment. Specifically, the acceptance criteria for Safety Injection ITAAC 9.d does not require the as built-SI pumps to deliver full flow in accordance with the design commitment. Accordingly, NRC staff issued RAI 558-9456, Question 14.03.01-1-X (ML18074A402), which requested that the applicant modify the ITAAC acceptance criteria to be consistent with the design commitment. In its May 18, 2018, response (ML18137A480), the applicant provided an update that modified the acceptance criteria for Safety Injection ITAAC 9.d to be consistent with the design commitment. This update is being tracked as a **confirmatory item 6.3-1.**

The applicant presented the design basis of the SIS in DCD Tier 2, Section 6.3.1, "Design Bases." The staff compared the specific design specifications with the acceptance criteria given in SRP Section 6.3. The staff finds that the APR1400 design specifications meet the SRP acceptance criteria with a few exceptions described below, where additional information is needed to establish compliance with the GDC, and to the SRP acceptance criteria.

6.3.4.2.1 Accumulators

The staff noted that the four APR1400 SITs are similar to typical PWR accumulators. The basic design is a stainless steel cladded carbon steel tank, which stores borated water pressurized by nitrogen. The injection water from the tank is routed to a DVI nozzle via a SIT discharge line connected to the DVI line (which is shared by the discharge of one SIP). The SIT injection flow path includes two check valves that isolate the tank from the RCS during normal operation. When the RCS pressure falls below the tank pressure, the check valves open, discharging the contents to the DVI nozzle. No operator action or electrical signal is required for the SITs to operate. Together, the SITs provide a means of rapidly refilling the downcomer following the onset of a LBLOCA and keeping it covered until the flow from the SIPs becomes available. The applicant provided the SITs with connections for filling, draining, pressurizing, venting, relieving, and sampling. Additionally, the staff noted that the applicant provided pressure and level instrumentation and alarms in the MCR as part of the design.

Through the review of the system description and the associated piping and instrumentation diagrams, the staff confirmed that the applicant classified the SITs as seismic Category I, that they are normally pressurized to 610 psig, and that they are filled with 52.6 m^3 of borated water with a concentration anywhere from a minimum of 2,300 ppm to a maximum of 4,400 ppm. The borated water's operating temperature ranges anywhere from 10 °C to 48.9 °C.

A passive flow controlling FD described in TR APR1400-Z-M-TR-12003-P, Revision 0, "Fluidic Device Design for the APR1400," is installed in the bottom of the SIT to provide two operational stages of safety water injection into the RCS, large flow mode and small flow mode. The applicant designed the FD to be a cylindrical vortex chamber with eight different inlet ports. Four inlet ports, called main supply ports, are supplied with SI water via a standpipe installed directly on top of the FD. When the tank water level drops below the top of the standpipe, SI water is no longer provided to the vortex chamber via the main supply ports. The other four inlet ports, called the control ports, hydraulically connect the whole tank with the vortex chamber and supply SI water continuously to the vortex chamber until the tank empties. The applicant designed the supply ports to face the control ports in such a way that as water injects into the vortex chamber via the supply and control ports (e.g. large flow mode), vortexing is minimized (i.e. flow resistance is minimized). When water injects into the vortex chamber via the control ports alone (e.g. small flow mode), vortexing is maximized (i.e. flow resistance is maximized) due to the angle at which the control ports face the vortex chamber.

At the center of the vortex chamber is the exit port, directing SI flow out of the tank and into the SIT discharge line. The flow switch from large flow mode to small flow is thus accomplished passively. The TR, APR1400-Z-M-TR-12003-P, details the applicant's full-scale testing of the SIT-FD. The staff has approved the TR in regards to the full-scale testing validating the design of the FD in accordance with the APR1400 SI performance requirements and endorses its use for the licensing review of the APR1400 DC application (ML15006A059).

The applicant designed the SIT for an internal gauge pressure of 700 psig. Safety relief valves accompany each accumulator and prevent an overpressure event from happening due to the maximum expected fill rate of liquid or gas into the SITs. The safety relief valves discharge into containment and have a set pressure of 700 psig with a capacity of 169,900 standard liters per minute (slpm) (6,000 standard cubic feet per minute, scfm) of gas or 870.6 L/min (230 gpm) of liquid.

The staff reviewed the design of the SIT-FD in accordance with the NUREG–0800, SRP Section 6.3, and the NRC's regulations. The applicant's SIT design was evaluated by the staff and determined to conform to all applicable GDC and NRC regulations because the SIT is designed to seismic Category I standards; has adequate pressure, volume, and boron concentration for safety injection; and has been appropriately tested to validate its design principles.

6.3.4.2.2 Safety Injection Pumps

The SIS consists of four identical SIPs to provide borated IRWST cooling water to the core in the event of a design basis accident.

The design flow of a single SIP is 3,085 L/min (815 gpm) at 868.7 m (2,850 ft) design head. The minimum net positive suction head required ($NPSH_r$) of the pumps at a maximum flow rate of 4,675 L/m (1,235 gpm) is 6.7 m (22 ft). Each SIP is equipped with its own minimum flow protection to prevent damage of the pump resulting from dead-headed operation.

The SIPs are equipped with ultrasonic flow meters to provide low flow alarms in the event that the pump is operating at a flow rate less than the required minimum flow. The pumps also have mechanical shaft seals provided with leakoff control to collect any leakage past the seals. The pumps have drain and flushing connections to facilitate a reduction of contamination before maintenance. The internals which come in contact with RCS water are stainless steel to permit compatibility with boric acid solutions.

The applicant stated in DCD Tier 2, Section 6.3.2.1.2, “Safety Injection System,” that “long-term cooling mode for a LBLOCA is accomplished by manually realigning the SIS for simultaneous hot leg and DVI nozzle injections.” The applicant also submitted technical report, APR1400-F-A-NR-14003-P, Revision 0, “Post-LOCA Long Term Cooling Evaluation Model,” which provides the long-term cooling and boron precipitation analysis of the APR1400. The staff reviewed the post-LOCA long term cooling technical report as part of Chapter 15 of this SE.

The staff reviewed the design of the SIPs to confirm that adequate protection is provided for small break sizes down to and including the maximum break size for which normal CVCS makeup is utilized. The staff’s SE regarding the applicant’s small-break LOCA analysis is documented in Section 15.6.5 of this SE.

6.3.4.2.3 Net Positive Suction Head

In order to be in compliance with GDC 35, the SIS pumps shall perform their intended functions during postulated accidents, among other things, such as the capability to perform said functions assuming a SF in the system as well as unavailability of offsite power. To show how the SIS pumps provide abundant emergency core cooling, the applicant has committed to RG 1.1, “Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps,” November, 1970, which has since been withdrawn and incorporated into RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” March, 2012, to which the applicant has also committed itself. In order to meet RG 1.82, the SIS should be designed so that sufficient NPSH margin is provided to the system pumps assuming the maximum expected temperature of the pumped fluid and no credit for containment pressurization during the accident. Additional guidelines for evaluating the adequacy of pump performance and the availability of the sump for recirculation cooling following a LOCA are presented in RG 1.82. The staff’s evaluation of whether the inlet design of the containment sump suction screen, assures that the containment sumps provide a reliable, long-term recirculation cooling capability, ECCS pump performance will be adversely affected by post-LOCA conditions impacting the sumps, and system operation and performance is consistent with the guidance of RG 1.82, is contained in Section 6.2.2 of this SE.

The applicant stated in DCD Tier 2, Section 6.3.2.2.3, “Safety Injection Pumps,” that the available NPSH is calculated in accordance with NRC RG 1.1/RG1.82. Other than the NPSH required, and the minimum NPSH available presented in DCD Tier 2, Table 6.3.2-1, the applicant provided no other information regarding the NPSH evaluation. Therefore, on August 5, 2015, the staff issued Audit Plan, "Staff Regulatory Audit Plan Regarding the Reactor, Reactor Coolant System, and Connecting Systems, and Auxiliary Systems as part of the Review of the APR1400 Design Control Document," which included auditing the NPSH calculation (ML15245A547). The staff reviewed the applicant's calculation regarding the NPSH margin for the SIPs and documented the audit results in the audit report (ML#########). The staff confirmed that the applicant calculated the NPSH for the SIPs in accordance with RG 1.1 and verified the value presented in DCD Tier 2, Table 6.3.2-1 is accurate.

The applicant provided an ITAAC item, 9.c, in DCD Tier 1, Table 2.4.3-4 that states "[t]he SI pumps have sufficient net positive suction head (NPSH)." To meet this design commitment, the applicant proposed "[t]ests to measure the as-built SI pump suction pressure will be performed. Inspections and analysis to determine NPSH to each SI pump will also be performed." The applicant's acceptance criteria states that "[a] report exists and concludes that the as-built NPSH available to each SI pump is greater than the NPSH required of 6.7 m (22 ft)." The staff determined that the NPSH required, based on DCD Tier 2, Section 6.3.2, "System Design," Table 6.3.2-1, "SIS Component Parameters," is 6.7 m (22 ft), which is calculated for the long-term cooling mode, e.g. post-, worst case, LOCA conditions. In accordance with GDC 35, 10 CFR 50.46, and 10 CFR 52.47(b)(1), and by following the guidance provided in SRP Section 6.3, the staff determined that ITAAC item 9.c, of DCD Tier 1, Table 2.4.3-4, shall be revised to ensure that the tests, inspections, and analyses are done assuming appropriate, worst case conditions (e.g. maximum expected fluid temperatures, no increase in containment pressure from that present prior to postulated LOCAs, minimum IRWST water level, etc.). Therefore, on August 20, 2015, the staff issued RAI 158-7997, Question 06.03-4 to address this issue (ML15295A497). On September 17, 2015, the applicant provided a response to RAI 158-7997, Question 06.03-4 (ML15260B338). In its response, the applicant stated that ITAAC item 9.c of Tier 1, Table 2.4.3-4 will be revised to include language regarding the worst case LOCA conditions. The applicant also presented in its response to the staff how the language will look in the next DCD revision. In the applicant's proposed markups, the staff noted that language was added to ensure this ITAAC analysis would be completed assuming appropriate, worst case conditions. The staff found this part of the response acceptable because now this ITAAC analysis represents appropriate conditions for which it should be performed. However, also in the proposed markups, the applicant deleted "6.7 m (22 ft)" from the acceptance criteria. The staff found this to be unacceptable because without the specified value in the acceptance criteria, the ITAAC becomes insufficient and 10 CFR 52.47(b)(1) requires, in part that the ITAAC be sufficient to provide reasonable assurance that the as-built facility will operate in conformity with the DC. Therefore, on June 17, 2016, the staff issued supplemental RAI 496-8630, Question 06.03-11 to address this issue (ML16169A366). The applicant responded to RAI 496-8630, Question 06.03-11, dated July 20, 2016 with a proposed revision to the DCD that added the effective required NPSH value back to the ITAAC acceptance criteria (ML16202A539), which the staff confirmed to be consistent with Tier 2 and thus acceptable. On March 10, 2017, the applicant submitted to the NRC Revision 1 of the DCD (ML17096A391). The staff confirmed that Revision 1 of the DCD contained the appropriate revisions, which were proposed as part of the response to RAI 158-7997, Question 06.03-4 and RAI 496-8630, Question 06.03-11. Therefore, these RAI questions are resolved and closed.

6.3.4.2.4 Piping and Valves

The applicant designed the APR1400 SIS piping to deliver borated water from the SITs and from the IRWST to the reactor vessel DVI nozzles. The applicant designed the DVI nozzles to restrict flow from the reactor vessel in the event of a DVI line break. The design requires piping that is fabricated from austenitic stainless steel and designed to ASME Section III. Furthermore, the applicant designed the SIS piping to seismic Category I standards with vents and drains to minimize potential for water hammer. In addition, the applicant designed the piping to be equipped with relief valves to protect against over pressurization within the SIS.

The APR1400 safety injection filling tank (SIFT) system, in conjunction with the system vents, prevents and removes gas accumulation at the top of the SIS piping. The APR1400 has two SIFTs fabricated from austenitic stainless steel which contain, normally, 3.79 m^3 (133.68 ft^3) of borated water concentrated up to 4,400 ppm. The SIFTs operate at atmospheric pressure. The SIFTs gravity fill the SIS piping every month in accordance with TS Surveillance Requirements (SR). Each SIFT is connected to two SIS trains and one shutdown cooling train. The boric acid makeup pump supplies borated water from the boric acid storage tank to the SIFTs. The SIFTs are equipped with level and temperature indicators and vent to the gaseous waste management system. The SIFTs are located in the auxiliary building at an elevation of 47.9 m (157 ft).

In DCD Tier 2, Section 6.3.1.6, "Protection Design Bases," the applicant stated "[t]he SIS is designed to seismic Category I requirements except safety injection filling tank (SIFT), piping, and valves in the SIS fill line." Per DCD Tier 2, Table 3.2-1, the applicant declared the SIFT as non-nuclear safety (NNS) seismic Category III. The piping and instrumentation diagram, DCD Tier 2, Figure 6.3.2-1, shows the SIFT and piping upstream of valve 700 to be DII and valve 700 and downstream of it to be BI classified. The staff was unable to determine the SIFT seismic classification; therefore, on August 20, 2015, the staff issued RAI 158-7997, Question 06.03-5 to address this issue (ML15295A497). On September 17, 2015, the applicant provided a response to RAI 158-7997, Question 06.03-5 (ML15260B338). In its response, the applicant clarified that the SIFT is designed to non-nuclear safety (NNS) seismic Category III standards. However, the staff noted that no DCD markup was generated to correct the inconsistency between DCD Tier 2, Figure 6.3.2-1 and DCD Tier 2, Table 3.2-1. Therefore, the staff closed RAI 158-7997, Question 06.03-5 and on June 17, 2016, the staff issued supplemental RAI 496-8630, Question 06.03-12 as follow-up to address this inconsistency issue (ML16169A366). The applicant responded to RAI 496-8630, Question 06.03-12, dated July 20, 2016, with a proposed revision to the DCD that corrected the inconsistency between DCD Tier 2, Figure 6.3.2-1 and DCD Tier 2, Table 3.2-1 (ML16202A539) which the staff found acceptable because it removed the inconsistency. On March 10, 2017, the applicant submitted to the NRC Revision 1 of the DCD (ML17096A391). The staff confirmed that Revision 1 of the DCD contained the appropriate revisions, which were proposed as part of the response to RAI 496-8630, Question 06.03-12. Therefore, this RAI question is resolved and closed.

In addition to the SIFTs, the applicant designed vents on the SIS lines to be used in conjunction with plant operating and maintenance procedures to minimize potential for water hammer. TS SR require that plant operators verify that SIS piping locations susceptible to gas accumulation are sufficiently filled with water. Selection of these locations susceptible to gas accumulation is based on a review of system design information, piping and instrumentation diagrams (P&IDs), isometric drawings, plan and elevation drawings, and calculations. The review is supplemented by system walk downs to validate system high points and to confirm the locations and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Following the guidance provided in DC/COL-Interim Staff Guidance (ISG)-019, the staff determined that the applicant did not provide an ITAAC to require a COL holder to compare the as-built plant configuration to the P&ID and isometric drawings to confirm that potential gas accumulation locations and intrusion mechanisms in systems important to safety (i.e. the SIS) have been properly identified and that appropriate prevention measures are in place. Therefore, on

May 26, 2016, the staff issued RAI 492-8614, Question 05.04.07-4, in part, to address this issue (ML16147A594) for SIS along with other important systems such as SCS and CSS. The applicant responded to RAI 492-8614, Question 05.04.07-4, dated July 8, 2016, with a proposed revision to the DCD that added an ITAAC item for verifying the emergency core cooling function of the SIS will not be impaired by gas accumulation (ML16190A320), which the staff found acceptable because it adequately addressed the ITAAC discussed in ISG-019. On March 10, 2017, the applicant submitted to the NRC Revision 1 of the DCD (ML17096A391). The staff confirmed that Revision 1 of the DCD contained the appropriate revisions to Tier 1 for the SIS, which were proposed as part of the response to RAI 492-8614, Question 05.04.07-4. With regard to the SIS, this RAI is resolved. However, the closure of RAI 492-8614, Question 05.04.07-4 is being tracked in Section 5.4.7 of this report because that is the section in which the RAI was asked.

The applicant designed the SIS injection lines with check valves and local vent and drain lines for periodic leak testing. The applicant located the SIS check valves used for containment isolation as close as practical to the containment penetrations. The design provides motor-operated isolation valves in each SIT discharge line which are administratively controlled to open. During normal operation, the operators remove power to prevent inadvertent closure. Furthermore, these motor-operated valves receive an automatic open signal when a SIAS occurs.

In DCD Tier 2, Section 6.3.2.2.5, "Valves," the applicant stated that all lines from the RCS up to and including the outermost CIVs are designed for full RCS pressure. Furthermore, the applicant described the SIS relief valves, which are required by applicable codes. The applicant stated that the SIT relief valves are sized to protect the tanks against the maximum fill rate of liquid or gas into the SITs. The SIT relief valves discharge directly into the containment. Following the guidance in SRP Section 6.3, the staff determined that additional information would be needed to confirm that the SIT relief valves have adequate capacity to maintain SIT integrity in the event of a maximum fill rate of liquid or gas into the SIT. Therefore, on August 5, 2015, the staff issued an audit plan to address its concern (ML15245A547). The staff reviewed the applicant's calculation regarding the SIT relief valve sizing and documented the audit results in the audit report (ML#########). The staff confirmed that the applicant adequately sized the SIT relief valves and that the values presented in DCD Tier 2, Section 6.3.2.2.5 are accurate.

Based on the review above, the staff concludes that the RCS and SIS piping interface and valve system meets the acceptance criteria in GDC 35 because the design of the RCS and SIS piping interface and valve system supports the SIS function of providing abundant emergency core cooling. The staff also concludes that the applicant's SIS's piping and valve design satisfies GDC 4 requirements because it provides assurance that dynamic effects of events such as flow instabilities and water hammer will not adversely affect the fundamental integrity and capability of the ECCS systems to provide core cooling in the event of accidents.

6.3.4.2.5 Heat Exchangers

The staff noted that there are no heat exchangers associated with the SIS. The containment spray and the shutdown cooling heat exchangers, which can be used to cool IRWST water under different conditions, are evaluated in Sections 6.2.2 and 5.4.7, respectively, of this SER.

6.3.4.2.6 Instrumentation and Control

The staff reviewed the provisions for I&C of the SIS. The following documents the staff's notes regarding I&C. The operators monitor and control the SIS trains from the MCR. The system design makes available process instrumentation to the operators in the MCR to assist in assessing post-LOCA conditions and to assist in ensuring SIS operation is normal. The staff noted that SIS controls are automatic and sequence the necessary operations of SI following actuation by the SIAS. The staff also noted that the design provides for redundancy for initiation of SIS actions, for example, four process variable sensors are used for each critical parameter. Any trip from two out of the four sensors provides the necessary coincidence for SI actuation. Low pressurizer pressure or high containment pressure are the signals that initiate SI. Four independent AC buses supply electrical power for SIS I&C, where EDGs provide a safety-grade back-up source of power.

The staff noted that sufficient I&Cs are available to the reactor operators in the MCR to provide adequate information about the SIS operation for normal and post-LOCA conditions. Coolant flow and temperature, SIT pressure and level, SIP discharge pressure, and valve positions are all provided with sufficient redundancy to satisfy the SF criterion.

The staff's detailed review of the I&C is discussed and documented in Chapter 7, "Instrumentation and Controls," of this SE.

6.3.4.2.7 In-Containment Refueling Water Storage Tank

The staff noted that the IRWST is a part of the IWSS. The staff noted that the IWSS performs water collection, delivery, storage, and heat sink functions inside the containment during normal and refueling operations as well as accident conditions. The seismic Category I IRWST, located at an elevation below 30.5 m (100 ft) in the containment, provides the safety-related source of borated water for SI. The tank is constructed of reinforced concrete with an austenitic stainless steel liner on surfaces that are in direct contact with borated water. The volume of borated water in the IRWST is maintained by TS.

The staff noted that operators monitor the IRWST for its water level, temperature, and boron concentration. If the boron concentration or borated water temperature fall outside of the TS limits, then the operators shall restore the IRWST boron concentration or borated water temperature within limits within 8 hours. Furthermore, if the IRWST borated water volume falls outside of the TS limit, then operators shall restore it within 1 hour. The staff confirmed that these specifications are appropriate. The spent fuel pool cooling and cleanup system maintains IRWST purity and the CVCS provides the means to add makeup to, and adjust the boron concentration of, the IRWST.

Due to the design of the IWSS, RCS break water and containment spray water are first collected in the holdup volume tank (HVT), which is a part of the IWSS. The IRWST and HVT are interconnected by two spillways. When the water level in the HVT gets high enough, the water spills into the IRWST via the spillways. The bottom of the spillway is submerged below the surface of the water in the IRWST and the top of the spillway is below the IRWST ceiling. The staff confirmed that the spillways are seismic Category I and are located at a high elevation to prevent high density debris which has entered the HVT from entering the IRWST. Seismically designed vertical trash racks are provided at the entrance of the HVT to prevent large debris from entering the HVT. The IRWST has four independent sets of strainers, one for each of the four SIPs. The applicant discusses the design features of the IRWST strainers in APR1400-E-N-NR-14001-P, "Design Features to Address GSI-191," Revision 0, and the staff's review is a part of Section 6.2 of this SER.

An emergency core cooling function of the IRWST is to provide sufficient water depth to prevent cavitation of the SIPs during DBA conditions. Water level in the tank varies during accidents.

Sufficient NPSH needs to be available throughout the course of these events. The staff evaluated protection against loss of NPSH above under the "Net Positive Suction Head" heading.

In DCD Tier 2, Section 6.8.4.2, "Available Net Positive Suction Head," the applicant stated that the minimum water volume of the IRWST during normal operation is 2,373.5 m^3, which provides sufficient available NPSH to the Ps and SC pumps or CS pumps. The staff noted that the applicant's TS require the IRWST borated water volume to be greater than or equal to 2,373.5 m^3 and less than or equal to 2,540.6 m^3. The basis of TS 3.5.4 states that the minimum water volume in the IRWST of 2,373.5 m^3 was used in the applicable safety analyses.

The staff reviewed the minimum water level calculation that the applicant presented in APR1400-E-N-NR-14001-P as part of the audit (ML#########). During the audit, the staff confirmed that the minimum water level of the IRWST during a LOCA, which the applicant used in the NPSH available calculation for the SIPs, factored into account the minimum water volume of the IRWST during normal operation as allowed by TS. In addition, the staff noted that the applicant conservatively accounted for water volume lost due to holdup volumes within containment. Finally, the staff confirmed that the minimum IRWST water volume during normal operation as allowed by LCO 3.5.4 is within the bounds of the safety analysis.

6.3.4.3 Testing, Inspection, and Qualification

The staff notes that SIS testing requirements are specified by GDC 37. The staff's review confirms that the design of the SIS permits online testing of individual trains and components to assess their operational status and availability and ensures that periodic testing of the SIS demonstrates that the system operates properly when an accident signal is received.

The staff noted that initial testing of the SIS verifies that the as-designed and as-constructed system functions as credited in the safety analysis. Initial testing is discussed in DCD Tier 2, Section 14.2. The tests applicable to the SIS in DCD Tier 2, Section 14.2 are:

- Section 14.2.12.1.21, "Safety Injection System Test"
- Section 14.2.12.1.22, "Safety Injection Tank Subsystem Test"
- Section 14.2.12.1.23, "Engineered Safety Features – Component Control System Test"
- Section 14.2.12.1.59, "Pre-core Safety Injection Check Valve Test"

In each of the above tests, the acceptance criteria involve whether the system functions as described in the design basis of the respective sections of DCD Tier 2, Sections 6.3 and 7.3.1. The staff reviewed the SIS pre-operational tests and determined that they are sufficient at demonstrating the SSCs will perform satisfactorily while in service because the tests and their associated acceptance criteria were developed from design basis information presented in DCD Tier 2, Section 6.3.

In DCD Tier 2, Section 14.2.12.1.22 and Section 14.2.12.1.59, "Pre-core Safety Injection Check Valve Test," the applicant provided the objectives, prerequisites, test methods, data required, and acceptance criteria for successfully completing these SIS tests. The staff finds these tests to be acceptable and adequate to ensure proper function of the SIT subsystem because the tests and their associated acceptance criteria were developed from design basis information presented in DCD Tier 2, Section 6.3. Based on the staff finding that these tests are acceptable and adequate, the staff finds that the applicant's pre-operational tests meet GDC 37.

The staff noted that each installed SI train is tested to measure SIP developed differential pressure at mini-flow and runout flow through the DVI lines and for SIPs 3 and 4, measure runout flow through the hot leg injection lines. Runout flow testing is conducted with the RCS at atmospheric pressure. These test results are used to confirm SIP performance characteristics over the operating range of the pump and to confirm system resistance characteristics. SIT testing is conducted to demonstrate that the SITs can be depressurized by venting for entry into shutdown cooling. In addition, each SIT is tested to measure changes in SIT pressure and level during SIT blowdown to demonstrate that the as-built resistance coefficients of the FD of each tank and discharge line are within the limits used to perform the safety analyses of Chapter 15. Check valves in the SI lines are leak tested using the local vent and drain lines.

The staff reviewed in-service testing of SI valves and pumps in Section 3.9.6 of this SE. The SIP can be tested at minimum and rated flow using the bypass line of the mini-flow line, periodically. Periodic tests and inspections of the SIS provide reasonable assurance of proper operation in the event of an accident. The complete schedule of SIS tests and inspections is provided in DCD Tier 2, Chapter 16 and reviewed by the staff in Chapter 16 of this SE.

The staff noted that some active components of the system, typically motor operated valves, are located inside the containment building. These components will be exposed to a harsh environment (high temperature steam-air mixture at high pressure) in the event of an RCS break or a steam line break inside containment. Therefore, in accordance with GDC 4, the SIS shall perform its intended function under these accident conditions. The electrical components of the system located inside containment shall be qualified to operate in the expected environment in accordance with GDC 4. The applicant designed the SIS so that components required to maintain a functional status be located outside the containment to eliminate exposure of this equipment to post-LOCA conditions, to the extent practicable. The components located inside containment, such as valves and I&C equipment, are designed to withstand LOCA conditions of temperature, pressure, humidity, chemistry, and radiation for the required extended period described in DCD Tier 2, Section 3.11 and reviewed by the staff in Chapter 3 of this SE. The SIS valves inside containment must be located above the maximum floor flooding level which protects the valve motor operators from submersion following an accident. The staff reviewed the provided information regarding flood levels and component location requirements in Chapter 3 of this SE.

GDC 2 requires that systems important to safety, like the SIS, be designed to withstand the effects of natural phenomena including earthquakes. Furthermore, RG 1.29 assigns seismic Category I designation to the SIS, meaning that the system is designed to remain functional after a safe-shutdown earthquake. The staff confirmed that seismic Category I is specified for the SIS. The system and its components are designed to seismic Category I requirements. The most severe ground motion spectra postulated as design basis information for the SIS, among other SSCs, is set forth in DCD Tier 2, Section 2.5.2, and whether the SIS design will withstand the SSE based on these spectra is evaluated in Chapter 3 of this SE.

In DCD Tier 2, Section 6.3.1.5, "Reliability Design Bases," the applicant stated that the SIPs are located in the auxiliary building as close as practicable to the containment structure. The staff noted that that the auxiliary building is seismic Category I and provides adequate protection of the SIS components from natural and external phenomena. Equipment seismic qualification is addressed in DCD Tier 2, Section 3.10, "Seismic and Dynamic Qualification of Mechanical and

Electrical Equipment." DCD Tier 2, Section 3.3, "Wind and Tornado Loadings," and Section 3.4, "Water Level (Flood) Design," discuss protection against other natural phenomena. These areas are evaluated by the staff in Chapter 3 of this SE.

6.3.4.4 System Reliability

In DCD Tier 2, Section 6.3.1.5, "Reliability Design Bases," the applicant stated that the SI and safe shutdown safety functions of the SIS can be accomplished assuming the failure of a single active component during a short-term mode of operation or assuming a single active or limited leakage passive failure of a component during a long-term post-accident mode of operation. The staff noted that the process variables (e.g. pressurizer pressure and containment pressure) used for the SIAS are derived from four independent pressurizer pressure sensors and four independent containment pressure sensors. Coincidental trip signals from two out of the four sensors for either parameter automatically initiate SI. The four SIAS logic channels for each parameter are independently powered from the same normal and emergency sources that power the associated motive equipment of the associated train. The design of the SIS I&C, including its quality, redundancy, and protection against the effects of a SF, is evaluated in Section 7.3 of this SE.

The applicant conducted a detailed failure mode and effects analysis for the SIS, presented in DCD Tier 2, Table 6.3.2-2, "Safety Injection System Failure Modes and Effects Analysis." GDC 35, requires that the SIS accomplish its safety functions assuming a SF. The staff reviewed the failure mode and effects analysis for the SIS, which includes electrical, mechanical, and I&C failure modes, and finds that no single active failure will affect more than one train of the ECCS since in each case of a mechanical, electrical, or I&C SF, the SIS can successfully perform its mission criteria. The redundancy incorporated into the system design allows the SIS to fulfill its safety function in spite of such failures, as further addressed in Section 15.6.5 of this SER. The detailed failure modes and effects analysis review is performed as part of Chapter 19 of this SE.

The staff reviewed the post-LOCA long term cooling technical report as part of the review of DCD Tier 2, Section 15.6.5.

6.3.4.5 Performance Evaluation

While the design of the SIS is reviewed in this Section of the report, the SIS performance evaluation under accident conditions is presented DCD Tier 2, Chapter 15, "Transient and Accident Analyses," and reviewed in the corresponding chapter of this SE.

The staff confirmed that the APR1400 SIS design meets functional design requirements to satisfy the performance requirements of GDC 17, GDC 27, and GDC 35.

6.3.4.5.1 Inspections, Tests, Analyses, and Acceptance Criteria

The ITAAC associated with the IWSS and the SIS are given in DCD Tier 1, Section 2.4.2 and Section 2.4.3, respectively. DCD Tier 1, Table 2.4.3-4 specifies: (1) the design commitments of the SIS; (2) the inspections, tests, and analyses to be performed; and (3) the acceptance criteria to ensure that the SIS is built as designed. Equivalent information for the IWSS is contained in DCD Tier 1, Table 2.4.2-4. The staff's review of the ITAAC included ensuring that all Tier 1 information is consistent with Tier 2 information, ensuring that Tier 1 figures and diagrams accurately depict the functional arrangement, location, and requirements of the associated systems, ensuring that the reactor systems Tier 1 design descriptions are clear and provide the key performance characteristics and safety-related functions of the systems, and ensuring the provided ITAAC is sufficient and necessary to ensure compliance with 10 CFR 52.47(b)(1).

During the staff's review, the staff noted that in DCD Tier 2, Section 6.3.1.5, "Reliability Design Bases," the applicant claimed adequate physical separation is provided between the redundant piping paths and containment penetrations of the SIS so that the SIS meets its functional requirements even with a SF. The staff determined that the applicant's current proposed Tier 1 ITAAC are not sufficient to ensure that adequate physical separation, to the extent practical, is provided for each division of the SIS to preclude the loss of the safety-related function by common-cause failure from postulated dynamic effects (i.e. missile and pipe break hazards), internal flooding, and fire. Therefore, on June 17, 2016, the staff issued RAI 496--8630, Question 06.03-13 to address this issue (ML16169A366). The applicant responded to RAI 496-8630, Question 06.03-13, dated July 29, 2016, with a proposed revision to the DCD that added an ITAAC item for verifying adequate physical separation between each division of the SIS (ML16211A299), which the staff found to be acceptable because the proposed ITAAC item was derived from and consistent with Tier 2 information and would ultimately assure that each division of the SIS would be adequately protected against the loss of the safety-related function by common-cause failure from postulated dynamic effects, internal flooding, and fire. On March 10, 2017, the applicant submitted to the NRC Revision 1 of the DCD (ML17096A391). The staff confirmed that Revision 1 of the DCD contained the appropriate revisions, which were proposed as part of the response to RAI 496-8630, Question 06.03-13. Therefore, this RAI question is resolved and closed.

Based on the review above, the staff concludes that the ITAAC associated with the SIS are sufficient and necessary in regards to meeting the requirements of 10 CFR 52.47(b)(1) because as reviewed, they provide reasonable assurance that when satisfactorily completed the SIS will operate in accordance with the design and the regulations.

6.3.4.5.2 Technical Specifications

TS relating to the SIS are presented in DCD Tier 2, Chapter 16, Section 3.5. Both the required actions and SR were reviewed together with the completion times allotted for corrective action and surveillance frequencies.

During the staff's review of LCO 3.5.1 and LCO 3.5.4, the staff noted that the applicant proposed SR to verify the boron concentration for the SIT injection water and the IRWST water, respectively. While the SR for boron concentration of each tank reflect the design values of DCD Tier 2, Section 6.3 as well as the safety analysis values provided in DCD Tier 2, Section 15.6.5, the SR do not specify the boron-10 atom percent of said boron concentrations. The staff noted that the APR1400 has provisions for recycling boron throughout the life of the plant as documented in DCD Tier 2, Section 9.3.4. Because recycled boron has the potential for refilling the IRWST and the SIT during refueling outages, the staff determined that the boron-10 atom percent of the IRWST and SIT may deplete during the life of the plant. In accordance with SRP Section 6.3, the staff was unable to verify the adequacy of the existing scope of the SR regarding boron-10 atom percent of the SIT injection water and the IRWST water. Therefore, on August 20, 2015, the staff issued RAI 158-7997, Question 06.03-7 to address this issue (ML15295A497). On January 7, 2016, the applicant provided a response to the NRC indicating that boron recycling operations are not used for the APR1400 design (ML16007A083). However, the applicant provided no DCD markups which indicate that boron recycling for the SIT and IRWST water can and will not be used for the APR1400. The staff considers RAI 158-7997, Question 06.03-7 resolved and closed but also determined that additional information should be added to DCD Tier 2, Section 6.3, to make it clear that SIT and IRWST water never contain recycled boron. Therefore, on June 17, 2016, the staff issued supplemental RAI 496-8630, Question 06.03-10 to address this issue (ML16169A366). In letter dated August 10, 2017, the applicant provided a revised response to supplemental RAI 496-8630, Question 06.03-10 (ML17222A209). In its response, the applicant clarified that, actually, boron recycling is used during normal operations and that the boron-10 atom percent of the SITs and IRWST can in fact be decreased gradually due to irradiation and recycling. Furthermore in its response, the applicant provided markups to the DCD which propose additional technical specification/surveillance requirements for periodically verifying the isotopic concentration of boron-10 in each SIT and in the IRWST every 24 months. The staff finds the applicant's response acceptable because the SIT water and IRWST water will now be required to be verified to have the appropriate boron-10 isotopic concentration. Requiring the verification of boron-10 isotopic concentration in the SITs and IRWST precludes the APR1400 from ever operating with SIT and IRWST boron-10 isotopic concentrations less than what is assumed in the safety analysis. The staff finds the 24 month verification interval also acceptable because the IRWST and SITs do not receive a significant neutron flux during normal plant operation and also because the IRWST water used as inventory for the SITs is only mixed with the reactor coolant during refueling outages. The staff is tracking RAI 496-8630, **Question 06.03-10 as a confirmatory item** pending the incorporation of the proposed markups from the applicant's revised response into the next revision of the DCD.

During the staff's review of LCO 3.5.2, the staff noted that the applicant proposed a SR 3.5.2.8, to verify, by visual inspection, that the IRWST strainers are not restricted by debris and strainers show no evidence of structural distress or abnormal corrosion. However, in the staff's review of LCO 3.5.4, the staff noted that the applicant did not propose a SR equivalent to SR 3.5.2.8, but for the HVT trash racks. The staff concludes that clogging and/or structural/corrosive degradation of the HVT trash racks could cause inoperability of the IRWST. In accordance with SRP Section 6.3, the staff was unable to verify the adequacy of the existing scope of the SRs regarding the HVT trash racks. Therefore, on August 20, 2015, the staff issued RAI 158-7997, Question 06.03-8 to address this issue ML15295A497). On September 17, 2015, the applicant provided a response to RAI 158-7997, Question 06.03-8 (ML15260B338). In its response, the applicant stated that the HVT trash racks will be included in the SR 3.5.2.8. The applicant also presented in its response to the staff how the revision will look in the next DCD revision. Based upon the staff's review of the applicant's TS SR 3.5.2.8 draft revision, the staff finds that the applicant's TS regarding the IRWST operability is adequate because visual inspection that the HVT trash racks are not restricted by debris ensures that the IRWST will be able to provide injection water for the ECCS during an accident. Furthermore, the staff concludes that the adequacy of the TS meets the requirements of 10 CFR 50.36(c)(3) because visual inspection that the IRWST sump, including the HVT trash racks, is not restricted by debris precludes facility operation outside safety limits (e.g. the IRWST will be operable when it is called upon to function during an accident). The staff concludes that the applicant's proposed revision is acceptable for the reasons described above. On March 10, 2017, the applicant submitted to the NRC Revision 1 of the DCD (ML17096A391). The staff confirmed that Revision 1 of the DCD contained the appropriate revisions, which were proposed as part of the response to RAI 158-7997, Question 06.03-8. Therefore, this RAI question is resolved and closed.

As part of the Chapter 6 presentation to the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee, an ACRS member questioned whether the IRWST vacuum protection swing panels need to be included in the design's technical specifications in order to ensure operability of the IRWST and downstream pumps (safety injection, containment spray, shutdown cooling pumps) during an accident. The ACRS questioned what would occur in the event that the IRWST swing panels failed to open in the event of vacuum conditions in the IRWST (such as an SI event where containment is not pressurized). Specifically, they asked whether the IRWST structural integrity or operability of any downstream pump could be challenged due to a lack of net positive suction head (NPSH), given the analysis assumes a minimum containment pressure. The staff reviewed Chapter 6 of the DCD and could not garner enough information to determine whether or not the vacuum protection swing panels are required to be operable in order to ensure operability of the IRWST itself and the downstream pumps that the IRWST feeds. Therefore, on May 22, 2017, the staff issued RAI 547-8819, Question 06.02.02-47 to address this issue (ML17142A457). On July 19, 2017, the applicant provided a response to RAI 547-8819, Question 06.02.02-47 (ML17200D082). In its response, the applicant provided a conservative NPSH calculation showing the amount of NPSH margin remaining for the safety injection and containment spray pumps when the IRWST is under vacuum conditions due to failed vacuum protection swing panels. The amount of margin remaining for the SI pumps is approximately 9ft-water and for the CS pumps is approximately 10ft-water. The NPSH margin remaining for this scenario far exceeds the NPSH margin calculated for the worst case accident scenario via the methodology in RG 1.82, as discussed in the section above titled “Net Positive Suction Head.” Therefore, based on the above, the staff finds the applicant’s response to the RAI question acceptable and the staff finds that the IRWST vacuum protection swing panels do not need to remain operable during an accident in order to ensure the operability of the IRWST itself and the downstream pumps that the IRWST feeds. Thus, the IRWST vacuum protection swing panels are not required to be in the APR1400’s technical specifications. RAI 547-8819, Question 06.02.02-47 is resolved and closed.

A further, detailed review of the APR1400 SIS TS are presented in Chapter 16 of this SE.

Based on the review above, the staff concludes that the TSs are acceptable.

6.3.5 Combined License Information Items

DCD Tier 2, Section 6.3.7 provides a list of COL information items related to the SIS that can also be found in DCD Tier 2, Table 1.8-2.

1. COL 6.3(1) – The COL applicant is to prepare operational procedure and maintenance programs as related to leak detection and contamination control.

2. COL 6.3(2) – The COL applicant is to maintain complete documentation of system design, construction, design modification, field changes, and operations.

The staff reviewed the COL information item list provided by the applicant in DCD Tier 2, Section 6.3. The staff finds the applicant’s COL information item list to be complete. The list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2, for SIS consideration.

6.3.6 Conclusion

The staff evaluated the proposed design of the SIS for the APR1400 as described primarily in DCD Tier 2, Section 6.3, and partially in DCD Tier 2, Section 6.8, and reviewed proposed design requirements that pertain to the SIS as specified in DCD Tier 1, Sections 2.4.2 and 2.4.3. Other DCD Tier 2 information was also evaluated with respect to SIS considerations, such as the proposed TS requirements provided in Chapter 16, and initial test program considerations described in DCD Tier 2, Section 14.2. The staff’s evaluation was performed in accordance with the guidance provided in SRP Section 6.3 and included review of piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and function design considerations, such as piping and component arrangements and locations; actuation and control features; normal and post-accident configurations; electrical power supplies; and leakage detection, isolation, and atmospheric release considerations. The staff evaluated the design criteria and design bases that were proposed by the applicant for the SIS, and the manner in which the design conforms to these criteria and bases and satisfies applicable NRC requirements such as the GDC and TMI action plan items.

The SIS is actuated to mitigate the consequences of an accident upon receipt of a SIAS. Each of the four SIS trains includes a SIP, SIT, and associated piping and valves and controls necessary to provide makeup water from the IRWST to cool the core. Each train is powered from separate safety-related emergency buses that receive emergency power from their respective EDG sets. The applicant designed the SIS with regards to completing its safety function following a postulated, most-limiting SF.

Based on the review above, the staff concludes that the design of the SIS is acceptable and meets the applicable regulations specified in the Section 6.3 of the Standard Review Plan.

The staff also finds the TSs and ITAAC that are proposed for the SIS acceptable. The ITAAC ensures that critical SIS design details are satisfied by the as-built plant. Therefore, the staff finds that the proposed ITAAC are necessary and sufficient to ensure that the SIS will be built in accordance with the design specifications, thereby satisfying the requirements specified in 10 CFR 52.47(b)(1). The staff also finds that the TSs are technically sufficient to ensure that the plant will be operated in accordance with the design specifications, and that the SIS will remain capable of adequately cooling the core during design basis accidents.

6.4 Control Room Habitability System

6.4.1 Introduction

The APR1400 Habitability Systems are designed to allow control room operators to remain in the control room and take actions to operate the plant safely under normal conditions, and maintain it in a safe condition under abnormal conditions, including a LOCA, as required by GDC 19, “Control Room.”

Habitability Systems provide the necessary support for Control Room Envelope (CRE) which includes the Technical Support Center (TSC). The Control Room HVAC System (CRHS) consists of the Control Room Emergency Makeup Air Cleaning System (CREACS) and the Control Room Supply and Return System (CRSRS). The review of the CRHS as a ventilation system is discussed in Section 9.4.1, “Control Room Area Ventilation System,” of this SE. The CREACS and iodine removal efficiencies of the CREACS are discussed in Section 6.5.1 of this report. Review of ITAAC requirements is discussed in Section 9.4.1 of this SE.

6.4.2 Summary of Application

DCD Tier 1: DCD Tier 1, Section 2.7.3, “HVAC Systems,” contains Section 2.7.3.1, “Control Room HVAC System.” The CRHS supplies air to the CRE. Each of the seismic Category I and Class 1E components and instruments for the CRHS are shown in DCD Tier 1, Tables 2.7.3-1, “Control Room HVAC System Components List,” and Section 2.7.3-2, “Control Room HVAC System Instruments List.”

DCD Tier 2: The MCR habitability systems are designed to protect the plant operators in the CRE from the effects of accidental releases of radioactive and toxic gases. The CRE includes the control room and TSC and allows control room operators to remain in the CRE to operate the plant safely under normal conditions and to maintain the plant in a safe state under accident conditions.

ITAAC: The CRHS ITAAC are shown in DCD Tier 1, Table 2.7.3.1-3, "Control Room HVAC System ITAAC." DCD Tier 2, Table 1.8-2, COL Item 14.3-2 states that ITAAC for emergency planning will be provided by the COL applicant.

TSs: The TSs for the CRHS are in DCD Tier 2, Chapter 16, Sections 3.7.11, "Control Room HVAC System (CRHS)"; Chapter 16, 5.5.11, "Ventilation Filter Testing program (VFTP)," and 5.5.18, "Control Room Envelope (CRE) Habitability Program."

6.4.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG–0800, Section 6.4, "Control Room Habitability System," and are summarized below. Review interfaces with other NUREG–0800 Sections can also be found in NUREG–0800, Section 6.4.

- GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents.

- GDC 5, as it relates to ensuring that sharing among nuclear power units of SSCs important to safety will not significantly impair the ability to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cool-down of the remaining unit(s).

- GDC 19, as it relates to the control room being provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions and providing radiation protection adequate to permit access to and occupancy of the control room under accident conditions.

- The regulations in 10 CFR 50.34(f)(2)(xxviii), "Contents of applications; technical information," as it relates to evaluations of potential pathways for radioactivity and radiation and associated design provisions to preclude certain control room habitability problems.

- The regulations in 10 CFR 52.47(b)(1), "Contents of applications; technical information," which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations.

Acceptance criteria adequate to meet the above requirements are contained in the following documents:

- RG 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors,” and RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors,” provide acceptable guidance for meeting control room habitability requirements.

- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

- RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered Safety Feature Atmosphere Cleanup Systems in Light Water Cooled Nuclear Power Plants.”

- ASME Code AG 1, “Code on Nuclear Air and Gas Treatment.”

- RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release.”

- TMI Action Plan Item III.D.3.4 (NUREG–0737), regarding protection against the effects of toxic substance releases, either onsite or offsite.

- GSI, Item B 36, “Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems.”

- GSI, Item B 66, “Control Room Infiltration Measurements.”

- GSI 83, “Control Room Habitability (Revision 3).”

The NUREG–0800 acceptance criteria are also based on conformance to NUREG–0696, “Criteria for Emergency Operations Facilities for Nuclear Power Reactors,” for guidance in establishing the habitability criteria for the TSC portion of the CRE.

6.4.4 Technical Evaluation

Review of the Habitability Systems in the DCD was performed in accordance with SRP Section 6.4, Section III, “Review Procedures.” The Habitability Systems are composed of the CRE along with the CRHS and its associated CREACS.

6.4.4.1 Control Room Envelope

The CRE was reviewed to determine if it included those facilities discussed in the acceptance criteria of the Control Room Emergency Zone in NUREG–0800, SRP Section 6.4. The CRE includes the MCR, atomic energy bureau room, meeting room, service room, instrument maintenance shop, corridors, HVAC areas, HVAC equipment rooms, Technical Support Center (TSC) and TSC record storage room, computer room, computer room office, computer room package air conditioning unit (PACU) room, kitchen and dining room, locker room, shower room, and toilets. (Refer to DCD Tier 2, Section 6.4.2.1, “Definition of Control Room Envelope,” DCD Tier 2, Figure 6.4-1, “Control Room Envelope Flow Diagram,” and DCD Tier 2, Figure 9.4.1-1, “Control Room HVAC System Flow Diagram”). As stated in DCD Tier 2, Sections 6.4.1, “Design Bases,” and 6.4.5, “Testing and Inspection,” the CRE design conforms to RG 1.196. As stated in DCD Tier 2, Sections 6.4.5, the CRE periodic testing conforms to RG 1.197.

NUREG–0800, SRP Section 6.4, Acceptance Criteria 1 recommends that the control room emergency zone shall include the following:

1. I&Cs necessary for a safe shutdown of the plant, i.e., the control room, including the critical document reference file;

2. Computer room, if it is used as an integral part of the emergency response plan;

3. Shift supervisor's office; and

4. Operator washroom and the kitchen.

5. The control room emergency zone should conform to the guidelines of RG 1.196, May 2003, "Control Room Habitability at Light Water Nuclear Power Reactors," and RG 1.197, May 2003, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."

Although the APR1400 CRE does not have a dedicated shift supervisor's office, it has enough rooms to host shift supervisor handling office activities. As such the staff believes the CRE meets the intent of NUREG–0800, SRP Section 6.4, Acceptance Criteria 1A through 1E, and is therefore acceptable to the staff. Additionally, the CRE and the Habitability System were reviewed for the acceptance criteria of NUREG–0800, SRP Section 6.4, by reviewing the results of other SRP reviews. The CRHS was reviewed as a ventilation system using SRP Section 9.4.1, and the filters in CREACS were reviewed using SRP Section 6.5.1. As discussed above, the staff finds that the APR1400 design meets Acceptance Criterion 1 in NUREG–0800, SRP Section 6.4, and is therefore acceptable.

6.4.4.2 Ventilation System Criteria

6.4.4.2.1 Isolation Dampers

The design and construction of isolation dampers are in accordance with ASME N509, "Nuclear Power Plant Air-Cleaning Units and Components," and ASME AG-1, "Code on Nuclear Air and Gas Treatment." All isolation dampers used to isolate the CRE from the outside are leaktight and the leakage class of these dampers is zero leakage in accordance with Appendix DA-I, of ASME AG-1, "Seat and Frame Leakage." The isolation damper components are safety Class 3 and seismic Category I.

Based on the above, the staff finds that the APR1400 design meets Acceptance Criterion 2A in NUREG–0800, SRP Section 6.4, and is therefore acceptable.

6.4.4.2.2 Single Failure Criterion

The staff reviewed the CRHS system for SF vulnerability in conjunction with Section 9.4.1 of this SE. The staff found that all safety-related portions of the Control Room HVAC System are provided with redundancy, and all safety-related HVAC equipment are powered by an independent Class 1E source.

The staff does not see that any single active component failure could result in loss of the system's safety functions.

Based on the above, the staff finds that the APR1400 design meets Acceptance Criterion 2B in NUREG–0800, SRP Section 6.4, and is therefore acceptable.

6.4.4.2.3 Occupancy Limitations

The habitability systems are designed to support a minimum of five people in the CRE during all normal conditions and at least 30 days of accident-operating conditions.

The CRHS has an isolation mode that isolates the Division 2 and 3 outside air intakes and recirculates the CRE atmosphere. NUREG–0800, SRP Section 6.4, Section III, states that, "the air inside a 2830 m^3 (100,000 cubic feet (ft^3)) control room would support five persons for at least six days. Thus, CO_2 buildup in an isolated emergency zone is not normally considered a limiting problem." The CRE volume is 5,663 m^3 (200,000 ft^3), which is well above the criteria in NUREG–0800, Section 6.4, Section III used to evaluate Control Room Personnel capacity and the potential for CO_2, buildup in the control room operators. Therefore, because the CRE volume is greater than the CRE volume needed to support Control Room Personnel, the staff concludes that the CRE design is sufficient to remove concern for the buildup of CO_2 for five persons for at least 6 days.

Based on the above, the staff finds the APR1400 design meets the Acceptance Criterion 2 in NUREG–0800, SRP Section 6.4, and is therefore acceptable.

6.4.4.3 Pressurization Systems

During normal and emergency modes of operation, the Control Room HVAC System maintains the CRE at a minimum 3.175 mm (0.125 in) water gauge of positive pressure with respect to the surrounding areas. Since CRE is pressurized all the time, system pressurization rate is not an issue.

Therefore, the staff finds the APR1400 design meets the Acceptance Criterion 3 in NUREG–0800, SRP Section 6.4, and is therefore acceptable.

6.4.4.4 Atmosphere Filtration Systems

The atmospheric filtrations systems are described in other Sections of the Tier 2 DCD and are also reviewed under other Sections of NUREG–0800. Specifically, Iodine Filtration, described in DCD Tier 2, Section 6.5.1, "ESF Filter Systems," is reviewed against NUREG–0800, SRP Section 6.5.1, and the Control Room HVAC System, described in DCD Tier 2, Section 9.4.1 is reviewed against NUREG–0800 SRP Section 9.4.1. The staff's evaluation of these Tier 2 DCD Sections is contained in Sections 6.5.1 and 9.4.1 of this SE. The staff determined that no additional information beyond the documentation contained in Sections 6.5.1 and 9.4.1 of this SER is needed. As discussed in Sections 6.5.1 and 9.4.1 of this SE, the staff finds that the Habitability Systems conform to the guidelines of RG 1.52, ASME Code AG--1, and therefore meet Acceptance Criterion 4 in NUREG–0800, SRP Section 6.4, and is therefore acceptable.

6.4.4.5 Relative Location of Source and Control Room

6.4.4.5.1 Radiation Sources

The staff reviewed the APR1400 design against NUREG–0800, SRP Section 6.4, Acceptance Criterion 5, "Relative Location of Source and Control Room, Radiation Sources," as discussed below. In addition, review of the design against TMI Action Plan, Item III.D.3.4 and 10 CFR 50.34(f)(2)(xxviii) is discussed in this Section.

NUREG–0800, Section 6.4, Acceptance Criterion 5A, "Radiation Sources," states that the control room ventilation inlets should be separated from the major potential release points by at least 31 meters (100 ft) laterally and by 16 meters (50 ft) vertically, or be based on dose analyses. As it applies to radiation sources, 10 CFR 50.34(f)(2)(xxviii) requires that the applicant evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source-term release, and to make the necessary design provisions to preclude such problems. This requirement is identified as TMI Action Plan Item III.D.3.4.

Upon receipt of an engineered safety feature actuation system – safety injection actuation signal (ESFAS-SIAS) or if there is a manual actuation or high radioactivity is detected in the CRE outside air supply duct which results in an engineered safety feature actuation system – control room emergency ventilation actuation signal (ESFAS-CREVAS), the CRE normal air supply is automatically isolated, and the GDC 19 habitability requirements are met by the control room emergency makeup air cleaning system (CREACS). The CREACS provides emergency ventilation, filtration and pressurization for the CRE through two outside air intakes on the roof of the auxiliary building, one per division. The system automatically selects the intake with the lower radiation levels, and isolates the other intake.

In RAI 59-7970, question 06.04-1, the staff requested additional information on how the control room ventilation systems intake monitor ESFAS-CREVAS setpoints were determined so that the control room ventilation system operation is as designed and as modeled in the DCD Tier 2, Chapter 15 analyses. In its response dated August 31, 2015 (ML15243A565), the applicant provided detailed information to show that for each DBA evaluated in DCD Chapter 15, the total time for the airborne activity concentration to reach the setpoint and CREACS to start filtration operation is less than the time delay of 5 minutes assumed in the DBA radiological consequence analyses. The staff reviewed the applicant's calculation of the total time for isolation and initiation of filtration, and finds that it used conservative assumptions for the calculation of time for the radioactivity concentration in the intake to reach the ESFAS-CREVAS signal setpoint, and that it included reasonable estimates of time for startup of the emergency diesel generators, signal processing, damper movement and air cleaning unit fan startup. Therefore, the staff has determined that RAI 59-7970, Question 06-04.1 is resolved and closed.

The staff reviewed the two CREACS divisional air intake locations relative to radioactivity release points during an accident. The staff found that the intakes are physically separated and are located far enough away from the containment vent, fuel handling area vent, main steam safety valve vents and atmospheric dump valve vents to prevent direct pathways of radioactivity. The short-term (accident) radiological atmospheric dispersion factors (χ/Q s), for the control room are given in DCD Tier 2, Tables 2.3-2, "Onsite χ /Q for Reactor Containment Building Release to MCR North and South Intakes and MCR Roof Centerline," through 2.3-12, "Onsite χ/Q for Fuel Handling Area Exhaust Release to MCR North and South Intakes.

The analysis of the radiological consequences of DBAs in the control room is discussed in DCD Tier 2, Chapter 15. DCD Tier 2, Section 6.4.2.3 discusses the details of the analysis assumptions on control room envelope unfiltered inleakage, which are used as input to the DBA radiological consequence analysis. Analysis assumptions on the control room HVAC system operation, including the CREVAS operation, are taken from information in DCD Tier 2, Section 6.4. The impact on control room habitability is evaluated for each of the DBAs analyzed in DCD Tier 2, Chapter 15, and the dose results are listed in DCD Tier 2, Table 6.4-2, "MCR and TSC Does from Design Basis Accidents." In a clarification call with the applicant on July 15, 2015, the staff noted that the control room dose values reported in DCD Tier 2, Table 6.4-2 are not consistent with the values reported in the DCD Tier 2, Chapter 15 DBA-specific discussions. The applicant agreed that the values were inconsistent and stated that they would update Table 6.4-2, with the correct values in the next revision of the DCD. The staff confirmed that this issue was corrected in DCD Revision 1.

The limiting DBA was determined to be the loss-of-coolant accident (LOCA), with a resulting control room total effective dose equivalent (TEDE) of 0.044 Sv (4.4 rem). DCD Tier 2, Table 6.4-1, "The Accident Radiation Source Description and Radiation Shielding Design for MCR and TSC," provides information on the accident source and radiation shielding design for the control room dose analyses, and the DBA control room dose calculations in DCD Tier 2, Chapter 15 include the contribution from direct radiation shine from the radioactive material in the MCR charcoal filters, the containment atmosphere, and the radioactive plume outside the facility. The staff has evaluated the control room air intakes and the assumed unfiltered inleakage intakes relative to the distance to the DBA release locations. The control room dose analyses in DCD Tier 2, Chapter 15 used short-term atmospheric dispersion factors for the control room that appropriately accounted for the relative locations of DBA releases to the control room intake and assumed inleakage location. More detail on the staff's review of the control room atmospheric dispersion factors can be found in Section 2.3.4 of this SER. The staff has determined that the APR1400 design conforms to RG 1.183, for control room habitability analyses because the release and control room receptor locations for the APR1400 plant have been specifically modeled in the analyses. More detail on the staff's review of the control room DBA radiological consequences analyses is given in Section 15.0.3 of this SER.

Based on the above, the staff finds that the APR1400 design is satisfactory and complies with 10 CFR 50.34(f)(2)(xxviii) as it applies to radiation sources, and therefore meets Criterion 5A in NUREG–0800, SRP Section 6.4.

6.4.4.5.2 Toxic Sources

The staff reviewed the APR1400 design against NUREG–0800, SRP Section 6.4, Acceptance Criterion 5B, "Toxic Gases." This criterion states that the minimum distance between the toxic gas source and the control room will depend on the amount and the type of the gas in question, and other site-specific parameters. NUREG–0800, SRP Section 6.4, Acceptance Criterion 5B is met by demonstrating conformance with the guidance of RG 1.78, as described below.

As it relates to toxic gases, TMI Action Plan, Item III.D.3.4 recommends that control room operators be adequately protected against the effects of the accidental release of toxic and radioactive gases such that the nuclear power plant can be safely operated or shut down under DBA conditions.

As stated in DCD, Tier 2, Sections 2.2, the COL applicant is to provide site-specific information on nearby industrial, transportation, and military facilities as required in NRC RG 1.206. The potential sources of toxic or otherwise potentially hazardous gases from nearby industrial, transportation, and military facilities will be addressed by the COL applicant.

As stated in DCD, Tier 2, Section 6.4.5, "Testing and Inspection," the COL applicant is to provide the automatic and manual operating procedures for the control room HVAC System, which are required in the event of postulated toxic gas release. The capability of the Habitability Systems to cope with toxic or hazardous gases will be addressed by the COL applicant.

The staff notes that the applicant committed to the provisions for donning self-contained breathing apparatus (SCBA) with 6-hour bottled air supplies. RG 1.78, Section 4.3, states that "adequate air capacity for the breathing apparatus (at least 6 hours) should be readily available onsite to ensure that sufficient time is available to transport additional bottled air from offsite locations." The staff determines that the applicant's commitment of 6-hour bottled air supplies is satisfactory.

6.4.4.6 Radiation Hazards

The staff reviewed the APR1400 design against NUREG–0800, SRP Section 6.4, Acceptance Criterion 6, "Radiation Hazards." NUREG–0800, SRP Section 6.4, Acceptance Criterion 6B states that applicants for DCs should meet the requirements of GDC 19. Compliance with GDC 19 is discussed below. Radiological protection of the TSC, which is included in the CRE, is also addressed in this section. Control room operators, as well as TSC occupants, are protected from radiation sources by a combination of shielding and distance.

GDC 19, requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. It also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions, without personnel receiving radiation exposures in excess of 5 rem (50 mSv) TEDE for the duration of the accident.

6.4.4.6.1 Radiation Shielding

The MCR and TSC are protected from radioactivity inside containment by over 0.91 m (3 ft) of concrete. The operators inside the MCR and TSC are also protected from the shine from the radioactive cloud that could pass over the auxiliary building by the concrete walls and the roof. The roof slab above the CRE is 0.46 m (1.5 ft) thick. The MCR and TSC are shielded from the shine from radioactive material accumulating on the CREACS filters by the floor between the CREACS filters and the MCR/TSC being 0.46 m (1.5 ft) thick as specified in DCD Tier 2, Table 6.4-1. Radiation streaming from the main steam valves is also shielded by 1.22 m (4 ft) of concrete surrounding main steam valve rooms and by another intervening concrete shield wall, which is 0.91 m (3 ft) thick. In the DBA radiological consequences analyses discussed in Chapter 15 of the DCD, the shielding around the MCR and TSC is modeled in order to estimate doses in the MCR and TSC from containment shine and direct radiation from the release cloud external to the building. The dose from containment shine and the external cloud are added to other dose pathways to give a total dose in the MCR and TSC that meets the dose criterion in GDC 19. Accordingly, this design addresses the radiation shielding concerns expressed in NUREG–0800, SRP Section 6.4.

Since the CRE is well-shielded with its enclosure inside the auxiliary building, the principal sources that affect operator dose in the control room are:

1. the radiation that bypasses the filter, because of filter inefficiency in the CREACS supply air

2. the unfiltered inleakage from all other sources

6.4.4.6.2 CRE Unfiltered Inleakage

The design value for CRE unfiltered inleakage in the pressurization mode is 170 cubic meters per hour (cmh) (100 cubic feet per minute (cfm)). This value includes 17 cmh (10 cfm) for ingress/egress through the control room envelope vestibule door. Although maintaining the

CRE at a positive pressure with respect to adjacent spaces helps minimize unfiltered inleakage, it does not provide assurance that there will be no sources of unfiltered inleakage other than from ingress and egress. The 170 cmh (100 cfm) value does not include the measurement uncertainty for CRE Habitability Testing. However, the measurement uncertainties only need to be added to measured inleakage values when they are above 170 cmh (100 cfm) as stated in RG 1.197, Section 1.4, Test Results and Uncertainty.

TS SR 3.7.11.4 requires performance of CRE unfiltered air inleakage testing in accordance with the CRE Habitability Program in Section 5.5.18 and RG 1.197. Therefore, CRE Habitability Testing (tracer gas testing) is needed to periodically measure the actual CRE unfiltered inleakage.

The staff has determined acceptable the 170 cmh (100 cfm) unfiltered CRE inleakage assumption, which includes 17 cmh (10 cfm) for ingress/egress, because it is used as an input assumption in the radiological evaluations in DCD Tier 2, Chapter 15 which show compliance with GDC 19.

NUREG–0933, “Main Report with Supplements 1-34,” GSI Item B 66, addresses the magnitude of the control room air infiltration rate. RG 1.197 provides methods acceptable to the staff for determining air infiltration, and is referenced in Chapter 16, TS Section 5.5.18 on the Control Room Envelope Habitability Program. Therefore, the staff considers GSI Item B-66 satisfied.

Based on the above discussions on “Radiation Shielding” and “CRE Unfiltered Leakage.” The staff determines NUREG–0800, Section 6.4, Acceptance Criterion 6 is met as it pertains to shielding for the APR1400 because the applicant has demonstrated protection from radiation hazards via distance and shielding using the above assumptions in its evaluation of the radiological consequences of DBAs in the control room, as discussed below.

6.4.4.6.3 Input Parameters to the Radiological Dose Analysis

The values presented in DCD Tier 2, Section 6.4 and Chapter 15 that pertain to the modeling of the control room HVAC system response and unfiltered inleakage were reviewed by the staff. The staff compared the input to the Chapter 15 dose analysis to the characteristics of the design and finds that the values of CRE volume, carbon filter efficiency, HEPA filter efficiency, outside recirculation, and total air flow, roof slab thickness, and the MCR charcoal filter shine to the CRE below were satisfactorily modeled in the dose analysis. The staff found that CRE post-accident filtration model properly considers the outside intake, recirculation, total and unfiltered inleakage, and total air flow in cfm for the purposes of the MCR dose analysis. More detail on the staff’s review of the control room DBA radiological consequences analyses is given in Section 15.0.3 of this SER.

6.4.4.6.4 Radiation Protection

As listed in DCD Tier 2, Table 15.0 10, “Results of Radiological Consequences of APR1400 Design Basis Accidents,” the CRE and control room HVAC system design limits the control room dose for all DBAs to less than the 0.05 Sv (5 rem) TEDE limit specified in GDC 19, and the NUREG–0800, SRP Section 6.4 dose acceptance criteria. This adequately protects control room envelope occupants during postulated accident conditions, and is therefore acceptable to the staff. Because the APR1400 design includes the TSC entirely within the CRE, the radiological consequences of DBAs in the TSC is effectively the same as the MCR dose and is also shown by the DCD Chapter 15 DBA radiological consequence analyses to result in doses less than 0.05 Sv (5 rem) TEDE. The staff’s review of the DBA radiological consequences analyses, including the control room and technical support center radiological habitability, is discussed in Section 15.0.3 of this SER.

6.4.4.6.5 TSC Location

NUREG–0696, Section 2.2, “Location,” provides guidance on TSC location and states that the walking time from the TSC to the control room shall not exceed 2 minutes. Also, provisions should be made to consider the effects of direct radiation and airborne radioactivity from implant sources on personnel traveling between the two facilities.

The staff reviewed the CRE TSC in accordance with NUREG–0696 guidance. DCD Tier 2, Chapter 13.3, “Emergency Planning,” states the TSC is provided in the APR1400 standard design, and is located in the CRE. The walking time from the TSC to the MCR does not exceed 2 minutes; therefore, the staff finds that the APR1400 proposed TSC location in the CRE will comply with NUREG–0696.

6.4.4.6.6 TSC Size

NUREG–0696, Section 2.4, “Size,” provides guidance on TSC size and, in part, states that “the TSC working space shall be sized for a minimum of 25 persons, including 20 persons designated by the licensee and five NRC personnel. This minimum size should be increased if the maximum staffing level specified by the licensee’s emergency plan exceeds 20 persons.” The TSC is sized to provide working space, without crowding, for the personnel assigned to the TSC at the maximum level of occupancy.

DCD Tier 2, Chapter 13.3, states the TSC is sized for a minimum of 25 persons, including five NRC persons and it provides the necessary space to maintain and repair TSC equipment. The staff finds that the size of the TSC is sufficient for storage of plant records and historical data; therefore, the APR1400 proposed TSC size is acceptable.

6.4.4.6.7 TSC Structure

NUREG–0696, Section 2.5, “Structure,” provides guidance on the TSC structure that states that the TSC complex should be able to withstand the most adverse conditions reasonably expected during the design life of the plant.

Since the proposed TSC shares the same building as the control room, it is protected from earthquakes, high winds and floods in the same manner as the CRE, and is protected from dynamic effects in the same manner; therefore, the staff finds that the design of the TSC conforms to the guidance of NUREG–0696, Section 2.5 and NUREG–0800, Sections 6.4 and 9.4.1 and, therefore, complies with the requirements of GDC 2 and GDC 4.

6.4.4.6.8 TSC Habitability

NUREG–0696, Section 2.6, “Habitability,” provides guidance on TSC habitability, stating that the TSC should have the same radiological habitability as the control room under accident conditions and that TSC personnel should be protected from radiological hazards, including direct radiation and airborne radioactivity from in-plant sources under accident conditions, to the same degree as control room personnel. NUREG–0696, Section 2.6 also states that applicable criteria are specified in GDC 19 and NUREG–0800, Section 6.4.

Regarding the TSC ventilation system, NUREG–0696, Section 2.6 guidance states that the TSC ventilation system should function in a manner comparable to the control room ventilation system and that a TSC ventilation system that includes HEPA and charcoal filters is needed, as a minimum.

Since the proposed TSC is incorporated into the CRE, it shares the same ventilation systems and is subject to the same radiological protection as the MCR, thus the above conclusions with respect to the radiological consequences in the control room also apply to the TSC, and the TSC design complies with NUREG–0800, SRP Section 6.4 and satisfies the dose criterion in GDC 19, as they apply to the proposed TSC location. Therefore, the TSC meets the guidance of NUREG–0696, Section 2.6.

Based on the above discussions, the staff concludes the TSC area of the CRE complies with the requirements GDC 2, GDC 4, and GDC 19. Additionally, the APR1400 design conforms to the guidelines of NUREG–0696, SRP Sections 2.4, 2.5, and 2.6 as they apply to TSC ventilation and habitability.

6.4.4.7 Toxic Gas Hazards

As discussed in "Toxic Sources," under 6.4.4.5.2 of this SER, issues on toxic gas hazards will be addressed by the COL applicant.

6.4.4.8 ITAAC

DCD Tier 2, Table 1.8-2, COL Item 14.3-2 states that ITAAC for emergency planning will be provided by the COL applicant. All other related ITAACs have been reviewed and documented in Section 9.4.1 of this SER.

6.4.4.9 Technical Specifications

TS and SRs for CRHS are addressed in DCD Tier 2, Section 16, TS 3.7.11, "Control Room HVAC System," B 3.7.11, "Control Room HVAC System," 5.5.11, "Ventilation Filter Testing Program," and 5.5.18, "Control Room Envelope Habitability Program."

The CRHS consists of CREACS and CRSRS. Both CREACS and CRSRS have two divisions. TS 3.7.11 requires two CRHS divisions be operable. TS B 3.7, LCO, states that each CREACS division is considered operable when individual components, including fans, are operable. RAI 8361 was issued to request the applicant to better define the operability of CREACS related to fans, in particular, the operability status when one of two fans in either CREACS division is not functioning.

TS B 3.7.11, Background, states that "continuous operation of each ACU for at least 10 hours per month with the heaters on reduces moisture buildup on the HEPA filters and absorbers." This statement does not match the 15 minutes testing period defined in SR 3.7.11.1. Also, "absorbers" in TS B 3.7.11, Background, appears to be a typo for "adsorbers." On December 12, 2015, the staff issued RAI 304-8361, Question 06.04-2, to address these matters.

On December 10, 2015, the applicant provided a response to RAI304- 8361, Question No. 06.04-2 (ML15344A195), stating:

> 1. *Each control room emergency makeup air cleaning system (CREACS) division has two 100 percent capacity fans and two 100 percent capacity*

electric heating coils. Each CREACS division is considered operable when one of two fans and one of two electric heating coils are operable. TS Bases B 3.7.11 will be revised to state that each CREACS division is considered operable when one of two fans and one of two electric heating coils are operable.

2. *APR1400 CREACS air cleaning units (ACUs) are designed and tested in accordance with RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants." RG 1.52, Section C.6.a states that each ESF atmosphere cleanup train shall be operated continuously for at least 15 minutes each month, with the heaters on (if so equipped), to justify the operability of the system and all of its components." As a result, APR1400 SR 3.7.11.1 requires operation of each CREACS division for ≥ 15 minutes with heaters operating in accordance with RG 1.52 although CE STS SR 3.7.11.1 requires operation of each CREACS train for "≥ 10 continuous hours with heaters operating or (for systems without heaters) ≥15 minutes."*

   TS Bases B 3.7.11 will be revised to modify the mismatched ACU operation time of "10 hours per month" to "15 minutes per month" and modify the typo, "absorbers" to "adsorbers."

Since each fan in each CREACS division has 100 percent capacity, and that the 15 minutes per month ESF ACU testing period, with heaters on, is specified in RG 1.52, the staff considers the changes of TS Bases 3.7.11 are acceptable. The staff considers RAI 304-8361, Question 06.04-2 resolved. RAI 304-8361 is being tracked as a **confirmatory item** pending incorporation of this revision into the next revision of the DCD.

6.4.5 Combined License Information Items

Table 6.4-1 provides a list of habitability systems related COL information item numbers and descriptions from DCD Tier 2, Table 1.8-2:

Table 6.4-1 – APR1400 Combined License Information Items

Item No.	**Description**	**DCD Tier 2 Section**
6.4(1)	The COL applicant is to provide automatic and manual operating procedures for the control room HVAC system, which are required in the event of a postulated toxic gas release.	6.4.3

Item No.	Description	DCD Tier 2 Section
6.4(2)	The COL applicant is to provide the details of specific toxic chemicals of mobile and stationary sources and evaluate the MCR habitability based on the recommendations in NRC RG 1.78 to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19.	6.4.4.2
6.4(3)	The COL applicant is to identify and develop toxic gas detection requirements to protect the operators and provide reasonable assurance of the MCR habitability. The number, locations, sensitivity, range, type, and design of the toxic gas detectors are to be developed by the COL applicant.	6.4.6

The staff determined the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for control room habitability considerations.

6.4.6 Conclusion

The staff's review was based on NUREG–0800, SRP Section 6.4, and it addresses plant-specific control room habitability data as discussed in DCD Tier 2, Section 6.4 and the control room air conditioning system as discussed in DCD Tier 1, Section 2.7.3.1.

Pending resolution of the confirmatory item related to RAI 304-8361, Question 06.04-2, regarding the APR1400 Habitability Systems and CRE compliance with GDC 4 and GDC 19, and 10 CFR 50.34(f)(2)(xxviii) the staff concludes that the APR1400 Habitability Systems design complies with 10 CFR 52.47(b)(1). Because the APR1400 design is a single unit, GDC 5 is not applicable. Conformance to the guidelines of RG 1.75 is addressed by a COL information item. The ITAAC and TS requirements will ensure that the CRHS, including CREACS, can be properly inspected, tested, and operated in accordance with DCD requirements, and therefore the staff finds that the APR1400 Habitability Systems design complies with 10 CFR 52.47(b)(1). The above conclusions also apply to the proposed TSC as they apply to TSC radiological habitability.

6.5 Fission Product Removal and Control Systems

The APR1400 Fission Product Removal and Control Systems are designed to prevent or limit the release of fission products following a postulated DBA or fuel handling accident. These systems include the ESF Filter Systems, Containment Spray System, and Containment.

6.5.1 ESF Atmosphere Cleanup Systems/ESF Filter Systems

The ESF Filter Systems consist of filter assemblies, heaters, fans, dampers, and ductwork. They remove particulate and gaseous radioactive material from the atmosphere. Three ESF filter systems work in conjunction with the ventilation systems given below:

- Control Room Emergency Makeup Air Cleaning System (CREACS)
- Auxiliary Building Controlled Area Emergency Exhaust System (ABCAEES)
- Fuel Handling Area Emergency Exhaust System (FHAEES)

6.5.1.1 Introduction

The function of the APR1400 ESF Filter Systems (referred to as ESF Atmospheric Cleanup Systems in NUREG–0800) is to mitigate the consequences of postulated accidents by removing released radioactive material from the ventilated spaces serviced by the systems.

The CREACS, part of the Control Room HVAC System (CRHS), filters the outside makeup air, which has potential to carry radioactive iodine and particulates after a DBA to the CRHS.

The ABCAEES, part of the Auxiliary Building Controlled Area HVAC System, filters the radioactive iodine and particulates in the exhaust air from the mechanical penetration rooms and the safety-related mechanical equipment rooms, which are cooled by safety-related cubicle coolers after a DBA.

The FHAEES, part of the Fuel Handling Area HVAC System, filters the radioactive iodine and particulates in the exhaust air from the fuel handling area after a fuel handling accident in the Auxiliary Building.

6.5.1.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries specifically for the ESF Filter Systems. The ESF filter systems are discussed in DCD Tier 1, Section 2.7.3, “HVAC Systems.” The Sections for the ventilation systems that have ESF filtration capability provide the descriptions and piping and instrumentation diagrams of these ventilation systems, along with design bases and SEs.

DCD Tier 2: The applicant has provided a system description of the ESF filter systems in DCD Tier 2, Section 6.5.1, “Engineered Safety Feature Filter Systems,” summarized here, in part, as follows:

DCD Tier 2, Section 6.5.1.1, “Design Bases,” states that each ESF filter system is designed to meet the requirements of RG 1.52, design and performance recommendations with provisions to filter air, remove moisture, and utilize charcoal adsorption to remove iodine.

- The CREACS is designed in accordance with the requirements of GDC 2, GDC 4, GDC 19, and GDC 60.
- The ABCAEES is designed in accordance with the requirements of GDC 2, GDC 4, and GDC 60.

The FHAEES is designed in accordance with the requirements of GDC 2, GDC 60, and GDC 61.

6.5.1.3 Regulatory Basis

The relevant requirements for the Commission regulations for the ESF filter systems, and the associated acceptance criteria, are given in NUREG–0800, Section 6.5.1, “ESF Atmospheric

Cleanup Systems," and are summarized below. Review interfaces with other SRP Sections also can be found in NUREG–0800, Section 6.5.1.

- GDC 19, as it relates to maintaining the control room in a safe condition under accident conditions, including LOCAs.
- GDC 41, as it relates to providing systems to control the release of fission products to the environment and to control the concentration of hydrogen, oxygen, and other substances in containment following postulated accidents.
- GDC 42, "Inspection of Containment Atmosphere Cleanup Systems" as it relates to designing containment ESF atmosphere cleanup systems to permit inspection.
- GDC 43, "Testing of Containment Atmosphere Cleanup Systems," as it relates to designing containment ESF atmosphere cleanup systems to permit pressure and functional testing.
- GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to design of systems for radioactivity control under normal and postulated accident conditions.
- GDC 64, as it relates to monitoring releases of radioactivity from normal operations, including anticipated operational occurrences, and from postulated accidents.
- The regulations in 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations.

Acceptance criteria adequate to meet the above requirements are provided in:

- RG 1.52 as it relates to the design, inspection, and testing of the ESF filter systems.

6.5.1.4 Technical Evaluation

Review of the ESF Filter Systems in the DCD was performed in accordance with SRP Section 6.5.1, Revision 3, Section III, "Review Procedures." The applicant's design description, applicant's evaluation of the design versus requirements, and results and conclusions reached in this review are as follows:

Each ESF Filter System consists of two independent trains. Each train has an activated charcoal carbon adsorber with dampers, moisture separator, electric heater, pre-filter, High Efficiency Particulate Air (HEPA) filter, and postfilter. A booster fan and isolation dampers are included to provide the flow to the ventilation stack for the discharge of filtered air. Each ESF filter train is powered by an emergency bus that can be powered by a diesel generator.

Ventilation systems are aligned to ESF filter systems to support plant operations and accident mitigation. ESF filters in the CREACS start automatically in response to ESFAS-SIAS, ESFAS-CREVAS, or remote manual actuation. ESF filters in the ABCAEES start automatically in response to ESFAS-SIAS or remote manual actuation. ESF filters in the FHAEES start automatically in response to high radiation signal in the common discharge duct, ESFAS-FHEVAS, or remote manual actuation.

In DCD Tier 2, Section 6.5.1.1, the applicant stated that each ESF Filter System is designed to meet the guidance in RG 1.52, ASME N509, and ASME AG-1, as acceptable for design, inspection, and testing of ESF atmospheric cleanup systems to adequately protect public health and safety.

A detailed comparison of the ESF filter systems in the DCD was made to the acceptance criteria of NUREG–0800, Section 6.5.1, Section II, Acceptance Criteria as follows:

Control Room Dose Design Criteria (GDC 19)

The APR1400 design relies on the CREACS as the system to provide adequate protection against radiation and hazardous chemical releases in order to permit access and occupancy of the control room under accident conditions.

As described in DCD Tier 2, Sections 6.4.1 and 9.4.1, the ESF filter system component in the CREACS is aligned automatically upon receipt of an ESF actuation signal, including SI, or detection of high radiation levels. In addition, the CREACS can be manually aligned.

The applicant used the DBA radiological consequences analysis guidance in RG 1.183, to develop the APR1400 DBA control room dose analyses to show compliance with GDC 19, as it relates to maintaining the control room in a safe condition with adequate radiation protection under accident conditions. The DBA radiological consequences analyses in DCD Tier 2, Chapter 15 show that the ESF filter system within the CREACS limits the control room dose to less than the control room dose criterion in GDC 19 of 0.05 Sv (5 rem) TEDE. The control room dose for each of the analyzed DBAs was shown to be less than 0.05 Sv (5 rem) TEDE in DCD Tier 2, Table 6.4-2, “MCR and TSC Doses from Design Basis Accidents.” Detailed review of the DBA control room dose analyses is discussed in Section 15.0.3 of this SE. This review finds that the control room design and ESF filter system meet the radiological habitability requirements of GDC 19 by limiting the post-DBA dose in the control room for all applicable DBAs to less than 0.05 Sv (5 rem) TEDE.

Review of GDC 19, as it relates to compliance with RG 1.52 regarding the design, testing, and maintenance of the CREACS (with the exception of the CRACS ESF filter system components), and control room habitability is addressed in Sections 9.4.1 and 6.4 of this SER respectively. Review of GDC 19, as it relates to compliance with RG 1.52 as to the design, testing, and maintenance of the CRACS ESF filter system components is discussed below.

Note: CREACS is credited in DBA dose analyses for control room habitability as noted above for GDC 19. ABCAEES is credited for cleanup of the ESF system leakage for the LOCA dose analysis. FHAEES is NOT credited in DBA dose analyses.

GDC 41, GDC 42, and GDC 43

GDC 41, GDC 42, and GDC 43 relate to containment ESF atmosphere cleanup systems. Since the APR1400 design does not have ESF cleanup systems inside containment, these GDC do not apply to APR1400 ESF Filter Systems.

GDC 61

The staff found that GDC 61 applies to FHAEES. Section 9.4.2 of this SE documents the staff review of FHAEES against GDC 61.

Based on the staff review in Section 9.4.2 of this SER, the staff finds that the FHAEES complies with the requirements of GDC 61.

GDC 64

Review of GDC 64 as it relates to compliance with RG1.52 as to the design, testing, and maintenance of ESF filter systems is discussed below.

6.5.1.4.1 Conformance to RG 1.52

The staff reviewed the provisions of the ESF filter systems in DCD Tier 2, Section 6.5.1, “Engineered Safety Feature Filter Systems.” The review was conducted to determine if the guidance of RG 1.52 was met.

DCD Tier 2, Section 6.5.1.1, “Design Bases,” states that the ESF Filter Systems are redundant, designed to seismic Category I, and are powered by an emergency bus that is backed up by an EDG. The structural ability of the filters to operate after a DBA is addressed by their safety-related status and demonstrated by their seismic design. These are specified in DCD Tier 2, Table 3.2-1.

By reviewing the specifications described above and comparing the ESF Filter Systems to RG 1.52, as documented DCD Tier 2, Table 6.5-2, the staff finds that the CRHS complies with the requirements of RG 1.52. Discussion on carbon adsorbers, in the following section, has more information to support this statement.

6.5.1.4.2 Carbon Adsorbers

DCD Tier 2, Table 6.5-1 specifies the use of activated charcoal and a design system efficiency of 99 percent for removal of iodine and organic iodides. The staff finds that this specification conforms to RG 1.52, Table 2.

Other than a stated commitment to meet the guidance in RG 1.52, the staff determined that there was insufficient detailed information in the DCD to demonstrate conformance to the provisions of RG 1.52. Therefore, on October 5, 2015, the staff issued RAI 251-8320, Question 06.05.01-1, the staff requested that the applicant provide details on design and testing in order to conform to RG 1.52:

1. The maximum charcoal loading for the adsorbent trains.

2. Design consideration of iodine desorption and adsorbent auto-ignition.

3. Carbon laboratory test method, whether ASTM D-3803 or another.

4. Total activated carbon bed depth.

5. The Methyl Iodine Penetration Acceptance Criterion while performing laboratory testing for carbon adsorbers.

In its response to RAI 251-8320, Question 06.05.01-1, dated April 22, 2016 (ML16113A437), the applicant stated that:

> *1. The carbon adsorber is designed for a maximum loading of 2.5 milligrams of total iodine (radioactive plus stable) per gram of activated carbon. No*

more than 5 percent of the impregnant (50 milligrams of impregnant per gram of carbon) is used.

2. *The temperature of carbon adsorbent shall not exceed the design limiting temperature of 300 °F due to radioactivity-induced heat in the adsorbent to prevent iodine desorption in accordance with Section 4.10 of ASME N509. Also, the temperature of the carbon adsorbent shall not exceed the ignition temperature of the carbon adsorber, 626 °F in accordance with Section FF of ASME AG-1.*

   The greatest amount of iodine is captured in the carbon adsorber of the control room emergency makeup air cleaning unit (ACU) and the auxiliary building controlled area emergency exhaust ACU under loss of coolant accident (LOCA) conditions. Based on an analysis of the capture of radioactive iodine (elemental iodine and organic iodine) during 720 hours, the maximum radioactivity-induced heat in the adsorbent in the engineered safety feature (ESF) ACUs is as follows:

- *Control room emergency makeup ACU : 0.14 watts*
- *Auxiliary building controlled area emergency exhaust ACU: 16.3 watts*

This maximum radioactivity-induced heat is produced in the auxiliary building controlled area emergency exhaust ACU under LOCA conditions. The maximum amount of radioactivity-induced heat within 720 hours is 16.3 watts. The maximum local temperature inside the carbon adsorber is around 220°F with the ACU fan shut down and the isolation dampers of the inlet and outlet closed. This temperature is well below the design limiting temperature to prevent iodine desorption, 300°F and the ignition temperature of the carbon adsorber, 626 °F.

3. *The laboratory tests for carbon adsorbers are performed in accordance with NRC RG 1.52. Testing is conducted in accordance with ASTM 3803-1991(R2009).*

4. *Each carbon adsorber of the ESF filter systems contains a four inch bed depth.*

5. *The Methyl Iodine Penetration Acceptance Criterion while performing laboratory test for carbon adsorbers is described in Section 5.5.11.c of Chapter 16.*

The staff reviewed Section 5.5.11.c of DCD Tier 2, Chapter 16, its Methyl Iodine Penetration Acceptance Criterion while performing laboratory test for carbon adsorbers is 0.5 percent penetration at 70 percent relative humidity (RH). The acceptance criterion specified in DCD Tier 2, Chapter 16, Section 5.5.11.c, is different from the acceptance criterion, 0.5 percent penetration at 95 percent RH, specified in the standard test method, ASTM D 3803, “Standard Test Method for Nuclear Grad Activated Carbon.” According to Section C.4.i of RG 1.52, systems with humidity control can perform laboratory testing at 70 percent RH instead of 95 percent RH. Since each ACU in the three ESF Filter systems (CREACS, ABCAEES, FHAEES) consists of humidity control features (moisture separator and electric heating coil), the staff finds that testing at 70 percent RH is acceptable because these adsorbent trains had humidity control, which conforms to RG 1.52.

Based on the response to RAI 251-8320, Question 06.05.01-1, dated April 22, 2016, the staff confirms that the carbon adsorber design of the ESF Filter systems follows the guidance of RG 1.52, ASME N509 and ASME AG-1. The staff considers Part 1 of RAI 251-8320, Question 06.05.01-1 resolved.

6.5.1.4.3 High Efficiency Particulate Air Filter

For the ESF filter systems, the applicant specifies in DCD Tier 2, Table 6.5-1 that the efficiency of HEPA filter units is 99 percent. According to DCD, Tier 2, Table 9.4.1-1, the efficiency of all safety-related HEPA filter units is 99 percent (design basis) and 99.97 percent (purchase specification).

Other than a stated commitment to meet the guidance in RG 1.52, the staff determined that there was insufficient detailed information to verify how the specific RG 1.52 guidance is met for the HEPA filters to have sufficient design margin to accommodate fission product loading during an accident without restricting flow rate. The increase in pressure drop between the clean and dirty conditions for any of the ESF adsorber units should be within the normal expected pressure increase for a typical filtered exhaust fan design. Therefore, in RAI 251-8320, Question 06.05.01--1, the staff requested that the applicant provide additional information on HEPA filter design.

In its response to Part 2 of RAI 251-8320, Question No. 06.05.01-1 (ML16113A437), the applicant stated that:

> *A typical single HEPA filter with a size of 24x24x11½ inches, and a minimum rated airflow of 1,500 scfm and has a dust holding capacity of approximately 1,200 grams. The pressure drop of a HEPA filter while clean is less than 1.3 inch water gage and the HEPA filter in a dirty condition with the dust holding capacity of approximately 1,200 grams has a differential pressure across the HEPA filter of 2.6 in. water gage.*
>
> *The maximum mass loading of HEPA filters in the ESF ACUs is much smaller than the dust holding capacity of the HEPA filter. Therefore, the HEPA filters of the ESF ACUs are designed to have sufficient margin to accommodate fission product loading during an accident. Also, the rated airflow capacity of the HEPA filter has 20 percent minimum margin of air flowrate at the design condition.*
>
> *The pressure drop across the ESF filter system, which is needed to size the fan in the ACU, is calculated based on the dirty condition of each filter in the ACU. Therefore, the increase in pressure drop caused by the dirty condition of the filter of ESF ACUs does not degrade the performance of the fan.*

After reviewing the above RAI response, the staff confirms that HEPA filter design of the ESF Filter systems has sufficient design margin to accommodate fission product loading. The staff considers Part 2 of RAI 251-8320, Question 06.05.01-1 resolved.

The status of RAI 251-8320, Question 06.05.01-1 has been revised to **confirmatory item** waiting for future DCD revision.

6.5.1.4.4 Instrumentation Requirements

DCD Tier 2, Section 6.5.1.5 states that flow rate, pressure drop, and status indication of the ACUs are provided based on ASME AG-1 and Table 1 of RG 1.52. The automatic operation and continuous indication of system parameters in DCD Tier 2, Section 6.5.1.5, "Instrumentation Requirements," are claimed by the applicant to demonstrate that the guidance of RG 1.52 and ASME AG-1 is met.

The staff reviewed P&IDs and found that, except status indication for electric heaters, the flow rate, pressure drop, status indication, and temperatures of all the ACUs are provided based on Table 1 of RG 1.52. The relevant readout, recordings, and alarms are monitored in the MCR and RSR.

6.5.1.4.5 Release Point Monitors

For CREACS, two radiation monitors are provided in each outside air intake to monitor the airborne radioactivity of outside makeup air. The high radiation signal causes automatic isolation of CRE.

For ABCAEES, radiation monitors are provided at the common discharge duct of the normal exhaust ACUs to sample air particulate and iodine before it is released to the environment.

For FHAEES, a radiation monitor is provided at the common outlet duct of the fuel handling area normal and emergency exhaust ACUs to monitor the airborne radioactivity of the exhaust air from the fuel handling area.

The staff review of the radiation instrumentation is contained in Section 11, "Radioactive Waste Management," of this SE.

The staff finds that the applicant has adequately provided for monitoring all radioactive releases from the ESF Filter Systems, and that NUREG–0800, Section 6.5.1, Part III, Paragraph 5 is met.

6.5.1.4.6 ITAAC

For the ESF Filter Systems. The applicant proposed ITAAC requirements in DCD Tier 1, Sections 2.7.3.1 (CREACS), 2.7.3.5 (ABCAEES), and 2.7.3.2 (FHAEES). These ITAAC have been reviewed in Sections 6.4, 9.4.1, 9.4.2, and 9.4.5 of this SER. The staff finds that sufficient information has been provided to satisfy NUREG–0800, Section 14.3 and NUREG–0800, Section 14.3.7.

6.5.1.5 Combined License Information Items

No applicable COL items were identified in the DCD. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for ESF Filter Systems consideration.

6.5.1.6 Conclusion

This review addresses the three ESF Filter Systems used by the three ventilation systems discussed in DCD Tier 2, Section 6.5.1 and their mitigation of control room and offsite dose. The staff finds that the applicant has used RG 1.183 to calculate the DBA radiological releases and expected CRHS inlet conditions that would result from accidents in the proposed DCD. The staff concludes that, pending closure of the **confirmatory item related to RAI 251-8320, Question 06.05.01-1,** as discussed above, the proposed design and operation of the ESF Filter

Systems will provide adequate fission product removal in post-accident environments. Following the guidance in SRP Section 6.5.1, acceptability of the APR1400 ESF Filter Systems was determined by reviewing the plant-specific data against the criteria stated in the RG 1.52. SRP Section 6.5.1 states that conformance to the requirements of the provisions of RG 1.52 constitutes acceptable bases for satisfying the requirements of GDC 19, GDC 41, GDC 42, GDC 43, GDC 61 and GDC 64, and 10 CFR 52.47(b)(1). Therefore, with exception of the RAIs identified above, the staff finds that the ESF Filter Systems used in the APR1400 for atmosphere cleanup meet the requirements of GDC 19, GDC 41, GDC 42, GDC 43, GDC 61 and GDC 64; and 10 CFR 52.47(b)(1).

6.5.2 Containment Spray System (CSS)

6.5.2.1 Introduction

In the event of a design basis LOCA there is an assumed core degradation that results in a significant release of radioactivity to the containment atmosphere. This activity would consist of noble gases, particulates, and a small amount of elemental and organic iodine.

The CSS is a safety-related system design for heat removal and fission product removal. This system mitigates the DBA that release fission products into the containment.

6.5.2.2 Summary of Application

DCD Tier 1: The Tier 1 information for this Section is found in the DCD Tier 1, Section 2.11.2, “Containment Spray System.”

DCD Tier 2: The applicant has described the CSS in DCD Tier 2, Section 6.5.2 “Containment Spray Systems.” The heat removal function and design evaluation calculations description appears in DCD Tier 2, Section 6.2.2, which contains sections on all major components. Details of the fission-product removal for DBA are provided in DCD Tier 2, Chapter 15, “Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents.” Two key components of the system are the IRWST and the trisodium phosphate (TSP) baskets in the HVT. These are described in DCD Tier 2, Section 6.3.2. The applicant described testing and monitoring of the IRWST and the TSP in TSs 3.5.4 and 3.5.5 (DCD Chapter 16), and has provided additional information of the pH calculations in a proprietary document, ““Containment Sump pH Analysis” Calc. No. 1-035-N387-008,” provided to the staff for auditing on November 13, 2015.

ITAAC: The ITAACs associated with DCD Tier 2, Section 6.5.2 are given in DCD Tier 1, Table 2.11.2-4.

TS: The TSs associated with DCD Tier 2, Section 6.5.2 are given in DCD Tier 2, Chapter 16, Section 3.5.5, “Containment Spray System.”

6.5.2.3 Regulatory Basis

The relevant requirements for the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 6.5.2 of NUREG–0800 and are summarized below. Review interfaces with other SRP Sections can be found in Section 6.5.2.I of NUREG–0800.

- GDC 41, as it relates to containment atmosphere cleanup and control of fission product releases in postulated accidents.

- The regulations in 10 CFR 52.47(b)(1), which requires DC applications to contain the required ITAAC.

6.5.2.4 Technical Evaluation

The staff reviewed the information provided in DCD Tier 2, Sections 6.5.2, Revision 0 and the supplemental information provided in applicant letter dated November 13, 2015 (ML15317A522) against the guidance of SRP Section 6.5.2 (NUREG–0800). Section 6.5.2 contains a number of detailed acceptance criteria that should be met concerning system performance and the calculation of fission product removal from the containment atmosphere. The applicant has also cited ANSI/ANS 56.5, “PWR and BWR Containment Spray System Design Criteria,” as a standard for system design.

The CSS is an engineered safety feature which serves to remove both heat and fission products from the containment atmosphere. The subject of heat removal is discussed in Section 6.2.2 of this SER, while fission product removal is the focus of this review. This feature of the CSS is intended to mitigate the effects of DBAs in which iodine transport and speciation is of paramount concern. The pH is an important consideration for iodine retention in water. Hence, this review will consider pH in spray and sump solutions. Additionally, the staff performs an independent calculation of pH in conjunction with accident analyses in Section 15.0.3 of this SER.

The IRWST is a large tank of borated water in containment. A minimum of 86,768 ft^3 (2457 m^3) is maintained at room temperature (70-120°F, or 21-49°C) with a minimum boric acid concentration of 4000 ppm. TSP is stored in baskets at the HVT. The TSP is fully dissolved in the return flow from the sprayed water or break flow.

Containment sprays draw water from the IRWST and send it through CSS piping and out the spray nozzles near the top of the containment airspace. The water then drains down back to the IRWST as illustrated in DCD Tier 2, Figure 6.2.2-5, “Sprayed Regions.” As described in DCD Tier 2, Section 6.5.2.2, “System Design (for Fission Product Removal),” approximately 75 percent of the total containment free volume is sprayed. Unsprayed regions are mostly comprised of covered regions such as pressurizer and SG compartments.

Initially, the IRWST water has a pH of about 4.5 according to the calculation procedure used in the EPRI PWR Primary Water Chemistry Guidelines. The DCD mentions an initial pH of 4.15, which rises above a pH 7, 157 minutes after a DBA. This timing meets the SRP Section 6.5.2 recommendation that pH be raised above 7 before the recirculation mode of the CSS is actuated (SRP Acceptance Criteria II.1.G, p. 6.5.2-5). The staff was unable to corroborate this calculation since the DCD contains insufficient detail on the actual calculation performed by the applicant to do so. In its November 13, 2015 letter, the applicant provided supplemental information that included a proprietary calculation report (Calc. No. 1-035-N387-008-01). However, when the report numbers were compared to the ones in the DCD, the staff found that they do not match. For that reason, on January 11, 2016, the staff issued RAI 363-8446 Question 06.05.02-22, requesting that the applicant explain these errors. In its response to RAI 363-8446, Question 06.05.02-2, dated February 5, 2016, the applicant stated that the numbers provided in DCD Tier 2, Section 6.5.2, Table 6.5-4 are accurate and they will be revising the proprietary calculation report to reflect this change. In August 2016 the staff was able to review the revised calculation report (Calc. No. 1-035-N387-008-02) and performed a confirmatory analysis. Based on this revised calculation, the staff finds the pH calculation acceptable because it meets the recommendations of SRP Section 6.5.2 and NUREG–0800 BTP 6-1 “pH for Emergency Coolant Water for PWRs.”

The applicant mentions that a minimum of 58,358 lb. (26,471 kg) of TSP is stored in baskets inside the HVT, in good position to fully dissolve with the containment water. This mineral dissolves readily and forms a strongly basic solution. Thus, as water cycles through the sprays several times, the pH of the IRWST will gradually rise. This material is important in accident mitigation, and the applicant has listed TSs and SRs to assure both the quantity and quality of stored TSP in DCD Tier 2, Chapter 16, SR 3.5.5. The pH analysis provided in the revised calculation report (Calc. No. 1-035-N387-008-02) mentions the same minimal quantity of TSP needed to meet the pH value as the one provided in DCD Table 6.5-4, “Major Parameters Used in pH Calculations.”

Some of the containment atmosphere does not get sprayed by the CSS nozzles. These are described in the DCD as “unsprayed regions.” The applicant calculated that 25 percent of the containment volume does not get sprayed. Although high, this region is in good communication with the sprayed region and the mixing rate between them was used in the analysis as recommended by the SRP.

The importance of pH control in containment water is to ensure retention of fission product iodine. Many forms of iodine are scrubbed from the containment atmosphere by spray water and deposited in the RWSP. Under neutral or basic conditions iodine will stay dissolved in water, primarily as iodide (I^-) and, to a lesser extent, iodate (IO_3^-). However, acidic solutions can produce more volatile species (chiefly I_2), which can evaporate from the water and re-enter the containment airspace. In this form, it is much more likely to be released outside of containment through venting or leakage.

Assessing the consequences of a DBA requires calculation of fission product removal from the containment airspace by containment sprays. The acceptable methods for such calculation are described in SRP Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” III.4,, and constitute first-order rate equations. The applicant does not include noble gases or organic iodides consistent with the guidelines in the SRP. The applicant’s analyses in DCD Tier 2, Chapter 15 include removal of elemental iodine (I_2) by natural deposition onto wetted wall and removal of particulate iodine by the sprays. Both processes are estimated using models given in SRP Section 6.5.2.

Additionally, the applicant credits removal of particulate iodine through natural processes inside containment using a model by Powers, NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments,” that is consistent with the models in the SRP, and also follows guidance in RG 1.183, Appendix A, position 3.2, with respect to credit for aerosol natural deposition in the containment. The staff’s review of aerosol removal through natural processes is discussed in more detail in Sections 6.5.3 and 15.0.3, “Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors,” of this SER.

The removal of iodine by rate processes is not unlimited, as there is a maximum amount of iodine that can be dissolved in a given volume of water. This equilibrium is established for each species between the two phases (water and air). It is commonly measured using the Henry’s Law constant, i.e., the ratio between equilibrium concentrations (C*) in liquid and gas:

$$H = C_l^* \Big/ C_g^*$$

Sometimes the inverse of H is called the partition coefficient; to maintain consistency with the SRP, the phrase "partition coefficient" is identical to H. The decontamination factor (DF) is the ratio of activity before and after some decontamination process. In this case, it is a ratio of total activity to activity remaining in the gas phase after some has dissolved in water, and can be derived from the partition coefficient using the equation:

$$DF = \frac{n_g + n_l}{n_g} = 1 + \frac{HV_l}{V_g}$$

where n_l, V_l, n_g, and V_g represent moles and volumes of liquid and gas, respectively. The equilibrium partitioning between air and water for I_2 depends strongly on temperature and solution pH. The analysis should assume sufficient mixing by recirculation so that an equilibrium value is a reasonable estimate of the end result of transient dissolution and revaporization processes.

As discussed in DCD Tier 2, Section 15.6.5.5.1.1, the applicant's DBA analyses assume that all I_2 is removed from the containment atmosphere at a rate of 20 hr^{-1} and I_2 removal ceases after a DF of 200 has occurred at 2.25 hours, which is consistent with the guidance in SRP Section 6.5.2 for elemental iodine. The applicant used information in SRP Section 6.5.2 for removal of elemental iodine by sprays which is applicable during injection of fresh (non-recirculated) water. DCD Tier 2, Section 15.6.5.5.1.1, "Containment Leakage," states that the "APR1400 does not have a recirculation mode of operation during the CSS operation period because the CSS takes suction from IRWST for the entire duration of the design basis event." However, the IRWST will receive a flow of contaminated water from the containment during the accident, via spillage to the holdup volume tank where the TSP baskets are located for pH control. In RAI 111-7971, Question 06.05.02-1, issued July 24, 2015, the staff asked that the applicant provide further information on the issue of water movement inside containment during CSS operation and the effect on the calculated elemental iodine removal rate using SRP Section 6.5.2 methods. By letter dated September 2, 2015 (ML15245A313), the applicant confirmed that although there is no change from an injection mode to a recirculation mode for the CSS because the CSS takes suction from the IRWST for the entire duration of CSS operation, there is circulation of coolant within the containment, as described above. The sprayed water, which contains iodine removed from the containment atmosphere, is collected in the HVT then spills into the IRWST, both of which have adequate pH control to retain the iodine within the water as TSP is dissolved and mixed into those volumes. As the spray water spills into the IRWST, it travels along with the water from the HVT which includes dissolved TSP and is at a very high pH, therefore iodine re-evolution to the containment atmosphere is considered to be negligible. The applicant's response provided proposed revised text for DCD Tier 2, Section 15.6.5.5.1.1 to clarify that the APR1400 does not change the CSS operation mode from injection mode to recirculation mode. The staff finds that the applicant's clarification on coolant circulation and iodine transport within the APR1400 containment during CSS operation resolves the staff's question, in RAI 111-7971, question 06.05.02-1.

Removal of particulate iodine from the containment atmosphere by sprays is input to the DBA dose analyses as a spray removal coefficient calculated by the applicant as 6.25 hr^{-1}. When a DF of 50 is reached, the spray particulate removal coefficient is decreased by a factor of ten, in accordance with the guidance in SRP Section 6.5.2. Particulate iodine removal by sprays is not credited after 4 hours. The applicant used methods to estimate particulate iodine removal by sprays that are consistent with SRP Section 6.5.2 guidance and are therefore acceptable.

6.5.2.5 Combined License Information Items

Item No.	Description
6.5(1)	The COL applicant is to provide the operational procedures and maintenance program as related to leak detection and contamination control.
6.5(2)	The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations.

6.5.2.6 Conclusion

The applicant has described the operation of the containment sprays, including the control of pH in containment water and the removal of fission products by the sprays. The staff therefore concludes, based on the information supplied by the applicant, that the requirements of GDC 41 will be met.

6.5.3 Fission Product Control Systems and Structures

The release of fission products following a postulated DBA is mitigated by several APR1400 design features. This section provides the evaluation of those features that prevent or limit the release of fission products from primary containment. The DBA radiological consequences analyses and assumed sequence of events that demonstrate the effectiveness of these fission product removal and control systems in maintaining radioactivity releases within regulatory limits are presented in DCD Tier 2, Chapter 15.

The fission product control system (FPCS) contains or processes fission products not removed from the containment atmosphere and the ventilated spaces serviced by other systems. The primary fission product control systems following a DBA are the CSS and the containment. The fission product leakage to the environment is reduced below the release limit by the fission product removal function of the containment spray and the leaktight pressure boundary of the containment.

6.5.3.1 Introduction

The purpose of this section is to evaluate the APR1400 systems that are designed to ensure that radiological releases during normal and accident conditions are below the reference dose values used in the evaluation of plant design features with respect to postulated reactor accidents established in 10 CFR 52.47(a)(2), “Contents of applications; technical information.” The system and component design criteria for fission product control systems are outlined in RG 1.52, Regulatory Positions C.1, C.2, and C.3.

6.5.3.2 Summary of Application

The FPCS is described in multiple DCD sections for the primary containment structure and ventilation systems, and for gaseous elemental iodine (I2) control. These sections are given below and have been reviewed with respect to design features associated with fission product control system.

DCD Tier 1: There are no DCD Tier 1 entries specific to the FPCS.

DCD Tier 2: DCD Tier 2, Section 6.5.3.

The primary mechanism to limit release of fission products that are produced following a DBA is provided by the containment structures. A description of the primary containment structures, acceptable leakage criteria, and design features used for fission product control is provided in the following DCD Tier 2, Sections:

- 3.8.1, “Concrete Containment”
- 6.2.1, “Containment Functional Design”
- 6.2.4, “Containment Isolation System”
- 6.2.6, “Containment Leakage Testing”

Containment purge system operation is not required and is isolated during a DBA. The heat and fission products released into the containment during a DBA, such as a LOCA or an MSLB, are removed by the CSS, which is reviewed in Section 6.5.2 of this SER. The APR1400 design does not include a containment hydrogen purge system.

Fission product retention credit may be taken by the applicant for other systems:

- filtration and adsorption units as evaluated in Section 6.5.1 of this SER.
- pH control of IRWST water as evaluated in Section 6.5.2 of this SER.

6.5.3.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG–0800, Section 6.5.3, “Fission Product Control Systems and Structures,” and are summarized below. Review interfaces with other SRP Sections also can be found in NUREG–0800, Section 6.5.3.

- GDC 41, as it relates to the containment atmosphere cleanup system being designed to control fission product releases to the environment following postulated accidents.
- GDC 42, as it relates to the containment atmosphere cleanup system being designed to permit periodic inspections.
- GDC 43, as it relates to the containment atmosphere cleanup system being designed to permit appropriate functional testing.
- The regulations in 10 CFR 52.47(a)(2), as it relates to ensuring that nuclear power plant radiological releases during accident conditions are below the reference dose values used in the evaluation of plant design features with respect to postulated reactor accidents.

- The regulations in 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC has been constructed and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations.

Acceptance criteria adequate to meet the above requirements include and are provided in:

- RG 1.52, Regulatory Positions C.1, C.2, and C.3, as they relate to system and component design criteria for fission product control systems.
- RG 1.183
- RG 1.206

6.5.3.4 Technical Evaluation

The technical evaluation is based on review of the primary and secondary containment structures and associated systems designed to mitigate the release of fission products. The secondary containment is not applicable to the APR1400. This review includes DCD Tier 2 Sections identified above. The design features used for FPCS include structural barriers provided by the primary containment, filtration by ventilation systems, natural deposition, containment spray and chemical addition. The design basis of each of these FPCS features was compared against acceptance criteria provided in SRP Section 6.5.3, RG 1.52, RG 1.183, and RG 1.206. Airborne radioactivity released into the containment during a DBA LOCA is bounding over that released during a rod ejection accident, which is the only other DBA evaluated for offsite and control room dose that includes radioactive material release to the containment. Therefore, the LOCA analysis was reviewed as the bounding case for determining acceptable FPCS design. Specific acceptance criteria applicable to each design feature for determination of an acceptable FPCS design are given in SRP Section 6.5.3 and include the following:

6.5.3.4.1 Primary Containment

The containment encloses the reactor vessel, SGs, reactor coolant loops, and portions of the auxiliary and ESF systems. The containment provides reasonable assurance that leakage of radioactive material to the environment does not exceed the acceptable dose limit as defined in 10 CFR 50.34 even if a LOCA occurred.

The APR1400 containment building consists of a cylindrical shell and dome with a steel liner plate. The containment is designed as an essentially leak-tight barrier that accommodates the calculated temperature and pressure conditions resulting from the complete spectrum of postulated breaks, up to and including a double-ended slot break in the RCS or secondary system piping.

The design leakage rate for the containment is 0.1 percent free volume per day at the DBA pressure. During the CILRT, the containment is isolated and pressurized in accordance with NEI 94-01 and ANSI/ANS 56.8. The acceptance criteria specified in NEI 94-01 and ANSI/ANS 56.8 for the CILRT includes margin for possible deterioration of the containment leakage integrity during the service intervals between tests. Therefore, the measured leak rate (Lam at peak test pressure, Pa) is limited to less than 0.75 of the maximum allowable value (La). The staff review on primary containment leakage testing is discussed in sections 6.2.6 of this SER.

6.5.3.4.2 Natural Deposition

The APR1400 DBA radiological consequences analyses take credit for natural deposition inside primary containment during a DBA LOCA and control element assembly ejection (CEAE) accident. The radiological evaluations are based on the guidance in SRP Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," and RG 1.183. Detailed review of natural deposition as modeled in the APR1400 DBA dose analyses is discussed in Section 15.0.3 of this SER.

The APR1400 primary containment is designed to contain the energy released from the RCS in the event of a LOCA or CEAE. The M&E released into the containment is from the reactor coolant. Coolant released from the primary system causes an increase in containment steam mass, which in turn increases containment pressure and temperature. This rise is limited by steam cooling and condensing on contact with colder surfaces. The passive heat sink inside the primary containment consists of all painted and unpainted concrete and metal surfaces that are exposed to the primary containment atmosphere. The specific passive heat sinks considered in the containment pressure-temperature analysis and their parameters are given in DCD Tier 2, Table 6.2.1-23, "Passive Heat Sink Data." A minimum heat sink surface area was conservatively considered.

As described above, water vapor condenses on contact with the internal surfaces during a LOCA or CEAE. Natural processes that result in deposition of fission products on the surfaces inside containment are credited for fission product removal from the containment atmosphere in the DBA LOCA and CEAE radiological consequences analyses. Reduction in airborne radioactivity in the containment by natural deposition within the containment is acceptable per RG 1.183, Appendix A, Section 3.2. Natural deposition of radioactive particulates and elemental iodine on surfaces within containment is addressed in DCD Tier 2, Sections 6.5.2, 6.5.2.3.3 and 15.6.5.5.1.1. The APR1400 deposition model is based on natural deposition models in NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NRC, December 1997, and Supplements (Supplement 1, June 1999, and Supplement 2, October 2002).

The applicant's analysis takes credit for aerosol natural deposition in the containment based on the correlation model described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," incorporated into RADTRAD as the Powers model for containment aerosol natural deposition. The applicant used the 10^{th-}percentile removal coefficients in the Powers natural deposition correlation, in accordance with the DBA analysis guidance in RG 1.183 for currently operating reactors. On July 6, 2015, the staff issued RAI 60-7972, Question 06.05.03-1, requesting the applicant to demonstrate that the Powers natural deposition model is appropriate to use for the APR1400, considering that the Powers natural deposition model was developed using design information from currently operating PWRs and BWRs. In its response to RAI 60-7972, Question 06.05.03-1, dated September 2, 2015, the applicant provided information to show that the APR1400 design parameters are within the applicability range for NUREG/CR-6189. The APR1400 is a PWR design with a large dry containment, similar to those used to develop the correlation in the Powers natural deposition model. NUREG/CR-6189, Section III.A gives a correlation of reactor thermal power to the containment free volume range that is applicable to the aerosol deposition model. The APR1400 containment free volume is within the range of containment volumes for a

3,983 MWth nominal power PWR in that correlation. Additionally, when a 2-percent power uncertainty is applied, the containment volume remains in the applicable correlation range. Based on the information showing that the APR1400 nominal power and containment volume are within the correlated ranges that are used in the basis for the Powers natural deposition model, staff finds that the applicant has shown that the Powers natural deposition model is applicable for the APR1400 design. Therefore, RAI 60-7972, Question 06.05.03-1 is resolved and closed.

Detailed discussion of the staff's review and acceptance of the modeling of aerosol and iodine removal in containment for the LOCA and CEAE is included in Section 15.0.3 of this SER in conjunction the staff's review of the DBA radiological consequence analyses which show compliance with the requirements of 10 CFR 52.47(a)(2).

As described above, the staff has determined that the APR1400 DBA dose analysis credit of natural deposition as a means of fission product control is consistent with the guidance in RG 1.183 and SRP Section 15.0.3, and is therefore acceptable.

6.5.3.4.3 Containment Building Ventilation System

The RCB purge system is designed to clean up the containment atmosphere during normal operation and to maintain suitable environmental conditions during refueling condition. Since the RCB purge system is isolated by CIVs upon receipt of an ESFAS-CIAS or an ESFAS-CIAS signal, it is not considered part of the FPCS. A detailed review of this system is documented in Section 9.4.6 of this SE.

6.5.3.5 Combined License Information Items

No applicable COL items were identified in the DCD. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for the fission product control system.

6.5.3.6 Conclusion

Several plant features serve to reduce or limit the release of fission products following a postulated accident. These systems include the containment structures, CSS, and ESF filter systems. The APR1400 DBA radiological consequences analyses take credit for fission product removal and control systems and structures. The APR1400 DBA dose analyses also incorporate credit for passive fission product removal in the primary containment, in accordance with RG 1.183 guidance on DBA radiological consequences analyses using an alternative source term. The DCD discusses the performance capability of each system used for fission product control, including operation following a DBA.

For the reasons described above, the staff has determined that the FPCS design meets the guidance of RG 1.52, Regulatory Positions C.1, C.2, and C.3, the recommendations in SRP Section 6.5.3, and is acceptable. Accordingly, the staff finds that the APR1400 FPCS design is in compliance with the applicable GDCs of 10 CFR Part 50, Appendix A and satisfies the applicable requirements of 10 CFR 52.47.

6.6 Inservice Inspection and Testing of Class 2 and 3 Components

6.6.1 Introduction

Inservice inspection (ISI) programs are based on the requirements of 10 CFR 50.55a, which requires that ASME Code Class components meet the applicable inspection requirements set forth in ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." ISI includes preservice inspection (PSI) prior to initial plant startup.

6.6.2 Summary of Application

DCD Tier 1: There are no final DCD Tier 1 entries for this area of review. The system-based descriptions of DCD Tier 1, Chapter 2, "Design Descriptions and ITAAC," address ASME design-related Code requirements for system components.

DCD Tier 2: The applicant has provided a DCD Tier 2 description of its ISI program for ASME Class 2 and 3 components in Section 6.6, summarized here, in part, as follows:

PSI and ISI of ASME Code Class 2 and 3 components is performed in accordance with ASME Section XI. The code of record for PSI and ISI at the DC stage is the 2007 edition with the 2008 Addenda of ASME Section XI. Welds and other areas requiring periodic inspection are made accessible. Examination techniques to be used include visual, surface, and volumetric methods. Preservice examination and subsequent inservice examination are conducted using equivalent equipment and techniques. ASME Code Class 2 and 3 systems are pressure tested as applicable. The ISI program is augmented to address high-energy fluid system piping between CIVs.

The application addresses:

- Components subject to examination
- Accessibility
- Examination techniques and procedures (e.g., visual, liquid penetrate, magnetic particle, eddy current, ultrasonic, radiography)
- Inspection intervals
- Examination categories and requirements
- Evaluation of examination results
- System pressure tests
- Augmented ISI to protect against high-energy piping failures

In each of these areas, the application references the applicable ASME Code requirements. A COL applicant that references the APR1400 DC is to identify the implementation milestones for the ASME Section XI ISI program for Class 2 and Class 3 components. The COL applicant is also to identify the implementation milestone for the augmented ISI program. (See Combined License Information Items 6.6(1) and 6.6(2)).

ITAAC: The ITAAC associated with DCD Tier 2, Section 6.6 are given several Sections of DCD Tier 1. These ITAAC indicate that inspections will be performed on as-built components and piping, and that reports exist that conclude the following:

(1) As-built ASME Code components, piping, and supports are designed and constructed in accordance with ASME Section III requirements.

(2) The ASME Section III requirements are met for nondestructive examination of the pressure boundary welds in as-built ASME Code components and piping.

(3) The results of hydrostatic testing of the as-built ASME Code components and piping conform to ASME Section III requirements.

TS: There are no TS for this area of review.

6.6.3 Regulatory Basis

The relevant requirements for the Commission regulations for this area of review, and the associated acceptance criteria, are specified in NUREG–0800, Section 6.6, "Inservice Inspection and Testing of Class 2 and 3 Components," and are summarized below. Review interfaces with other Sections also can be found in the NUREG–0800, SRP Section 6.6.

- GDC 36, as it pertains to designing the ECCS to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel.

- GDC 37, as it pertains to designing the ECCS to permit appropriate testing to assure structural integrity, leak tightness, and the operability of the system.

- GDC 39, as it pertains to designing the containment heat removal system to permit inspection of important components, such as the torus and spray nozzles to assure the integrity and capability of the system.

- GDC 40, as it pertains to designing the CHRS to permit appropriate pressure and functional testing.

- GDC 42, as it pertains to designing the containment atmospheric clean up system to permit appropriate inspection of components such as filter frames and ducts.

- GDC 43, as it pertains to designing the containment atmospheric clean up system to permit appropriate periodic pressure and functional testing to assure structural integrity of components and the operability and performance of active components of the system, such as fans, filters, and dampers.

- GDC 45, "Inspection of Cooling Water Systems," as it pertains to designing the cooling water system to permit appropriate periodic inspection of important components, such as heat exchangers.

- GDC 46, "Testing of Cooling Water Systems," found in 10 CFR Part 50, Appendix A, as it pertains to designing the cooling water system to permit appropriate periodic pressure and functional testing to assure structural and leak-tight integrity of its components.

- The regulations in 10 CFR 50.55a, as it pertains to specification of the preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and 3 systems and components.

- The regulations in 10 CFR 52.47(b)(1) requires that the DC application contain proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations.

6.6.4 Technical Evaluation

The staff reviewed DCD Tier 2, Section 6.6, "Inservice Inspection of ASME Code Class 2 and 3 Components," in accordance with SRP Section 6.6. The ASME Code of record for the APR1400 is the 2007 Edition with the 2008 Addenda (see DCD, Tier 2, Section 5.2.1.1.). ASME Section XI, IWC and IWD presents the PSI and ISI requirements for ASME Code Class 2 and Class 3 components.

6.6.4.1 Components Subject to Inspection

The applicant's definition of ASME Code Class 2 and 3 components and systems subject to an ISI program is acceptable if it is in agreement with the NRC quality group classification system or the definitions in ASME Section III, NCA-2000. In addition, the staff reviewed any exceptions to the inspection and testing of components that are different from the ASME Section XI requirements.

APR1400 DCD Tier 2, Section 6.6.1, "Components Subject to Examination," states that DCD Tier 2, Table 3.2-1, specifies safety classes for components that have a safety function in accordance with ASME Section III, NCA-2000 and that DCD Tier 2, Section 3.2.2, "System Quality Group Classification," defines the relationship between the safety classes and RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." DCD Tier 2, Section 3.2.2, states that systems and components are assigned to quality groups in accordance with the quality group classification system defined in RG 1.26. The Quality Group definitions in DCD Tier 2, Section 3.2.2 also indicate that Quality Group B components are designed to ASME Section III, NC (Class 2) requirements and Quality Group C components are constructed to ASME Section III, ND (Class 3) requirements. On this basis, the staff determined that the definition of Class 2 and Class 3 components subject to inspection is acceptable because it in agreement with the NRC quality group classification system and the definitions in ASME Section III, Article NCA-2000. The definition of the components subject to inspection meets the requirements of the ASME Code and the NRC quality group classification system, and is therefore, acceptable. The staff's review to verify that systems and components are appropriately classified in accordance with regulatory requirements and NRC quality group classification guidance is documented in Section 3.2.2 of this SER.

DCD Tier 2, Section 6.6.1 states that ASME Code Class 2 and 3 pressure retaining components in the APR1400 design are examined in accordance with the requirements of ASME Section XI, Subarticles IWC-2500 and IWD-2500, respectively. The applicant also states that ASME Section XI, Subsubarticles IWC-1220 or IWD-1220 allow the exemption of certain components or portions of components from examination and those exempted items are listed in the in-service inspection program. However, upon reviewing the DCD, the staff could not determine if any specific ASME Code Class 2 or 3 components in the APR1400 design would be generically exempted from examination. This issue was discussed with the applicant in a public meeting on June 30, 2015, and the applicant responded that exemptions from Code examination requirements, as permitted by ASME Section XI, Subsubarticles IWC-1220 and IWD-1220, will be listed in the PSI program and the ISI program to be provided by the COL applicant. The applicant also provided a COL information 6.6(4), which states that the COL applicant is to provide the PSI program and the ISI program. The applicant's response is acceptable because it provides a reasonable assurance that ASME Section XI, Subsubarticles IWC-1220 and IWD-1220, will be met. The applicant also provided a markup copy of the proposed changes in a letter dated July 17, 2015 (ML15198A557) and revised the DCD accordingly. The staff determined that the applicant's revisions to the DCD have appropriately addressed the issue and consequently finds that DCD Tier 2, Section 6.6 is adequate as it pertains to components subject to inspection.

6.6.4.2 Accessibility

The design and arrangement of Class 2 and Class 3 systems should include allowances for adequate clearances to conduct the examinations specified in ASME Section XI, Articles IWC-2000 and IWD-2000 at the frequency specified. The design and arrangement of system components are acceptable if an adequate clearance is provided in accordance with ASME Section XI, Subarticle IWA-1500, "Accessibility." Also, 10 CFR 50.55a(g)(ii) requires that Class 2 and 3 components, including supports, be designed and provided with access to enable the performance of ISI.

DCD Section 6.6.2 states that provisions for accessibility are incorporated in the design processes for ASME Code Class 2 and 3 components in accordance with ASME Section XI, Subarticle IWA-1500. The applicant also states that provisions are made in the design and layout of Code Class 2 and 3 systems to allow for conformation with the requirements of ASME Section XI, Articles IWC-2000 and IWD-2000. In addition, the DCD states that ASME Code Class 2 and 3 components requiring inspection are designed for and provided with access to enable the performance of ASME Section XI inspections onsite. The staff determined that the information described above is acceptable because it meets the requirements of ASME Section XI and 10 CFR 50.55a.

The staff noted an inconsistency in the treatment of Class 1, Class 2, and Class 3 components regarding accessibility to inspect. In DCD Section 5.2.4, a COL Item is provided that requires the COL applicant to address the accessibility of Class 1 components for ISI if the design of the APR1400 Class 1 component is changed from the DCD design. However, there is no COL information item to address this issue for Class 2 and 3 components even though the requirement for accessibility to inspect is the same. This issue was discussed with the applicant in a public meeting on June 30, 2015, and the applicant responded that a new COL item will be added to the DCD requiring COL applicants to address the provisions for accessibility of Class 2 or 3 components for inspection if the design is changed from the DCD design. Subsequently, in a letter dated July 17, 2015 (ML15198A557), the applicant provided a markup copy of the proposed changes and revised the DCD accordingly. The changes are acceptable because they provide a reasonable assurance that the requirements of 10 CFR 50.55a(g)(3) will be met for the APR1400. The staff determined that the applicant's revisions to the DCD have appropriately addressed the issue and consequently that DCD Tier 2, Section 6.6 is adequate as it relates to accessibility.

6.6.4.3 Examination Techniques and Procedures

The applicant's examination techniques and procedures used for preservice inspection or ISI of the system are acceptable if they meet the following criteria:

- The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with ASME Code, Section XI, Articles IWA-2000, IWC-2000, and IWD-2000.

- Alternative examination methods, combinations of methods, or newly developed techniques to those above are acceptable provided that the results are equivalent or superior, in accordance with 10 CFR 50.55a(a)(3).

- The methods, procedures, and requirements regarding qualification of personnel performing ultrasonic examination reflect the requirements provided in ASME Code, Section XI, Division 1, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination."

- The performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws reflects the requirements provided in Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Division 1 of ASME Code, Section XI.

DCD Tier 2, Section 6.6.3, "Examination Technique and Procedures," states that the examination techniques to be used for in-service inspection include visual, surface, and volumetric examination methods. The DCD also states that the techniques and procedures meet the requirements of ASME Section XI, Articles IWC-2000 and IWD-2000 and that PSI and ISI are conducted with equivalent equipment and techniques. In addition, DCD Tier 2, Section 6.6.3 states that ultrasonic examination personnel, equipment, and procedures are qualified in accordance with ASME Section XI, Appendix VII and VIII. The staff could not determine whether the information provided was in accordance with ASME Section XI because the applicant did not identify which specific examination methods would be used. This issue was discussed with the applicant in a public meeting on June 30, 2015, and the applicant responded that techniques such as ultrasonic, magnetic particle, liquid penetrant, and visual examination will be used and will meet the requirements of ASME Section XI. The applicant also responded that the DCD will be revised to state that the examination techniques and procedures to be used will be specified by the COL applicant in the PSI and ISI program. Subsequently, in a letter dated July 17, 2015 (ML15198A557), the applicant provided a markup copy of the proposed changes and revised the DCD accordingly. The applicant's changes are acceptable because they provide a reasonable assurance that the examination techniques and procedures applied to Class 2 and 3 components of the APR1400 design will comply with the requirements of ASME Section XI. The staff determined that the applicant's revisions to the DCD have appropriately addressed the issue consequently that DCD Tier 2, Section 6.6 is adequate as it relates to examination techniques and procedures.

6.6.4.4 Inspection Intervals

The required examinations and pressure tests should be completed during each 10-year interval of service, hereinafter designated as the "inspection interval." In addition, the scheduling of the program should comply with the provisions of ASME Code, Section XI, Articles IWA-2000, IWC-2000, and IWD-2000, as related to inspection intervals of the ASME Code, Section XI.

APR1400 DCD Tier 2, Section 6.6.4 states that inspection schedules and intervals for Class 2 and 3 components are in accordance with ASME Section XI, Subarticles IWA-2400, IWC-2400, and IWD-2400. The DCD also indicates that the length of the inspection interval is defined as 120 months. The inspection interval specified for the APR1400 Class 2 and 3 components are consistent with the definition in Section XI of the ASME Code, and therefore, are acceptable.

6.6.4.5 Examination Categories and Requirements

The examination categories and methods specified in the DCD are acceptable if they agree with the criteria in ASME Code, Section XI, Articles IWA-2000, IWC-2000, and IWD-2000. Every area subject to examination should fall within one or more of the examination categories and should be examined at least to the extent specified. DCD Tier 2, Section 6.6.5 lists the specific examination categories for ASME Code Class 2 and 3 components. The examination categories provided in the DCD are acceptable because they are identical to the examination categories provided for Class 2 and 3 components in ASME Section XI.

6.6.4.6 Evaluation of Examination Results

The methods for evaluation of examination results are acceptable if they are in compliance with ASME Section XI, Articles IWC-3000 and IWD-3000. Proposed repair and replacement activities are acceptable if they are in compliance with ASME Section XI, Article IWA-4000.

DCD Tier 2, Section 6.6.6, “Evaluation of Examination Results,” discusses the evaluation of examination results. The applicant stated that the evaluation of nondestructive examination results for ASME Code Class 2 and 3 systems and components is in accordance with ASME Section XI, Articles IWC-3000 and IWD-3000. The DCD also states that when acceptance standards for a particular component or examination category are in preparation, the evaluation is based on the acceptance standards specified in the ASME Section III edition applicable to the construction of the component. Repair and replacement activities for ASME Code Class 2 and 3 components are in accordance with ASME Section XI, Article IWA-4000. Based on this method of evaluating examination results, and the use of the appropriate ASME Code rules for repair, the applicant’s evaluation of examination results for APR1400 components meets the requirements of ASME Section XI, and is therefore acceptable.

6.6.4.7 System Pressure Tests

The program provided in the DCD for Class 2 and 3 system pressure testing is acceptable if it meets the criteria of ASME Section XI, Articles IWC-5000 and IWD-5000. DCD Tier 2, Section 6.6.7, “System Pressure Test,” states that Class 2 systems subject to system pressure tests are tested in accordance with ASME Section XI, Articles IWA-5000 and IWC-5000, and Table IWC-2500-1. The DCD also states that Class 3 systems subject to system pressure tests are tested in accordance with ASME Section XI, Articles IWA-5000 and IWD-5000, and Table IWD-2500-1. The staff finds that the program for system pressure testing is acceptable because it meets the requirements of ASME Section XI.

6.6.4.8 Augmented IS to Protect against Postulated Piping Failure

The augmented ISI program for high-energy fluid system piping between CIVs is acceptable if:

- Access is provided in order to enable the performance of ISI examinations.

- During each inspection interval, 100 percent of the circumferential and longitudinal welds are examined with the boundary of the piping.

- Inspection ports are provided if access is restrained due to guard pipes.

- The areas subject to examination should be defined in accordance with ASME Section XI, Article IWC-2000, Examination Category C-F for Class 2 piping welds.

DCD Tier 2, Section 6.6.8, "Augmented IS Protect against Postulated Piping Failure," describes the augmented inspection program for high-energy fluid system piping between CIVs and provides the program criteria. The program includes the high-energy fluid systems described in DCD Tier 2, Section 3.6.1 and 3.6.2. Protective measures, structures, and guard pipes do not prevent the access required to conduct the required in-service examinations. The extent of examination completed during each inspection interval provides 100 percent volumetric examination of circumferential and longitudinal pipe welds. Also, the areas subject to examination are defined in accordance with ASME Section XI examination categories C-F-1 and C-F-2 for Class 2 piping welds. The staff finds that the augmented ISI program is acceptable because it incorporates provisions for accessibility, the appropriate examination categories, and 100 percent volumetric examination of the piping welds.

6.6.4.9 Relief Requests and Code Cases

Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," of this SER provides the staff's review of the programmatic aspects of the PSI and ISI program. As such, the staff's evaluation of relief from ASME Code requirements and Code Cases is documented in Sections 5.2.4.4.1.8 and 5.2.4.4.1.9 of this SER, respectively.

6.6.5 Combined License Information Items

Table 6.6-1 provides a list of inservice inspection and testing of ASME Code Class 2 and ASME Code Class 3 related COL information item numbers and descriptions from DCD Tier 2, Table 1.8-2:

Table 6.6-1 APR1400 Combined License Information Items

Item No.	Description	DCD Tier 2, Section
6.6(1)	The COL applicant is to identify the implementation milestones for ASME Section XI ISI program for ASME Section III Class 2 and 3 components	6.6
6.6(2)	The COL applicant is to identify the implementation milestone for the augmented ISI program.	6.6
6.6(3)	The COL applicant is to address the provisions to accessibility of Class 2 or 3 components for ISI if the design of the APR1400 Class 2 or 3 components is changed from the DCD design.	6.6
6.6(4)	The COL applicant is to provide the PSI and the ISI program.	6.6

The staff determines the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for ISI of Class 2 and 3 components consideration.

6.6.6 Conclusion

To ensure that no deleterious defects develop during service, ASME Code Class 2 system components, selected welds and weld heat-affected zones are inspected prior to reactor startup and periodically throughout the life of the plant. In addition, ASME Code Class 2 and 3 systems receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

The applicant has stated that the ISI program complies with the rules published in 10 CFR 50.55a and ASME Section XI, the 2007 Edition with the 2008 Addenda. The final ISI program is required to meet the latest ASME Code, Section XI Edition/Addenda incorporated by reference 12 months before the date scheduled for initial loading of fuel. The ISI program will consist of PSI and ISI plans. The staff finds that the description of the ISI program is acceptable and meets the inspection and pressure testing requirements of 10 CFR Part 50, Appendix A, GDC 36, GDC 37, GDC 39, GDC 40, GDC 42, GDC 43, GDC 45, and GDC 46, and 10 CFR 50.55a. The inservice testing of pumps, valves, and dynamic restraints is further discussed in Section 3.9.6 of this SER. The staff determined that the applicant's proposed revisions to the DCD have appropriately addressed the issue and are being tracked as **confirmatory items** pending the incorporation of confirmatory items **06.06-1, 06.06-2, and 06.06-3, into the next revision of the DCD**.