19.0 PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation," 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 19 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).

This chapter documents the staff's review of the probabilistic risk assessment (PRA) and severe accident evaluation (SAE) of DCD Tier 2 Revision 0 (APR1400-K-X-FS-14002 [Agencywide Documents Access and Management System (ADAMS) Accession No. ML15006A059]), Revision 1 (ML17096A324), Revision 2 (ML18079A383), and Revision 3 (ML18228A647). The review followed NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," hereafter also called SRP, specifically, Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors."

19.0.1 NRC Regulatory Requirements and Related Policies

This section of the SER applies to DCD Sections 19.0, 19.1, "Probabilistic Risk Assessment," and 19.2, "Severe Accident Evaluation."

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 52.47(a)(1) states that each DC application must include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters.

10 CFR 52.47(a)(2) states that it is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of radioactive fission products.

10 CFR 52.47(a)(4) states that each DC application must include an analysis and evaluation of the design and performance of structures, systems, and components (SSCs) with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

10 CFR 52.47(a)(8) states that a DC application must include the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island (TMI) requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

10 CFR 52.47(a)(23) states that a DC application for light-water reactor (LWR) designs must include a description and analysis of design features for the prevention and mitigation of severe accidents, (e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass).

10 CFR 52.47(a)(27) states that a DC application must include a description of the design-specific PRA and its results.

Several NRC policy statements describe the appropriate way to address SAs and use of the PRA:

- NRC Policy Statement on Severe Reactor Accidents - Regarding Future Designs and Existing Plants (50 FR 32138, dated August 8, 1985); specifically, Section B, "Policy for New Plant Applications"

- NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (51 FR 28044, dated August 4, 1986 as corrected and republished in 51 FR 30028, dated August 21, 1986)

- NRC Policy Statement on Nuclear Power Plant Standardization (52 FR 34884, dated September 15, 1987)

- NRC Policy Statement on Regulation of Advanced Nuclear Power Plants (59 FR 35461, dated July 12, 1994)

- NRC Policy Statement on Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (60 FR 42622, dated August 16, 1995)

Staff requirements memoranda (SRM) direct the staff's review of risk-related topics. Guidance for implementing features in new designs to prevent and mitigate the effects of severe accidents is found in these documents:

- SECY-90-016, "Evolutionary Light-Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements" (ML003707849), and the related SRM, dated June 26, 1990 (ML003707885)

- SECY-89-102, "Implementation of the Safety Goals" (ML17159A892), and the related SRM, dated June 15, 1990 (ML003707881)

- SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs" (ML003708021), and the related SRM, dated July 21, 1993 (ML003708056)

- SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design" (ML003708224), and the related SRM, dated January 15, 1997 (ML003708192)

- SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design" (ML003708316), and the related SRM, dated June 30, 1997 (ML003708232)

Guidance on application of PRA-based seismic margin analysis is also found in SECY-93-087 and the related SRM, dated July 21, 1993.

Guidance on the application of PRA standards and the use of peer review is found in SECY-00-0162, “Addressing PRA Quality in Risk-Informed Activities” (ML003732744), and the related SRM dated October 27, 2000 (ML003763875).

Guidance related to the scope, uses, and technical adequacy of the PRA are included in the following regulatory guides:

- Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis”

- RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”

Guidance related to the quality of PRA and quality assurance are also found in RG 1.174 (Sections 2.2.3 and 2.5).

Other regulatory guides address combustible gas during severe accidents such as:

- RG 1.7, “Control of Combustible Gas Concentrations in Containment”

- RG 1.216, “Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design-Basis Pressure”

The following NUREGs are used to support the staff’s review:

- NUREG-0800, Revision 2, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors”

- NUREG/CR-2300, “PRA Procedures Guide”

- NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants”

- NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities”

- NUREG/CR-6850, Supplement 1, “Fire Probabilistic Risk Assessment Methods Enhancement”

- NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants”

- NUREG/CR-5500, “Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998”

- NUREG/CR-5485, “Guidelines on Modeling Common Cause Failures in Probabilistic Risk Assessment”

- NUREG/CR-5497, “CCF Parameter Estimations, 2010 Update”

- NUREG/CR-4772, “Accident Sequence Evaluation Program Human Reliability Analysis Procedure”
- NUREG/CR-1278, “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plants”
- NUREG/CR-6890, “Reevaluation of Station Blackout Risk at Nuclear Power Plants”
- NUREG-1335, “Individual Plant Examination: Submittal Guidance”
- NUREG/CR-6365, "Steam Generator Tube Failures”
- NUREG/CR-6906, “Containment Integrity Research at Sandia National Laboratories - an Overview”
- NUREG/CR-4334, “An Approach to the Quantification of Seismic Margins in Nuclear Power Plants”
- NUREG/CR-4482, “Recommendations to the NRC on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants"
- NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines”
- NUREG-2169, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database”
- NUREG-1449, “Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States”
- NUREG/CR-1278, “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications”
- NUREG/CR-6883, “SPAR-H Human Reliability Analysis Method”
- NUREG/CR-7114, “A Framework for Low Power/Shutdown Fire PRA”
- NUREG-0737, “Clarification of TMI Action Plan Requirements”

DC and combined license (COL) interim staff guidance (ISG) are also used to evaluate the application, including:

- DC/COL-ISG-3, “Interim Staff Guidance, PRA Information to Support Design Certification and Combined License Applications”
- DC/COL-ISG-020, “Interim Staff Guidance on Implementation of a PRA-Based Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment”
- Digital I&C-ISG-03, “Risk-Informed Digital Instrumentation and Controls”

19.0.2 Structure of the APR1400 DCD Chapter 19

DCD Section 19.0 documents the regulatory requirements, structure of Chapter 19, and COL information.

DCD Section 19.1 describes the PRA, its results and COL information.

DCD Section 19.2 evaluates severe accidents, including an assessment of preventive and mitigative features. It describes containment performance, accident management, and COL information. The potential design improvements are also assessed.

DCD Section 19.3 addresses the APR1400 conformance with SECY-12-0025, “Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami,” and COL information.

DCD Section 19.4 identifies the APR1400 strategies that are implemented in the event that a large area of the facility is lost due to explosions or fire, and COL information.

DCD Section 19.5 describes the design features and functional capabilities of the APR1400 in response to the potential effects of the impact of a large commercial aircraft, and COL information.

19.0.3 Combined License Information

Summary of Application

DCD Section 19.0 contains one COL information item: COL 19.0(1).

> The COL applicant is either to confirm that the PRA in the DC bounds the site-specific design information and any design changes or departures, or to update the PRA to reflect the site-specific design information and any design changes or departures.

Technical Evaluation

The staff considers the above COL information item one of the most important actions during the COL application stage in developing the plant-specific PRA. The applicant established this COL information item in a manner consistent with 10 CFR 52.79(d)(1) and the guidance provided in the SRP Section 19.0 and RG 1.206, which states that if the COL application references a DC then the plant-specific PRA information should use the PRA information for the DC with any updates to account for site-specific design information and any design changes or departures.

The staff finds that the above COL information item was established in conformance with 10 CFR 52.79(d)(1), SRP Chapter 19, and RG 1.206 guidance, and therefore, it is acceptable.

19.1 Probabilistic Risk Assessment

The design-specific PRA developed for APR1400 includes a Level 1 and Level 2 PRA for internal and external events (i.e., internal flooding and internal fire) at full power, as well as during low power and shutdown (LPSD) conditions. A PRA-based seismic margin analysis (SMA) was performed to evaluate the impact on the design from a seismic event. Other external events were assessed using the qualitative/quantitative screening approach.

The applicant stated their assessments of internal events at full-power followed the guidance from NUREG/CR-2300, “PRA Procedures Guide,” and NUREG-1150, “Severe Accident Risks,” to the extent practicable and were performed consistent with consensus standards endorsed by the NRC in RG 1.200: ASME/ANS RA-Sa–2009, “Addenda to ASME/ANS RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (the PRA Standard).

DCD Tier 1:

There are no DCD Tier 1 entries for this area of review. However, the following DCD Tier 1 sections discuss systems designed to prevent or mitigate a SA:

- Section 2.4.1, “Reactor Coolant System”
- Section 2.4.2, “In-Containment Water Storage System”
- Section 2.4.3, “Safety Injection System”
- Section 2.4.4, “Shutdown Cooling System”
- Section 2.5.1, “Reactor Trip System and Engineered Safety Features Initiation”
- Section 2.5.2, “Diverse Actuation System”
- Section 2.5.3, “Qualified Indication and Alarm System”
- Section 2.5.4, “Engineered Safety Features-Component Control System”
- Section 2.6.1, “AC Electric Power Distribution System”
- Section 2.6.2, “Emergency Diesel Generator System”
- Section 2.6.3, “DC Power System”
- Section 2.6.4, “Instrumentation and Control Power System”
- Section 2.6.6, “Alternate AC Source”
- Section 2.7.1.5, “Auxiliary Feedwater System”
- Section 2.7.2.1, “Essential Service Water System”
- Section 2.7.2.2, “Component Cooling Water System”
- Section 2.7.2.3, “Essential Chilled Water System”
- Section 2.7.5.2, “Fire Protection System”
- Section 2.11.1, “Containment Structure”
- Section 2.11.2, “Containment Spray System”
- Section 2.11.4, “Containment Hydrogen Control System”

DCD Tier 2:

DCD Section 19.1 presents a description of the applicant’s PRA and its results.

19.1.1 Uses and Applications of the PRA

19.1.1.1 During Design Phase

Summary of Application

The applicant stated that the PRA is an integral part of the design process and is used to optimize the plant design with respect to safety. The APR1400 PRA is used to:

- influence the selection of design alternatives;
- provide input to other design areas, i.e., physical security (Section 13.6);
- identify inspections, tests, analyses, and acceptance criteria (ITAAC) (Section 14.3);
- inform technical specifications (TS) (Chapter 16);
- establish the scope of the reliability assurance program (RAP) (Section 17.4);
- provide input to human factors engineering (HFE) (Chapter 18); and
- support the SAE (Section 19.2).

Technical Evaluation

The staff's evaluation presented in this section focuses on the use of APR1400 design-specific PRA during the design phase. The staff's evaluation was in accordance with SRP Chapter 19, Section 19.0, Part I, which states that the purpose of the staff's review is to ensure that the applicant has adequately addressed the Commission's objectives regarding the appropriate way to address the use of PRA in the design of facilities under review.

As required by 10 CFR 52.79(d)(1) and 10 CFR 50.71(h), the staff expects that the APR1400 PRA is continued to be updated, improved, and useful throughout its life, from DC, through COL application, and until the end of the APR1400 nuclear power plant (NPP) operation. In preparing for the DC, the applicant has used the APR1400 PRA in a number of areas as follows.

- to estimate core damage frequency (CDF) and large release frequency (LRF)
- to assess risk significance
- to identify design vulnerabilities
- to develop design- and PRA-related insights
- to increase design safety by examining design options
- to determine the balance of prevention and mitigation capabilities
- to identify areas that should receive special attention under ITAAC
- to identify systems and components to be included in the RAP
- to identify human errors that need to be considered in advanced control room design
- to provide a structure for determining procedural and TS needs
- to provide risk insights for identifying vital equipment
- to provide PRA results for evaluating the potential costs and benefits of severe accident mitigation design alternatives.

Based on the information provided in the DCD, RAI responses, and on the PRA regulatory audit (ML18253A034), the staff finds that during the design stage, the applicant has effectively used the PRA to identify and assess design features for prevention and mitigation of severe accidents in conformance with the SRP Section 19.0. As described later in this report, the applicant identified design features that will reduce significant risk contributors. The applicant has appropriately used PRA to identify important operator actions. The applicant has demonstrated that the APR1400 design compares favorably against the Commission's goals of less than 1E-4/year for CDF and less than 1E-6/year for LRF described in the SRMs on SECY-90-16 and SECY-89-102. Likewise, the applicant demonstrated that the APR1400 design satisfies the Commission's containment performance goals.

The staff confirmed that the applicant has identified and described in its DCD Chapter 19 the robustness and levels of defense in depth of the APR1400 design, and how well this design would respond to the severe accidents. The applicant has integrated the PRA results including importance and uncertainty analyses, sensitivity studies, and assessments of the significant human errors to produce the risk insights. These insights address systems and components that contribute most to safety when they function as designed, as well as those that decrease safety most if they fail. Also, the applicant has characterized the initiating events that significantly contribute to the assessed risk.

Among the APR1400 PRA applications as identified above, with regard to the use of PRA in identifying the ITAAC, the regulation in 10 CFR 52.47(b)(1) requires, in part, that application for a DC includes the ITAAC necessary to demonstrate that the facility has been constructed and will be operated in conformity with the NRC regulations. In addition, SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria," clarifies that "If the results of the PRA indicate that a particular component or function of a system is risk significant, that component or function should be verified by ITAAC."

The staff noticed that APR1400 DCD Revision 0 does not describe the process used to develop the ITAAC from PRA results as described above. Therefore, to reach a conclusion that the application meets 10 CFR 52.47(b)(1) and establishes the appropriate ITAAC associated with the results of the PRA, RAI 434-8352, Question 19-87, was issued on March 8, 2016, in which the staff requested the applicant to (1) describe in detail how the APR1400 PRA was used in determining the scope of ITAAC, and (2) identify ITAAC that have been derived from the important PRA assumptions and insights (ML16068A187).

On September 12, 2017, the applicant responded to Question 19-87, stating that for the identification and development of APR1400 ITAAC, the top-level information (i.e., principal performance characteristics and safety functions of the SSCs) was selected for validation after considering the appropriate treatments of important insights and assumptions from the PRA. For the resulting design commitments, acceptance criteria were specifically developed to ensure the performance, physical condition, or analysis for these SSCs are satisfactorily demonstrated. The important insights and assumptions from the PRA were used to assess the appropriate top-level design features for inclusion in DCD Tier 1 (ML17255A951). On January 26, 2018, the applicant submitted the supplemental and final response to Question 19-87, showing the changes to DCD Section 14.3, specifically Table 14.3.4-2, as a result of the PRA update (ML18026A655). The staff reviewed Table 14.3.4-2, against the final PRA and finds it consistent with the PRA assumptions and insights. The staff confirmed that DCD Tier 2 Revision 2, was updated as shown in the RAI response. The staff finds that the applicant has provided adequate consideration of the ITAAC utilizing PRA results and insights, therefore RAI 434-8352, Question 19-87, is resolved and closed.

The staff's review confirmed that during the design stage, the applicant has properly and effectively used the design-specific PRA in support of APR1400 design as described in the Commission policy statement on the use of PRA methods in nuclear regulatory activities (60 FR 42622, dated August 16, 1995) and SRP Section 19.0. The APR1400 PRA was used to gain insights about the robustness of the design and its tolerance of severe accidents, and provide risk-informed input to pre- and during certification activities, thus achieving the Commission's objectives outlined in RG 1.206, Section C.I.19.2. The staff finds that the use of APR1400 PRA during design stage was reasonable and acceptable, and is in conformance with SRP Section 19.0.

19.1.1.2 During Combined License Application Phase

Summary of Application

DCD Section 19.1.1.2 does not address the uses of the PRA related to a specific COL application, but notes that the COL applicant should describe how it uses the PRA to support its programs, including any risk-informed applications proposed at the time of COL application.

Technical Evaluation

10 CFR 52.47(a)(27) requires that an application for a DC address the design-specific PRA and its results. 10 CFR 52.79(a)(46) requires an applicant for a COL address the plant-specific PRA. SRP Section 19.0, Part I, also provides that it is acceptable to not address in the DCD, as part of the DC application, the uses of the PRA relevant to a specific COL application. The applicant established COL Information Item, COL 19.1(1):

> The COL applicant is to describe the uses of PRA in support of licensee programs, and to identify and describe risk-informed applications being implemented during the combined license application phase.

The staff finds that the above COL information item will enable the staff to assess the uses of the PRA during COL application phase, consistent with 10 CFR 52.79(a)(46) and the guidance in SRP Chapter 19 and RG 1.206, and therefore, it is acceptable.

19.1.1.3 During Construction Phase

Summary of Application

DCD Section 19.1.1.3 does not address the uses of the PRA during the construction phase, but notes that the licensee should describe how PRA is used during construction and in support of any risk-informed applications proposed for implementation during this phase.

Technical Evaluation

10 CFR 52.47(a)(27) requires that an application for a DC address the design-specific PRA and its results. 10 CFR 52.79(a)(46) requires an applicant for a COL address the plant-specific PRA. SRP Section 19.0, Part I also provides that it is acceptable to not address in the DCD, as part of the DC application, the uses of the PRA relevant to a specific COL application during construction phase. The applicant established COL Information Item, COL 19.1(2):

> The COL applicant is to describe the uses of PRA in support of licensee programs, and identify and describe risk-informed applications being implemented during the construction phase.

The staff finds that the above COL information item will enable the staff to assess the uses of the PRA during the construction phase, consistent with 10 CFR 52.79(a)(46), the SRP and RG 1.206 guidance, and therefore, it is acceptable.

19.1.1.4 During Operational Phase

Summary of Application

DCD Section 19.1.1.4 does not address the uses of the PRA during the operational phase, but notes that the licensee should describe how the PRA is used during operation, how the PRA is used to inform operational programs (e.g., the maintenance program and the maintenance rule program), and how it supports any risk-informed applications proposed for implementation when operations begin.

Technical Evaluation

10 CFR 52.47(a)(27) requires that an application for a DC address the design-specific PRA and its results. 10 CFR 52.79(a)(46) requires an applicant for a COL address the plant-specific PRA. SRP Section 19.0, Part I, also provides that it is acceptable to not address in the DCD, as part of the DC application, the uses of the PRA during operation of APR1400 power plant. The applicant established COL Information Item, COL 19.1(3):

> The COL applicant is to describe the uses of PRA in support of licensee programs, and identify and describe risk-informed applications being implemented during the operational phase.

The staff finds that the above COL information item will enable the staff to assess the uses of the PRA during the operational phase, consistent with 10 CFR 52.79(a)(46), the SRP, and RG 1.206 guidance, and therefore, it is acceptable.

19.1.2 Quality of PRA

In DCD Section 19.1.2, the applicant identifies the attributes of the PRA that make it suitable for use in the design process and to support DC. Methods to achieve and maintain appropriate quality are described as follows:

- Use of qualified personnel
- Use of procedures to control documentation
- Use of procedures to control corrective actions.

The staff's review of the PRA scope, level of detail, standards and guidance, and PRA maintenance are described in greater detail below.

19.1.2.1 PRA Scope

Summary of Application

The application stated that the APR1400 PRA consists of an assessment of the potential for accidents (core damage), an assessment of the containment response to these accidents, and characterization of the magnitude and frequencies of radionuclide releases. The applicant performed a Level 1 PRA for internal initiating events (IEs), internal floods, and internal fires. The applicant quantified risk at power and in all other operating modes. The at-power internal events PRA includes the following elements:

- Initiating event analysis,
- Accident sequence analysis,
- Success criteria analysis,
- System analysis (including system dependencies),
- Data analysis and common cause analysis,
- Human reliability analysis, and
- Quantification.

For each accident sequence in the model, the applicant calculated the CDF and assigned an associated plant damage state (PDS).

The applicant performed a Level 2 PRA, analyzing each PDS to determine the associated containment response and to characterize the magnitude and frequencies of radionuclide releases. From these results, the applicant calculated LRF.

Other analyses of hazards included the seismic event and other external events. Not all external hazards were evaluated in detail. Several external events have been screened from the assessment during the design phase, mainly on a qualitative basis.

Evaluation of internal fire events was based on the methodology described in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities." It reflected bounding assumptions and was refined to avoid masking risk from other sources. The evaluation of internal flooding used the approach described in the PRA Standard. Although a seismic PRA would allow the estimations of CDF and LRF during the design stage, without a specific site and details of construction, a seismic PRA cannot be performed. Instead, to demonstrate adequate seismic capacity, the applicant used a PRA-based SMA as discussed in SECY-93-087 and DC/COL-ISG-020.

Technical Evaluation

As discussed in RG 1.200, the scope of a PRA is defined in the following terms:

1. Metrics used in characterizing the risk, expressed by CDF and LRF,
2. Plant operating states for which the risk is to be evaluated, and
3. Causes of initiating events (hazard groups) that can potentially challenge and disrupt the normal operation of the plant.

The staff's review finds that the scope of APR1400 PRA described in DCD Chapter 19.1 fully met all terms itemized above as discussed in the following sections of this report. The staff finds that the scope of APR1400 PRA is consistent with the expected scope for a design-specific PRA as described in SRP Chapter 19. The APR1400 risk assessment is comprehensive in scope and addresses all essential and applicable internal and external events for all operating modes. Additional details can be found in the following sections of this report.

The staff finds that the scope of the APR1400 risk assessment is considered to be sufficient and acceptable for a DC application and for producing the risk insights during DC stage because the scope of APR1400 PRA is consistent with applicable guidance, including the consensus PRA Standard, the guidance outlined in SRP Chapter 19, and the guidance in RG 1.206, Section C.I.19.

19.1.2.2 PRA Level of Detail

Summary of Application

The applicant asserts that the PRA reflects as much detail as possible. Some detailed design information is not available at this stage (e.g., the location of equipment within a room, the routing of power and control cables or piping). Operating procedures and other supporting information are not yet developed, and some reliability data may be limited.

In compliance with 10 CFR 52.79(d)(1), a COL applicant referencing the APR1400 DC is required to confirm that the assumptions used in the PRA remain valid. Internal events and external events must be considered for all modes of operations. Walkdowns will be performed as necessary (e.g., for PRA-based SMA, internal floods, and internal fires). PRA input to other programs (e.g., RAP) and other analyses are to be verified, such as severe accident mitigation design alternatives (SAMDA) and HFE (i.e., development of normal operating procedures, emergency operating procedures, and training).

Technical Evaluation

DCD Section 19.1.2.2, "PRA Level of detail," identifies the elements of the detailed design that are currently unavailable to support the design-specific PRA development. Due to these limitations, the APR1400 PRA was developed with various assumptions and some limited information during DC stage. In offsetting, the applicant followed the SRP Chapter 19 guidance by providing justification as to why the APR1400 PRA is still suitable to support its DC application without some detailed design information. SRP Section 19.0 states that if detailed design information is not available or if it can be shown that detailed modeling does not provide additional significant information, it is acceptable to make bounding-type assumptions consistent with the guidelines in RG 1.200. Accordingly, the applicant has made reasonable bounding assumptions in its PRA in conformance with the SRP Chapter 19, and RG 1.200. The staff's evaluations of the PRA assumptions and the potential impact on the PRA caused by the lack of detailed design information (e.g., cable routing, pipe routing) are documented in Section 19.1.2.3 and other corresponding sections of this report.

The applicant established the following COL 19.1(4) to ensure that the plant-specific PRA will be realistically developed reflecting the actual plant design, planned construction, and anticipated operational practices.

> The COL applicant and holder are to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA (including PRA inputs to RAP and SAMDA) remain valid with respect to internal events, internal flood and fire events (fire barrier and fire barrier penetrations, routings and locations of pipe, cable, and conduit), and HRA analyses (development of operating procedures, emergency operating procedures, and severe accident management guidelines and training), external events including PRA-based seismic margins and [high confidence of low probability of failures] HCLPF fragilities, Seismic spatial interactions, and LPSD procedures.

Based on the technical information provided in the DCD and audited in the electronic reading room (ERR), and as further evaluated in the following sections of this report, the staff finds that the level of detail of the APR1400 PRA is consistent with the specificity described in SRP Chapter 19. This level of detail is commensurate with the uses of the PRA, and sufficient to gain risk insights, in conjunction with the acceptable assumptions made in the PRA at the DC stage. The staff concludes that the APR1400 PRA reasonably reflects the actual plant design, to the extent possible.

19.1.2.3 PRA Technical Adequacy

Summary of Application

The applicant states that the technical adequacy of the PRA is consistent with the guidance in the ASME/ANS PRA Standard and RG 1.200, subject to the constraints of the design phase.

Technical Evaluation

The staff's evaluation of APR1400 PRA technical adequacy presented in this section followed the guidance provided in the SRP Section 19.0, RG 1.174, and RG 1.200. SRP Section 19.0 states that DC applicants should justify why their PRAs are adequate in terms of scope, level of detail, and technical adequacy. Specifically, the applicant should justify why the PRA approach, methods, and data, as well as the requisite level of detail necessary for the staff's review and assessment, are appropriate for the intended uses. RG 1.174 indicates that one acceptable approach that could be used to assess PRA technical acceptability is to perform a peer review of the PRA. Pertaining to the peer review, SRP Section 19.0, Part II, "Acceptance Criteria," Item 9E states:

> Peer review of the DC PRA is not required prior to application. However, if a peer review was conducted prior to the application; the staff should examine the peer review report. If a certain aspect of the PRA deviates from accepted good practices, the applicant/holder should justify that this deficiency does not impact the PRA results or risk insights.

To justify why the APR1400 PRA is adequate in terms of technical acceptability, the applicant conducted an independent peer review against the ASME/ANS PRA Standard and used the peer review outcome to demonstrate that the APR1400 PRA technical quality is sufficient to support its DC application. The applicant uploaded the peer review report, which includes the peer review findings and suggestions, to the ERR for staff examination. In addition, the applicant provided the resolutions for dispositioning these findings, and made available the relevant technical rationale in the ERR for staff audit.

In making its determination on APR1400 PRA technical adequacy, the staff takes into consideration: (1) the information provided in DCD Chapter 19, (2) responses to the staff RAIs, (3) responses to staff's observations during the PRA audit, and (4) peer review results and resolutions. The staff's evaluation provided in this section only scrutinizes the overall PRA technical adequacy and results from the peer review. Additional evaluation on APR1400 PRA technical adequacy can be found later in the pertinent sections of this report by the specific cross-referenced technical areas.

The staff examined the peer review report posted in the applicant's ERR and observed that the APR1400 PRA peer review was conducted against the PRA Standard, ASME/ANS RA-Sa-2009, Sections 2 and 3, during the week of June 24 through June 28, 2013. The peer review was performed using the process defined in Nuclear Energy Institute (NEI) 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," as referenced in RG 1.200. The review was conducted by a team of six PRA experts with over 170 years of combined experience in the risk and reliability fields. All members of this team have had no previous involvement in the APR1400 PRA development.

The staff noted that the scope of this peer review included at-power internal events Level 1 PRA, at-power internal flooding Level 1 PRA, and the large release frequency portion of Level 2 PRA. The peer review report concluded that the APR1400 PRA substantially meets both the ASME/ANS PRA Standard and the draft Advanced Light Water Reactor (ALWR) Standard at Capability Category II or better for 88% of the applicable supporting requirements (SRs), with 90 percent met at Capability Category I or better. This peer review identified ninety "Findings and Observations (F&Os)," twenty-seven "Suggestions," fifty-nine "Findings," and four "Best Practices." The peer review also stated that, overall, the APR1400 PRA was found to substantially meet the ASME/ANS PRA Standard at Capability Category II.

In more detail, the peer review report states that of the 327 SRs contained in the ASME/ANS PRA Standard, forty-nine SRs were determined to be not applicable to the APR1400 PRA. Of the 278 remaining SRs, 251 SRs met at least Capability Category I, and twenty-seven SRs were rated as not met. The report further states that the IEs analysis identified a complete set of industry generic IEs. The IEs analysis properly grouped IEs with similar mitigation requirements using a systematic and structured approach.

The peer review report also says that the APR1400 accident sequence (AS) analysis fully described the specific scenarios that could lead to core damage. These scenarios addressed both system responses and operator actions. The peer review identified numerous inconsistencies between the mitigating systems documented in the AS notebook and those modeled in the event trees (ETs). However, the success criteria development was found to be well documented. The definition of core damage used was consistent with the definition in the standard. The system analysis was well performed within the constraints of modeling a design.

Regarding the human reliability analysis (HRA), the peer review report notes that the APR1400 design is in the pre-operational phase and a lot of the key physical arrangement information and procedural information needed for a detailed HRA are not currently available. From a process perspective, both the APR1400 pre-initiator and post-initiator HRAs largely satisfied the requirements of the standard. The most important issues are the "talk through" associated with risk significant operator actions and timing information, both of which are not available at this time. The definitions of component boundaries were found to be well established and consistent with the generic data sources.

The peer review report states that, overall, the APR1400 Level 2 analysis was well performed. The PDSs and containment event trees (CETs) were developed using standard Level 2 methodologies and incorporated the design-specific features that impacted severe accident progression. The CETs account for the various phenomenological processes, containment conditions, and containment failure modes that could occur under severe accident conditions. The containment structural analysis was sufficiently detailed in identifying the potential failure locations and the failure pressures and to support linkage of pressure challenges to failure times, locations, and modes.

The staff assessed the information documented in the peer review report against DCD Section 19.1.2, “Quality of PRA” and DCD Table 19.1-1 “Characterization of PRA Relative to Supporting Requirements in ASME PRA Standard.” The staff identified several discrepancies between these two documents. The staff further noticed that the peer review findings had not been completely dispositioned prior to submission of the DC application to the NRC and thus the impacts have not been totally captured. Therefore, in order for the staff to reach its conclusion regarding the APR1400 PRA technical adequacy, on March 8, 2016, in RAI 434-8352, Question 19-88, the staff requested the applicant to address the staff’s concerns and to revise the DCD accordingly (ML16068A187).

On September 28, 2017, the applicant responded to RAI 434-8352, Question 19-88 (ML17271A485). In its response, the applicant clarified that Table 19.1-1 is intended to say that most aspects of the PRA elements satisfy ASME/ANS PRA Standard Capability Category I or greater and decided to revise the DCD to reflect this position. In addressing other items in the RAI question, the applicant referenced APR1400-K-P-NR-013710-P, “APR1400 DC Full Power Internal Events and Internal Flooding Level 1 PRA - Resolution Evaluations of the Facts and Observations from Peer Review,” and indicated that the justifications for those SRs that cannot meet or does not meet at least Capability Category I and the potential impacts on the PRA are documented in this report. The report also specified the resolution of each finding identified by the peer review and the associated impacts of these findings on APR1400 DC application. The applicant made this report and the following documents available in the ERR for staff’s audit:

- APR1400-K-P-NR-013711, R0 - At-Power & LPSD Fire PRA Gap Analysis
- APR1400-K-P-NR-013712, R0 - At-Power & LPSD Level 2 PRA Gap Analysis
- APR1400-K-P-NR-013713, R0 - LPSD Internal Events PRA Gap Analysis
- APR1400-K-P-NR-013714, R0 - LPSD Internal Flooding PRA Gap Analysis.

The staff provided its observations on these documents during the PRA audit. Based on these observations, the applicant significantly revised DCD Table 19.1-1 consistently reflecting the status of APR1400 PRA quality as characterized by the peer review and staff's findings. The staff reviewed the proposed revised Table 19.1-1 and found no further issues with the characterization of APR1400 PRA quality relative to the SRs in ASME/ANS PRA Standard. More details and staff assessments on the at-power internal events PRA peer review, at-power internal flooding PRA peer review, at-power and LPSD internal fire PRA gap analysis, at-power and LPSD Level 2 PRA gap analysis, LPSD internal events PRA gap analysis, and LPSD internal flooding PRA gap analysis are provided later in the applicable sections of this evaluation. The staff finds that DCD Table 19.1-1 reasonably characterizes the APR1400 PRA technical adequacy, except the Quantification SRs QU-B7 and QU-B8 as discussed in Section 19.1.4.1 of this report. The staff confirmed that DCD Tier 2 Revision 2, was revised as committed in the RAI response. Accordingly, the staff considers RAI 434-8352, Question 19-88, resolved and closed.

Although the staff has had an opportunity to examine the peer review report and the peer review resolution evaluations report, the staff did not rely solely on this peer review for making conclusion regarding the quality and acceptability of APR1400 PRA. Through examining the peer review report, the staff obtained additional informative insights of the degree to which the APR1400 PRA has been assigned to the ASME/ANS PRA Standard capability categories. The peer review findings and suggestions were valuable when blending with RAIs to strengthen the staff's confidence in the PRA model and results. The staff used the APR1400 peer review as a starting point to identify the strengths and weaknesses with the PRA. The peer review greatly assisted the staff in its in-depth review, particularly by reducing the number of RAIs needed. Its results have been used to provide the staff an added level of confidence in the APR1400 PRA results and insights.

To ensure that the APR1400 PRA is sufficient to be used to support the risk-informed decision making, the applicant established COL 19.1(5), which states:

> The COL applicant and/or holder is to conduct a peer review of the PRA relative to the industry PRA Standard prior to use of the PRA to support risk-informed applications, as applicable.

SRP Chapter 19, RG 1.206, and RG 1.200 provide staff the detailed guidance on how to establish an acceptable approach for demonstrating the technical adequacy of a PRA used to support a regulatory application and to assess the technical adequacy needed for an application. The staff finds that the above COL information item has been established in a manner consistent with this guidance, and therefore, it is acceptable.

As discussed above and in other sections of this report, the staff finds that the APR1400 PRA is of sufficient quality, which can be used in the following ways.

- To assess the risks associated with the APR1400 design
- To identify strengths and weaknesses of APR1400 design features
- To evaluate APR1400 containment failure probabilities
- To compare the APR1400 risk results with the Commission's goals

- To provide an integrated perspective of the overall risk estimates
- To identify major contributors to uncertainty in estimated CDF and LRF
- To identify important contributors to plant risk.

The staff concludes that the quality of APR1400 PRA is sufficient for its intended functions such as supporting and improving the APR1400 design process, providing relative importance of sequences and SSC leading to core damage or containment failure, searching for design vulnerabilities, and identifying the PRA insights to support other DCD chapters.

19.1.2.4 PRA Maintenance and Upgrade

Summary of Application

The applicant states that the PRA model and supporting documentation will be maintained so that they continue to reflect the as-designed characteristics of the plant. A process is in place to:

- Monitor PRA inputs and collect any new information relevant to the PRA
- Maintain and upgrade the PRA to be consistent with the design
- Consider cumulative impacts of pending changes when applying the PRA
- Consider impacts of changes for previously implemented risk-informed decisions that used the PRA (e.g., RAP)
- Maintain configuration control of the computational methods used to support the PRA
- Document the PRA model and processes.

Technical Evaluation

The staff's evaluation presented in this section focused on the APR1400 PRA maintenance and upgrade program described in DCD Section 19.1.2.4, with respect to the guidance contained in the SRP. SRP Section 19.0, Part II, "Acceptance Criteria," Item 18 states:

> PRA maintenance should commence at the time of application for both DC and COL applicants. Once the certification is issued, the generic PRA would not need to be updated except as appropriate in connection with a DC amendment request.

The staff reviewed DCD Section 19.1.2.4 and found this section did not provide sufficient information to determine whether the PRA maintenance and update procedures are appropriately developed and the process is technically sound and implemented. Therefore, to verify that the APR1400 PRA will continue to reflect the certified plant design and be suitable for its identified uses, on March 8, 2016, in RAI 434-8352, Question 19-90, the staff requested that the applicant describe the process for monitoring PRA inputs and tracking the issues/findings (e.g., design changes, peer review findings, staff review findings, model errors) for which a PRA update is needed, and also the process for upgrading and maintaining the PRA to be consistent with the certified design (ML16068A187).

In the response, on August 8, 2016, to Question 19-90 (ML16253A272), the applicant stated that the resolution of review findings and other model issues is controlled by the APR1400 procedure “Risk Management Engineering Peer Reviews, Independent Reviews and Self Assessments.” The identified findings are entered into an appropriate tracking database for timely resolution. Minor items are to be resolved by periodic updates, nominally every four years; while significant issues are to be addressed immediately. With regard to the PRA maintenance and upgrade, the “Risk Management Engineering Procedure” ensures that the scope, level of detail, and capability of each model is commensurate with its intended use. The “Risk Management Engineering Configuration Control” procedure monitors and controls the PRA model ensuring its conformance with the ASME/ANS PRA Standards and RG 1.200. PRA analyst qualifications are controlled by the “Risk Management Engineering Training and Certification” procedure. The use of the PRA model is controlled by the “Risk Management Procedure, DC-DG-03-24” procedure, which ensures its compliance with the SRP Chapter 19. The applicant uploaded the above procedures into the ERR for staff audit. The applicant committed to revise the DCD to clarify the PRA maintenance and update process as clarified in the response.

The staff reviewed the response and audited the procedures posted in the ERR and finds that the process developed for maintaining and updating APR1400 PRA is practical, established in a manner consistent with the SRP Chapter 19 and ASME/ANS PRA Standards. The staff confirmed that DCD Tier 2 Revision 2, was revised as committed in the response to Question 19-90. Therefore, RAI 434-8352, Question 19-90, is resolved and closed.

In conformance with the guidance provided in the SRP and in RG 1.206 on the plant-specific PRA maintenance, the applicant established COL 19.1(6), which states:

> The COL applicant is to describe the PRA maintenance and upgrade program.

This COL information item is consistent with SRP Section 19.0, Part II, which states that the COL applicant is responsible for its plant-specific PRA maintenance and upgrade program and to clearly describe them in its application, therefore acceptable.

One of the major activities performed by the applicant during the DC safety review was the self-initiated conversion of APR1400 PRA software platform, which was used to develop its PRA. Although switching of the software would not impact the PRA results and insights, it may introduce errors to the PRA model caused by mistaken performance of the analysts. The staff has spent extra effort reviewing the top cutsets to ensure that changing of the software would not produce errors and inappropriate results. During the June 25, 2015, public conference call between the NRC and the applicant, the applicant informed the staff that the APR1400 PRA would be converted from SAREX computer code to another software, named CAFTA, to combine all APR1400 PRA models under a single top logic and to improve the quantification time. On March 8, 2016, in RAI 434-8352, Question 19-91, the staff requested the applicant to identify any resulting changes to the APR1400 PRA and DCD due to the conversion (ML16068A187).

On September 1, 2017, the applicant responded to RAI 434-8352, Question 19-91 (ML17244A645), stating that the CAFTA conversion had been completed for the APR1400 DC PRA models, including at-power IEs Levels 1 and 2 PRA, at-power internal fire Levels 1 and 2 PRA, at-power internal flooding Level 1 PRA, and LPSD IEs Level 1 PRA, and documented in the following PRA conversion notebooks:

- Full Power Internal Events Level 1 PRA (APR1400-K-P-NR-013230, Rev. 0)
- Full Power Internal Events Level 2 PRA (APR1400-K-P-NR-013231, Rev. 0)
- Full Power Fire Probabilistic Risk Assessment Level 1 PRA (APR1400-K-PNR-013440, Rev. 0)
- Full Power Fire Probabilistic Risk Assessment Level 2 PRA (APR1400-K-PNR-013441, Rev. 0)
- Full Power Internal Flooding (FP-IF) Level 1 PRA (APR1400-K-P-NR-013506, Rev. 0)
- Low Power and Shutdown Internal Events Level 1 PRA (APR1400-K-P-NR-013708, Rev. 0).

The applicant also clarified that the LPSD IEs Level 2 PRA, LPSD internal fire Levels 1 and 2 PRA, and LPSD internal flooding Level 1 PRA were originally developed using CAFTA therefore, the conversion for these PRAs was not needed.

The applicant made the above documents available in the ERR for staff's audit. The staff's noticed that the model conversion consisted of the following steps: (1) conversion of the SAREX event trees into a top logic model within CAFTA; (2) conversion of the SAREX system fault trees into CAFTA system fault trees and linking these system fault trees to the top logic model; and (3) conversion of the SAREX reliability database to a CAFTA format. When completed, the applicant performed a comprehensive review including a review of the top cutsets, non-risk significant cutsets, random set of middle cutsets, and cutsets at the end of the list.

The applicant affirmed in its comparison report that the APR1400 CAFTA PRA model produces correct results. Thousands of cutsets were reviewed and none were determined to be invalid with respect to the original model. During the conversion process, the applicant identified some errors and fixed them. The differences in the results were justified and documented in the comparison reports.

The applicant concluded that, based upon both high-level and detailed level comparisons of both cutsets and key importances, the CAFTA model is deemed to be at least as accurate as the SAREX model in all important aspects, and an improvement in certain respects due to corrected errors.

The staff's audit finds that the conversion of APR1400 PRA software was properly performed and the process and results were well documented. The conversion reports clearly described all the steps performed during the conversion in a very detailed manner. After the conversion, the verification was systematically and thoroughly conducted by the applicant ensuring that the CAFTA PRA model produces correct results. The new quantification results were fully recorded in the comparison reports. The staff finds that the applicant's conclusion above is reasonable and acceptable.

Based on the audit of the process applied by the applicant to switch the PRA software and the conversion reports, the staff finds no issues with the APR1400 PRA software conversion and does not identify any need to issue follow-up RAI; therefore RAI 434-8352, Question 19-91, is resolved and closed.

In conformance with the SRP Chapter 19, the applicant documented the key PRA assumptions and risk insights in its DCD Table 19.1-4, "Risk Insights and Key Assumptions." The staff's review finds this table incomprehensive in identifying the APR1400 PRA-related key assumptions and insights. Therefore, on March 8, 2016, in RAI 434-8352, Question 19-92, the staff requested that the applicant enhance and update Table 19.1-4 to identify all PRA key assumptions and PRA-based insights, and also the insights gained from the importance, sensitivity, and uncertainty analyses (ML16068A187). In the response to Question 19-92, dated August 3, 2016, the applicant committed to reassess Table 19.1-4 for additional important assumptions and insights (ML16222A925). During the course of the review of APR1400 PRA, both the staff and applicant treated Question 19-92, as an umbrella RAI for tracking those changes relevant to the PRA, including the PRA update. Hence, as discussed later in this report, several RAI questions have been subsumed to Question 19-92 for tracking purposes. Sequentially, on August 6, 2018, the applicant submitted its final response to Question 19-92 (ML18218A196), showing the changes. The staff's review and its basis for accepting of the response to Question 19-92, are provided in the following sections of this report. Based on the review of the DCD, the staff has confirmed incorporation of all changes tracked by RAI 434-8352, Question 19-92; therefore, RAI 434-8352, Question 19-92, is resolved and closed.

19.1.3 Special Design/Operational Features

The applicant states that the following design and operational enhancements on the APR1400 design offer improved plant safety, when compared to nuclear power plants designed with LWR technology that was current circa 1985.

- in-containment refueling water storage tank
- four-train safety injection system that injects borated water directly into the reactor vessel through direct vessel injection nozzles
- four pumps for component cooling water and essential service water systems
- emergency containment spray backup system
- cavity flooding system
- Hydrogen control system.

PRA influenced the selection of the following other design features.

- four emergency diesel generators
- a GTG as an alternate alternating current (AAC) source.

The evaluation of these features is discussed further in the following section.

19.1.3.1 Design/Operational Features for Preventing Core Damage

Summary of Application

Preventive features that are intended to minimize initiation of plant transients, to mitigate the progression of plant transients, and to prevent severe accidents, include the following safety systems and components.

- safety injection system
- shutdown cooling system
- containment spray system
- pilot-operated safety relief valves
- auxiliary feed system
- chemical and volume control system
- reactor protection system
- engineered safety features actuation system
- emergency diesel generators
- AAC power system
- direct current (dc) power system
- condensate and feedwater system
- main steam system
- essential service water system
- component cooling water system
- essential chilled water system
- heating, ventilation, and air conditioning systems
- instrument air system
- reactor coolant gas vent system
- in-containment refueling water storage tank
- digital control room
- auxiliary building.

19.1.3.2 Design/Operational Features for Mitigating the Consequences of Core Damage and Preventing Releases from Containment

<u>Summary of Application</u>

The applicant identifies containment features, mitigating systems, and human actions that are provided to mitigate the consequences of a core damage event and to prevent containment failure. These systems, identified below, mitigate the consequences of core damage and prevent containment failure.

- containment isolation system
- containment spray system
- pilot-operated safety relief valves
- containment hydrogen control system
- external reactor vessel cooling
- cavity flooding system.

The applicant identifies the following key preventive features as intended to reduce the potential for releases from containment.

- A large, prestressed concrete containment structure
- A steel lining of the containment to promote leak tightness
- A reactor cavity configured to promote retention of core debris during a severe accident

- A reactor cavity configured to promote spreading of core debris within the reactor cavity
- A reactor cavity designed with adequate distance between the floor elevation and the embedded portion of the containment steel liner to delay core melt debris contact with the liner in core melt scenarios.

Technical Evaluation

The staff reviewed the information on the APR1400 design and operational features for mitigating the consequences of core damage and preventing releases from containment as provided by the applicant and summarized herein using the guidance provided in SRP Chapter 19, SECY-90-016, SECY-93-087, and corresponding SRMs.

The DCD states that the APR1400 cavity design provides core melt spreading area of 0.02 m^2/megawatt-thermal (MWt) and a CFS to provide water to cool debris in the cavity and that the containment liner plate in the reactor cavity area is embedded 0.91 meters (m) (3 feet (ft)) below from the cavity floor at the minimum. SECY-90-016 states the following on “Core-Concrete Interaction - Ability to Cool Core Debris”:

> The EPRI [Utility] Requirements Document [(URD)] contains a number of design features that are intended to mitigate the effects of a molten core. To promote long-term debris coolability, the Requirements Document states that the cavity floor should be sized to provide 0.02 [square meters per mega-watt thermal] (m^2/MWt). The Requirements Document also specifies that the containment should be designed to ensure adequate water supply to the floor and that an alternate means of introducing water into the containment, independent of normal and emergency ac power, should be provided. Passive schemes for providing flooding of the floor areas beneath the vessel are proposed and described in general terms for both BWRs and PWRs. The Requirements Document also states that the steel shell or liner of the containment should be protected from core debris by at least 3 feet of concrete.

SECY-93-087 later states that “[t]he staff neither supports nor disputes the EPRI floor sizing criteria of 0.02 m^2/MWt.”

The APR1400 plant has a capacity of 1400 megawatt-electric (MWe), which is 4 percent higher than the largest plant for which EPRI URD guidance is applicable (i.e., 1350 MWe). However, it would not affect the cavity floor sizing criteria, which is defined per unit power, resulting in a larger floor area for large plants.

The staff notes that the floor area provided in the APR1400 design, in conjunction with the CFS, will promote debris coolability (via debris spreading, quenching by preexisting water in the cavity, and long-term heat removal by the overlying water pool) but will not necessarily ensure it. Accordingly, the staff has relied on the deterministic calculations described below (Section 19.2.3.3), rather than the EPRI criterion, in judging the adequacy of the reactor cavity design for MCCI.

A containment hydrogen control system provides 8 ignitors and 30 passive autocatalytic recombiners located strategically in the containment to control hydrogen in the containment.

SECY-90-016 provides the following on high-pressure core melt ejection:

> To limit direct containment heating, the ALWR Requirements Document states that the cavity/pedestal/drywell configuration should be designed to preclude entrainment of core debris by gases ejected from a failed reactor vessel. It also states that a safety-grade RCS safety depressurization and vent system (SDVS) will be provided. The staff's review has concluded that reactor vessel depressurization capability and cavity design features to entrap ejected core debris constitute an acceptable approach to the issue of high-pressure core melt ejection.

Consistent with SECY-90-016, pilot-operated safety relief valves (POSRVs) provide means of mitigating the consequences of high-pressure core damage scenarios that may lead to high-pressure core melt ejection.

The staff finds the applicant has identified containment features, mitigating systems, and human actions that are available to prevent accidents and mitigate their consequences consistent with guidance provided in SRP Chapter 19, SECY-90-016, and corresponding SRM.

19.1.3.3 Design/Operational Features for Mitigating the Consequences of Releases from Containment

Summary of Application

The applicant identifies the following containment features, mitigating systems, and human actions that are provided to mitigate the consequences of releases from containment. These systems minimize dose and consequences:

- Containment spray system (CSS): In the event of a containment leakage, CSS provides fission product scrubbing to mitigate consequences. The spray solution, mixed with trisodium phosphate, minimizes the iodine radionuclides and fission product aerosols in the building atmosphere, by removing them through the absorption of airborne fission products by the spray droplets.
- In-containment refueling water storage tank (IRWST): The IRWST minimizes spread of radioactive contamination outside the containment building, where the potential contamination from circulated water through the piping located outside the containment is minimized. The IRWST is also equipped with the underwater spargers to promote fission product scrubbing, where the fluids discharged through POSRVs are discharged through spargers.

Technical Evaluation

The staff reviewed information provided in the APR1400 DCD and agrees that the CSS and IRWST provide the capability to minimize offsite doses and consequences in the event of a containment leakage during a severe accident by retaining a part of radioactive isotopes inside the containment in compliance with the requirements of 10 CFR 52.47(a)(23).

19.1.3.4 Uses of the PRA in the Design Process

Summary of Application

The application states that the APR1400 PRA was used in the design process to achieve the following objectives:

- To identify features and requirements introduced to reduce or eliminate the known weakness and vulnerabilities in current reactor designs
- To indicate the effect of new design features and operational strategies on plant risk
- To identify PRA-based insights and assumptions used to develop design requirements.

The applicant investigated design improvements to reduce or to eliminate weaknesses in earlier designs, and adopted the following improvements.

- It increased the number of emergency diesel generators from two to four
- It extended 125 Vdc battery life from 8 hours to 16 hours.

The application describes how the design properly balances preventive and mitigative features.

Technical Evaluation

The staff reviewed the APR1400 PRA results and insights, including the importance ranking analysis and sensitivity analysis, and observed that during the design process the APR1400 PRA has played an important role in identifying features that merit consideration with respect to opportunities to reduce risk, and in evaluating the potential risk impacts due to some aspects of the design. As documented in DCD Section 19.1.4, the applicant identified risk-informed insights and specific human errors that are considered to be risk significant events. The PRA insights and importance ranking (i.e., significant SSCs, operator actions, and common cause events) have been used to identify specific improvements to reduce the contribution to risk and to support associated design decisions, (e.g., the design of emergency diesel generators (EDGs) and battery life as discussed in DCD Section 19.1.4 and cable routing, fire barriers, physical separation, flood protection, as discussed in DCD Section 19.1.5).

The staff confirmed that the applicant has used the APR1400 PRA during the design process in conformance with the Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," 60 FR 42622, August 16, 1995, and the SRM dated April 15, 1997, "Staff Requirements - COMSECY-96-061 – Risk-Informed Performance-Based Regulation (DSI 12)," which states, in part, that "the use of PRA technology should be increased in all regulatory matters." The staff finds the use of APR1400 PRA in the design process reasonable and acceptable because the applicant utilized the PRA results and insights to identify improvements and support design decisions.

19.1.4 Safety Insights from the Internal Events PRA for Operations at Power

19.1.4.1 Level 1 Internal Events PRA for Operations at Power

The applicant developed the Level 1 internal events PRA consistent with the ASME/ANS PRA Standard, to the extent possible (see the discussion in Section 19.1.2.3 of this report). The APR1400 PRA was developed using a small-event-tree method supported by a linked fault tree approach. DCD Section 19.1.4.1 describes the APR1400 Level 1 at-power internal events PRA and results. The section references APR1400-E-P-NR-14001-P, "PRA Summary Report," which was audited in the ERR. The report summarizes the APR1400 PRA development approach, results, and insights found in the APR1400 PRA Notebooks, which are KHNP's internal documentation of the APR1400 PRA.

Summary of Application

DCD Section 19.1.4.1 of the application describes the Level 1 internal events PRA for operations at power, including the results and insights. The technical elements include the following.

- Initiating event analysis
- Accident sequence analysis
- Success criteria analysis
- System analysis (including system dependencies)
- Data analysis (including special event data) and common cause analysis
- Human reliability analysis
- Quantification
- Results and insights.

The applicant used the list of PWR IEs from NUREG/CR-6928, "Industry-Average Performance for components and Initiating Events at U.S. Commercial Nuclear Power Plants," dated 2010, in combination with a thorough analysis of the APR1400 design to identify potential IEs, including support system IEs. The applicant used generic data from NUREG/CR-6928 and NUREG-1829 to estimate the frequency of each IE. The applicant developed an ET for each IE identified to be applicable to the APR1400 design. The ETs describe the accident sequences that can lead to core damage by questioning the ability of mitigating systems and operator actions to respond to an IE. The applicant showed that for each IE, the plant will be brought to a stable and safe condition within the mission time of 24 hours or it is assumed to lead to core damage. The applicant used thermal-hydraulic codes (i.e., MAAP and RELAP) to develop success criteria for mitigating systems and human actions required to respond to the IEs. The success criteria analysis defined key safety functions consistent with the PRA Standard and identified the minimal set of SSCs required for each system to successfully mitigate an event by addressing each key safety function as applicable.

The applicant used seven causes of system failure and unavailability modes that were represented in the initiating events analysis and sequence definition. The causes were random component failures, outages for maintenance and testing, support system failures, common cause failures (CCFs), initiating events which affect the systems, human error involving failure to restore equipment to its operable state, and human error involving failure to perform procedural actions. These causes were included as contributors in the fault tree models. The applicant assumed test and maintenance intervals to be bounded by the TSs in DCD Chapter 16. SR SY-A15 of the ASME/ANS PRA Standard was used to screen components or specific component failure modes. The applicant performed data analysis including component unreliability data, component unavailability data, common cause analysis, and special event data. The majority of component failure data was taken from NUREG/CR-6928. If the data was not available in NUREG/CR-6928, the applicant used other sources such as NUREG/CR-5500, "Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998." The applicant used generic data derived from NUREG/CR-6928 to evaluate component unreliability data and for component unavailability data in absence of plant data. The applicant applied the Alpha Factor methodology to calculate the probability of common cause events. The uncertainty distributions for the Alpha Factor parameters were developed in accordance with NUREG/CR-5485, "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment." Generic data for CCF from NUREG/CR-5497, "CCF Parameter Estimations, 2010 Update," and the latest CCF parameter updates from the NRC "Reactor Operational Experience Results and Databases" were applied to evaluate the CCF parameters. The applicant developed pre-initiator actions and post-initiator actions using multiple methods, including the Accident Sequence Evaluation Program (ASEP) framework described in NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," cause-based decision tree methodology (CBDTM), human cognitive reliability / operator reliability experiment (HCR/ORE) methodology, annunciator response model (ARM), and Technique for Human Error Rate Prediction (THERP) (NUREG/CR-1278, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications).

The applicant used the CAFTA computer code to develop its PRA model and quantify the APR1400 CDF and LRF. Human error probabilities (HEPs) were calculated using the EPRI HRA Calculator. The application reports the mean CDF from Level 1 internal events at power as 1.3E-6/year. Loss of offsite power (LOOP) IE strongly dominates the internal events CDF (36 percent) while MLOCA and general transients contribute 12 percent and 11 percent, respectively. The PRA results were analyzed to identify the significant contributors to the CDF from IEs, accident sequences, and basic events. The applicant performed post-processing to address HEP dependency within cutsets with multiple HEPs. In addition, the applicant presented the significant contributors to core damage, risk significant SSCs, and human actions and the insights gained from the PRA results, including uncertainty and sensitivity analyses.

The applicant calculated the total mean APR1400 at-power CDF from internal events to be 1.3E-6/yr.

<u>Technical Evaluation</u>

Initiating Event Analysis

The staff reviewed the IE analysis, including auditing the PRA notebooks, against RG 1.200 and SRP Section 19.0 to determine whether the applicant included the following.

- Sufficiently detailed identification and characterization of IEs

- Grouping of individual events according to plant response and mitigating requirements
- Proper screening of any individual or grouped IEs
- Estimate of the annual frequency of each IE or IE group.

In general, the APR1400 design is similar to operating PWRs with additional design improvement of redundancy and diversity. Therefore, the staff reviewed and compared the APR1400 IEs against the existing list of known PWR initiators in NUREG/CR-6928 for completeness. The staff finds that the IEs were similar. In some cases initiators such as SLOCA were reflected as a combination of multiple initiators. The analysis for design-specific IEs did not result in additional IEs. The staff finds that this initiator grouping approach adequately considered the expected plant response and consequence for each initiator to ensure it resulted in an appropriate grouping of similar events. The application considered failure of support systems that could lead to an IE. This systematic evaluation utilized fault tree modelling to determine the impact of support system failures. The staff finds that the support system IEs compared similarly with those identified in NUREG/CR-6928. While it was determined that some of the events were specifically applicable to the APR1400 design, the applicant provided the bases for excluding those events that do not result in a plant trip or significant plant perturbation. The staff finds this acceptable since it is consistent with the ASME/ANS PRA standard endorsed by RG 1.200. However, the applicant did not include (screened out) very small LOCA (VSLOCA), CCF of 4.16kV buses, and loss of dc (LODC) for ‘C’ and ‘D’ trains as potential IEs. Therefore, on February 22, 2016, the staff issued RAI 410-8357, Question 19-29 (ML16054A197), to address this issue. On April 12, 2016 the applicant submitted a response and on October 28, 2016, the applicant provided an updated response to RAI 8357, Question 19-29 (ML16103A400 and ML16302A484). The staff found the applicant’s updated response on October 28, 2016 acceptable because the applicant proposed to either screen out the potential initiators or group them into the existing IEs which is consistent with the ASME/ANS PRA standard endorsed by RG 1.200. The applicant updated DCD Section 19.1.4.1 to add VSLOCA frequency to the SLOCA frequency since both initiating events have similar plant responses, although, the VSLOCA may not result in a reactor trip in this plant design. The staff finds this approach to be acceptable and conservative because it eliminates the need to perform a detailed analysis of how the plant would respond to a VSLOCA that is highly unlikely to result in a plant trip or significant changes to the plant response when a similar plant response to a SLOCA is already being evaluated. The applicant provided reasonable and acceptable bases for screening out the CCFs of 4.16kV buses and LODC. The staff confirmed these events were determined to not lead to a plant perturbation. RAI 410-8357, Question 19-29, is resolved and closed.

In response to questions during the PRA supporting documentation audit, the applicant stated that the support systems such as component cooling water (CCW) and essential service water (ESW) are conceptual designs and the calculated IE frequencies developed from the associated fault tree modeling may not represent the final design of those support systems. The calculated frequencies were not found to be significantly different from the generic frequency data. Hence, the applicant used industry generic IE frequencies for the supporting system initiating events (SSIEs) in its PRA. While the design-specific support system IEs would be more realistic, the staff found it reasonable that the generic IE frequencies would provide the appropriate results and insights sought from the PRA, with the understanding that the designs of these systems are conceptual, detailed design information is not yet available, and the COL applicant may elect to utilize a modified design. In addition, as presented in the DCD, the calculated frequencies from the fault trees were not significantly different from the generic frequencies.

Since APR1400 is a new reactor design, its design-specific PRA does not utilize plant-specific design information, plant operating procedures, operating experience, or personnel interviews. However, the staff's review of the analysis noted that the applicant considered existing similar operating plant experience. The staff also observed that the applicant utilized generic IE frequencies in its PRA model for the same reasons stated above. This is acceptable for new reactor designs that do not have the benefit of operational experience as discussed in SRP Section 19.0.

The applicant completed a comprehensive review of the design and utilized insights from operating experience with similar plant designs to develop a set of potential IEs. The staff finds the IE analysis reasonable because it was performed in a manner consistent with RG 1.200 and SRP Section 19.0. Where possible and applicable, the APR1400 IE analysis met the high level requirements and SR Capability Category I of ASME/ANS PRA Standard as endorsed by RG 1.200, therefore, it is acceptable.

Accident Sequence Analysis

The staff reviewed the accident sequence analysis, including auditing the PRA notebooks, against RG 1.200 and SRP Section 19.0 to determine that the applicant adequately performed the following.

- Modeled the design-specific scenarios that can lead to core damage following each IE defined in terms of hardware, operator action, timing requirements, and desired end states (e.g., core damage or plant damage states)
- Considered functional, phenomenological, and operational dependencies and interfaces.

The applicant used event tree structure to describe the plant scenarios affecting key safety functions that could lead to core damage following each IE. The staff reviewed all 25 event trees to ensure they reasonably described the mitigating system response, operator actions, and recovery actions that support key safety functions within the defined mission time. In addition, the staff's review considered the applicant's modeling of dependencies that could affect the ability of mitigating systems to operate and complete their safety functions. The staff selected for detailed review and completeness the following event trees: LOOP, SLOCA, general transients, and LOFW based on a review of the top accident sequences and experience with PRA's for similar reactor designs. The staff finds that the event tree modeling, including identification of key safety functions, was performed with a reasonable approach consistent with the ASME/ANS PRA Standard, as endorsed by RG 1.200, and similar industry PWR PRAs. The model adequately considered the combination of successes, failures, and recovery actions that would lead to core damage or steady states during event tree development. The success criteria considered in the event tree development are evaluated in the success criteria analysis section below. The top events representing mitigating systems are evaluated in the systems analysis section below.

The staff found insufficient information in the DCD about how IEs affect the response and operability of mitigating systems and operators. Therefore, on February 23, 2016, the staff issued RAI 416-8358, Question 19-37 (ML16054A199), to address this issue. On April 12, 2016 the applicant provided a response and on June 23, 2016, the applicant provided an updated response to RAI 416-8358, Question 19-37 (ML16103A497 and ML16175A665, respectively). The staff found the updated response acceptable because the applicant provided tables that adequately show dependencies that can affect the mitigating systems, addressing staff's concerns. The applicant updated DCD Section 19.1.4.1 to add the dependencies between front line systems and support systems, and dependencies amongst support systems. RAI 416-8358, Question 19-37, is resolved and closed.

The analysis identified 13 key operator actions. The staff evaluated the appropriateness of each one by reviewing the timing and applicability of the actions within the accident sequences for which they are credited. These key operator actions were verified to play a vital role in preventing core damage or reducing the likelihood of core damage occurring. The staff finds that the operator actions were placed within the appropriate timing of the sequence of events.

The applicant developed detailed event trees taking into account the limited information and/or in-depth analyses available for new reactor designs. The staff finds the accident sequence analysis reasonable because it was performed in a manner consistent with RG 1.200 and SRP Section 19.0. Where possible and applicable, the accident sequence analysis fulfilled the high level requirements and SR Capability Category I of ASME/ANS PRA Standard as endorsed by RG 1.200, therefore, it is acceptable.

Success Criteria Analysis

The staff reviewed the success criteria analysis, including auditing the PRA notebooks, against SRP Section 19.0 and RG 1.200 to evaluate whether the applicant adequately the following.

- Determined the minimum requirements for each function (and ultimately the systems used to perform the functions) to prevent core damage (or to mitigate a release) given an IE
- Used best-estimate engineering analyses applicable to the actual plant design and operation, as available.

The staff reviewed the success criteria specified for the mitigating systems and human actions to ensure consistency with the design features. The staff also reviewed the PRA thermal-hydraulic results and supporting engineering analyses and compared the success criteria to those determined for the design basis accidents documented in DCD Chapter 15.

The staff finds that the applicant appropriately defined core damage and specified the plant parameters of core damage. The applicant used the definitions in the ASME/ANS PRA Standard as endorsed by RG 1.200 and assumed a mission time of 24 hours which represents accepted PRA industry practice. The bases for the success criteria were determined to be consistent with the features and design of the plant. However, the DCD did not describe the approach used to determine the success criteria for the different LOCA sizes identified. Therefore, on February 23, 2016, the staff issued RAI 417-8359, Question 19-39 (ML16054A289) to address this issue. On April 12, 2016, the applicant responded to

Question 19-39 (ML16103A387). The applicant's response stated that the LOCA break sizes were selected using information from NUREG/CR-6928. The staff found the applicant's response acceptable because the LOCA sizes were selected using generic LOCA sizes for which generic data exist. In addition, the staff evaluated the LOCA analysis documented in the success criteria notebook, which was referenced in the PRA Summary Report and found the results for the different LOCA sizes were similar in size to the NUREG/CR-6928 LOCAs. As a result, the applicant updated DCD Section 19.1.4.1 to reflect the source of the LOCA break sizes selected for the IE analysis. RAI 417-8359, Question 19-39, is resolved and closed.

The applicant used thermal hydraulic codes MAAP and RELAP to develop the PRA success criteria. The staff verified that these codes were used within their known limits of capability to specify success criteria for each of the key safety functions identified for each modeled IE. In addition, when applicable, the applicant used appropriate conservative generic analyses in the success criteria development. The staff finds these analyses acceptable because it resulted in a reasonable set of success criteria appropriate for the DC application stage.

The applicant did not evaluate and describe the reasonableness and acceptability of the differences between success criteria for internal events in DCD Chapter 19 and design basis plant response information in DCD Chapter 15. Therefore, on March 23, 2016, the staff issued RAI 417-8359, Question 19-42 (ML16054A289), to address this issue. On March 7, 2018 the applicant submitted a response to RAI 417-8359, Question 19-42 (ML18066A518). The staff found that the applicant's response revised the success criteria table to reflect results of a more recent Chapter 19 PRA success criteria analysis performed. As a result, the applicant updated DCD Tables 19.1-8 and 19.1-2 to reflect the new results. The staff finds this approach to be reasonable and acceptable because the updated tables reflect as a minimum success criteria similar to those developed for the conservative design basis analysis. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 417-8359, Question 19-42, is resolved and closed.

The applicant developed applicable success criteria for the APR1400 design using industry accepted codes and analyses. The staff finds the success criteria analysis reasonable because it was performed in a manner consistent with RG 1.200 and SRP Section 19.0. Where possible and applicable, the success criteria analysis fulfilled the high level requirements and SR Capability Category I of ASME/ANS PRA Standard, as endorsed by RG 1.200, therefore, is acceptable.

System Analysis

The staff reviewed the system analysis against SRP Section 19.0 and RG 1.200 to determine that for the developed models, the applicant adequately performed the following.

- Reflected the as-designed, as-built, as-operated plant (as applicable) including how it has performed during the plant history for operating plants
- Reflected the success criteria for the systems to mitigate each identified accident sequence
- Captured impact of dependencies, including support systems and harsh environmental impacts

- Included both active and passive components and failure modes that impact the function of the system
- Included common-cause failures, human errors, unavailability resulting from test and maintenance, etc.

Specifically, the staff reviewed the applicant's system analysis performed as documented in DCD Section 19.1.4.1.1.4 and audited the system analysis notebook, APR1400-K-P-NR-013106, “Full Power Level 1 PRA – System Analysis Guidance,” provided in the ERR. The audit included a sample of systems and their system failures and unavailability modes, CCFs, and intersystem and intra-system dependencies. The staff also reviewed a sample of the systems modeled for completeness. During the audit, the staff sampled specific system notebooks such as reactor protection system (RPS) and engineered safety features actuation system (ESFAS) based on previous DC reviews and RAI responses.

The staff could not find any information on the applicant’s digital instrumentation and control (I&C) modeling and therefore the staff requested additional information on the applicant’s modeling of their digital I&C systems including: system description (e.g., describe the functions, subsystem interfaces, operator actions, etc.), key assumptions (e.g., modeling, uncertainties), CCF analysis of both the hardware and software, including the basis and/or justification of this information, and failure effects, if modeled at the system/subsystem level. On October 22, 2015, the staff issued RAI 271-8290, Question 19-15, to address the information needed (ML15295A266). KHNP submitted its original response on March 17, 2016 (ML16077A326), and the final response on August 6, 2018 (ML18218A206). The final response fully addressed the staff’s requests with additional information on the locations of the reactor trip system (RTS) analyses and ESFAS analyses in the ERR, attachments describing the failure modes and effects analysis of the RTS and ESFAS, evaluations of several sensitivity cases related to the software reliability values and CCF. The applicant committed to update the DCD accordingly with additional information regarding the CCF within the digital I&C system. The staff reviewed the response and supporting documents against guidance in DI&C-ISG-3 and discussed the information with the Chapter 7 reviewers for consistency across deterministic and probabilistic information and found the response to be acceptable. The staff acknowledges the digital I&C is not fully designed during the DC stage. The staff finds the applicant’s proposed probabilities in this RAI to be reasonable to gain risk insights. The staff confirmed that the RAI response markup was incorporated into Revision 2 of the DCD. Therefore, the staff considers RAI 271-8290, Question 19-15, to be resolved and closed.

During the regulatory audit on PRA, the staff requested clarification on the dependencies between front line systems on support systems and support systems on other support systems. This information was missing from the DCD and RG 1.200 states that applicants should consider capturing the impact of dependencies in the models. The applicant provided additional tables to show the dependencies between front line systems on support systems and support systems on other support systems on September 11, 2015 (ML15254A450), and updated the DCD accordingly. The staff reviewed these tables against RG 1.200 and found them to be consistent with the APR1400 design.

The staff finds that the applicant reasonably modeled the as-designed plant and completed the failure and unavailability modes for each system. The applicant described system-level success criteria, assumptions, system dependencies, common-cause failures, and other criteria as outlined above. The staff finds the system analysis has been performed in a manner consistent with the SRP Section 19.0 and RG 1.200, and therefore, is acceptable.

Data Analysis

The staff reviewed the data analysis against SRP Section 19.0 and RG 1.200 to determine that the applicant adequately considered the following.

- Estimation of parameters associated with IE, basic event probability models, recovery actions, and unavailability events using plant-specific and generic data, as applicable
- Estimation is consistent with component boundaries
- Estimation includes a characterization of the uncertainty.

Specifically, the staff reviewed the data analysis including component failure data, unreliability data, unavailability data, and common cause analysis as documented in DCD Section 19.1.4.1.1.5, reviewed DCD Table 19.1-14, “Component Failure Rate Data,” which described component failure data, including the data source and uncertainty parameter, and audited the data analysis guideline notebook, APR1400-K-P-NR-013104, “Full Power Level 1 PRA – Data Analysis.” The data analysis section was also evaluated to ensure that each parameter was clearly defined, generic parameters were chosen consistent with the parameter definitions in high-level requirement (HLR)-DA-A of the ASME/ANS PRA Standard as endorsed by RG 1.200, parameter estimates were based on relevant generic industry evidence and accompanied by a characterization of the uncertainty, and the document was consistent with the applicable SRs of the ASME/ANS PRA Standard.

In relation to the digital I&C modeling discussed in the system analysis section above, the staff’s review did not find the CCF of digital I&C systems accounted for in the data analysis. This information was requested in RAI 271-8290, Question 19-15, as mentioned above. KHNP submitted a revised response (ML18218A206) on August 6, 2018, and committed to update the DCD with the requested failure probability. Since the APR1400 digital I&C is not fully designed yet at this stage, the staff reviewed the CCF probability assignment and finds it reasonable to capture the risk insights. The staff confirmed that the RAI response markup was incorporated into Revision 2 of the DCD. Therefore, the staff considers RAI 271-8290, Question 19-15, to be resolved and closed.

The staff finds the applicant has reasonably estimated the component failure data, unreliability data, unavailability data, and common cause analysis. The applicant described the uncertainty for parameters and sources for generic data used. The staff finds that the above data analysis has been performed in a manner consistent with the SRP and RG 1.200, and therefore, it is acceptable.

Human Reliability Analysis

The staff reviewed the human reliability analysis against SRP Section 19.0 and RG 1.200 to determine that the applicant adequately considered:

- Identification and definition of the human failure events that would result in IEs or pre- and post-accident human failure events that would impact the mitigation of IEs
- Quantification of the associated human error probabilities taking into account scenario (where applicable) and plant-specific factors (as available) and including appropriate dependencies (both pre- and post-accident).

Specifically, the staff reviewed the applicant's HRA results documented in DCD Section 19.1.4.1.1.6 and audited the HRA notebook, APR1400-K-P-NR-013105, "Full Power Level 1 PRA – Human Reliability Analysis." This review included looking at the list of risk important human actions as listed in DCD Table 19.1-24, "Level 1 Internal Events Key Operator Actions by RAW (CDF)" and Table 19.1-25, "Level 1 Internal Events Key Operator Actions by FV (CDF)."

The staff's review did not find sufficient information regarding the HFEs in the DCD Rev 0. Therefore, on November 16, 2015, the staff issued RAI 312-8343, Question 19-17 (ML15320A351), to request additional information, including a list of pre- and post-initiator HFEs, the corresponding probability and its bases, and the screening value in conformance with SRP Section 19.0. The applicant submitted a response on March 17, 2016 (ML16077A356). The staff discussed this RAI and the submitted response with the applicant at a public meeting on May 4-6, 2016 (ML17081A449). The staff noticed an inconsistency in the labeling of operator actions in the first RAI response. The staff asked the applicant for confirmation that the operator action "WOOPH-S-1A-2AB" from the DCD is the same as "WOOPV-S-1AB2AB" from the RAI response. The applicant submitted a revised response (ML16302A491) on October 28, 2016. The response confirmed the operator action labeling and contained tables of pre-initiator and post-initiator HFEs along with the details in two attachments and an update to the key operator actions by FV table. The staff reviewed the information using criteria in RG 1.200 and SRP Section 19.0 and found it to be acceptable. The staff confirmed that the RAI response markup was incorporated into Revision 1 of the DCD. Therefore, the staff considers RAI 312-8343, Question 19-17, to be resolved and closed.

The staff finds the applicant reasonably identified pre-initiator and post-initiator human failure events, dependencies, and risk important human actions. The staff finds that the above human reliability analysis has been performed in a manner consistent with the SRP and RG 1.200 guidance, and therefore, it is acceptable.

Quantification

The staff reviewed the applicant's quantification analysis to ensure the CDF was quantified using the appropriate models and codes and that the analysis approach can support the quantification of LRF. The staff also evaluated the analysis to ensure that the applicant has identified sufficient significant contributors to CDF, made reasonable assumptions in support of the PRA development, and characterized sources of model uncertainty properly.

The staff reviewed the quantification results in the DCD and all the information in the applicants APR1400 PRA Quantification Notebook, a document referenced in the APR1400 PRA Summary Report.

During the APR1400 DC review, the applicant decided to convert its PRA software platform from SAREX to CAFTA and update the DCD to reflect the CAFTA results and changes in the outcomes. The resulting CDF from the CAFTA model was found to be very close to the previously estimated CDF using SAREX software. The staff further compared the cutsets generated by the CAFTA model with those quantified by SAREX software. This comparison was performed to determine the reasonableness of the cutsets presented by the applicant. Due to the model corrections and minor changes during the conversion, there was not an exact match in the outcomes. The staff finds the applicant's conversion from SAREX to CAFTA logical and acceptable because the model was developed with an adequate level of detail to perform sensitivity and uncertainty analysis and develop a relative risk ranking of cutsets and SSCs. The results were determined to be close to the previous results and the applicant was able to explain changes in the ranking of important events, sequences, and SSCs. The quantification analysis developed insights by performing risk importance measures, uncertainty analysis, and sensitivity analysis. The results and risk insights are further evaluated in the section below.

The staff finds the APR1400 quantification analysis acceptable because it was performed in a manner consistent with RG 1.200 and SRP Section 19.0. Further, it was determined that when possible, the quantification analysis fulfilled the high level requirements and SR Capability Category I of ASME/ANS PRA Standard as endorsed by RG 1.200, therefore, it is acceptable.

Results and Insights

Core Damage Frequency

The staff finds the estimated APR1400 at-power IEs CDF of 1.3E-6/year (yr) is reasonably less than the Commission's goal of 1E-4/yr, as cited in NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants. The staff finds this CDF is reasonably low compared to the currently-operating PWR fleet due to the design features incorporated in the APR1400 and its design improvements.

Uncertainty Analysis

The applicant quantified the uncertainty in the Level 1 IEs PRA using the UNCERT code to run Monte Carlo simulations. The parametric uncertainty is summarized below:

5 percent value:	3.5E-7/yr
Mean value:	1.3E-6/yr
95 percent value:	3.4E-6/yr

The 95th percentile CDF value is more than an order of magnitude below the Commission's goal of 1E-4/yr. The modeling uncertainty was evaluated via sensitivity analysis.

Sensitivity Analysis

Four sensitivity cases were analyzed by the applicant in three different areas: digital I&C system common cause failure sensitivity case, RCP seal LOCA sensitivity case, and hot leg injection sensitivity case.

In the digital I&C sensitivity case, a series of sensitivity analyses were performed to increase the values of software CCFs for both operating software and application software by factors of 10 up to 100. Cases were also performed where operating system and application software CCFs for plant protection system (PPS) and diverse protection system (DPS) were increased by 10 then 100. When all CCFs were increased by a factor a 10, CDF increased by 42 percent. When all CCFs were increased by a factor of 100, CDF increased by about 500%. When all PPS and DPS software CCF was set to 0, or never fails, the CDF only decreased by 5 percent. These analyses were performed to demonstrate the redundancy in the design and low failure probability assigned to digital I&C in the design.

Additional sensitivity cases were performed to better understand the CDF sensitivity to the software reliability values used in the DI&C system modeling. The sensitivity cases revealed that relatively large increases in DPS software CCF has little impact on CDF. It also evaluated the importance of operator action to overcome software CCF.

The RCP seal LOCA sensitivity case was performed to determine the impact on CDF by increasing the conditional seal failure probabilities (CSFP). When the CSFPs are increased by a factor of about 40, the CDF increased by 3 percent. When the CSFPs were increased by a factor of about 400, the CDF increased by about 32 percent. When seals are guaranteed to fail, CDF increases by about 350 percent to 4.8E-6/yr. The purpose of this sensitivity case was to show the model is not very sensitive to CSFPs. The applicant originally assumed a RCP seal LOCA probability based on engineering judgement. The staff issued RAI 312-8343, Question 19-16, on November 16, 2015 (ML15320A351), to request more information about the RCP seal assumptions, leakage, detailed modeling, and failure information in conformance with SRP Chapter 19 “Acceptance Criteria.” The applicant submitted a revised response on August 6, 2018 (ML18218A201), which included a discussion of the basis for the engineering judgement, event trees the RCP seal LOCA were considered in, the limiting leakage rates for normal operation and accidents, and the timing of seal failure. The applicant also summarized Westinghouse report, APR1400-A-M-NR-16001 (WCAP-18067), Revision 0, “PRA Model for RCP Seal Failure Given Loss of Seal Cooling for APR1400 KSB HDD-254 Type F RCP Seals,” in the RAI response to provide additional information on the engineering judgement for the RCP seal LOCA probability. The staff reviewed the response and finds it acceptable because it provided reasonable justification for the assumed RCP seal probability based on the leak tests documented in the Westinghouse report, APR1400-A-M-NR-16001. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 312-8343, Question 19-16, is resolved and closed.

For a medium break LOCA, hot leg injection (HLI) is assumed to not be needed. HLI is required for large break LOCA to prevent boron precipitation from potentially plugging the core upper channels. A sensitivity case was performed that required HLI for a medium break LOCA. The results showed the CDF increases by 39 percent to 1.5E-6/yr if HLI was required for all medium break LOCA. Operator action to align HLI dominated the CDF increase.

The staff reviewed the sensitivity analyses consistent with SRP Section 19.0, and finds the studies useful to gain understanding of the uncertainty and robustness of the design. The staff finds the sensitivity analyses will be useful during the COL stage.

Risk Insights

The APR1400 is associated with many risk insights and the key risk insights are described below. The APR1400 CDF is dominated by LOOP events. This is not surprising since the APR1400 depends on active systems that need electrical power for operation. The total LOOP CDF is about 3.9E-7/yr as a result of high redundancy in trains and diversity in emergency power supplies.

The DPS is an important backup to PPS and engineered safety feature – component control system (ESF-CSS) systems and is not installed on the same qualified PLC platform as the PPS. DPS is on an independent non-safety platform to maintain diversity so DPS functions after PPS software CCF.

The APR1400 RCP seal package has a low probability of failure given that the RCPs are tripped. Due to this, seal LOCAs following LOOP events which result in RCP trip due to loss of power are considered to be insignificant with a CDF of 4.6E-10/yr.

The staff reviewed the risk insights and finds them to be reasonable and consistent with the staff's understanding of the design.

<u>Significant Sequences and Cutsets</u>

The staff evaluated the reasonableness of the top cutsets contributing to 50 percent of the internal events CDF to ensure the PRA model adequately reflected the APR1400 design and that the applicant had performed post-processing on the results to identify unexpected cutsets. The findings and applicant's resolution of each issue identified by the staff are as follows:

- The top 100 cutsets include two cutsets (#55, #73) with a single basic event (component) failure in response to the IE that results in core damage. These cutsets indicated that the APR1400 design is potentially vulnerable to a single failure event. During the audit, the applicant explained the assumption resulting in this vulnerability. The applicant in its PRA model assumed that a hot leg injection is needed in one of the two hot leg pipes during a large break LOCA event. This model assumption resulted in only one piping connection available to support hot leg injection during a hot leg break LOCA. Failure of this hot leg injection would lead to core damage. Additional analysis performed after the application had been submitted shows that the hot leg injection is not needed during a hot leg break LOCA. The staff finds the event not credible and recognizes that it is not an industry practice to model this scenario in the PRA. Furthermore, the applicant's analysis showed adequate cooling of the reactor without hot leg injection during a large break LOCA event. The DCD was revised to add an explanation to Section 19.1.4.1.2.5 clarifying the assumption utilized in the PRA modeling which led to the single failure cutsets. In addition, the applicant added a COL item to have the COL applicant reassess these single failure cutsets in its plant-specific PRA. Lastly, DCD Section 19.1.4.1.2.3 was also revised to include a discussion of the assumptions and modeling that resulted in the single failure cutsets. The staff finds the applicant's response reasonable and acceptable because the revision of the DCD clarifies that the design is not vulnerable to a single failure event and the applicant proposed a COL item (COL 19.1(26)) that will require a COL applicant to reflect this information in its PRA model.

- The staff identified three cutsets (#28, #51, #100) in Table 19.1-19, "Level 1 Internal Events Top Accident Sequences," with mutually exclusive basic events indicating the scenarios that redundant components/trains will be placed in the same test and maintenance configuration during operation ASME/ANS PRA Standard as endorsed by RG 1.200, Quantification SRs QU-B7 and QU-B8 require deletion of mutually exclusive events from the PRA results. The staff finds that these cutsets include a combination of basic events that are not expected to occur at the same time interval during plant operation and thereby impact the CDF result. The applicant included in its response to RAI 434-8352, Question 19-92, a revision to the DCD Section 19.1.4.1.1.4 to address these mutually exclusive cutsets and add a COL item (COL 19.1(27)) to have the COL applicant reevaluate these cutsets during its application. The staff finds that these are not credible scenarios that the plant will be placed in during operation. When deleted from the PRA results, these cutsets will contribute to a decrease in total CDF as confirmed by the applicant in its response to RAI 19-92. The applicant also confirmed that including these cutsets in the APR1400 PRA quantification would have insignificant impact on the results, insights, and the use of APR1400 PRA. The applicants approach is acceptable because the mutually exclusive events were identified and discussed in the DCD and expected to have an insignificant impact on the PRA results and insights. In addition, the COL information item will ensure the COL applicant addresses these cutsets in its plant-specific PRA.

Significant SSCs, CCFs, and Operator Actions

The staff reviewed the tables of risk important components, component common cause failures, and operator actions. The staff evaluated the method used to determine risk significance including the application of risk achievement worth, Fussell-Vesley importance, and the thresholds applied to each risk metric. In addition, the staff reviewed the tables of risk significant SSCs for completeness, (i.e., ensuring that components/trains in standby that did not meet the risk importance threshold were added to the tables of risk significant SSCs).

DCD Table 19.1-22, "Level 1 Internal Events Key CCF Events by RAW (CDF)," shows the significant common cause events based on RAW importance. The highest CCF events are failure of reactor trip due to either mechanical failure, or random failure of the reactor trip circuit breakers, the 125 1E Vdc batteries, and various AF system CCFs. Table 19.1-23, "Level 1 Internal Events Key CCF Events by FV (CDF)," shows the significant common cause events based on FV importance. In the response to Question 19-15 discussed previously, the applicant committed to update Tables 19.1-22 and 19.1-23 to include digital I&C CCF as part of the response to Question 19-92. Based on the review of the DCD Tables 19.1-22 and 19.1-23, the staff has confirmed incorporation of the changes described above; therefore, RAI 434-8352, Question 19-92, is resolved and closed.

The staff finds that the applied risk metrics and thresholds were in accordance with the accepted industry practices and consistent with RG 1.200. The applicant adequately applied the risk thresholds and accounted for the PRA model asymmetry to develop a comprehensive list of risk significant SSCs. The staff finds the lists reasonable and acceptable because they are comprehensive and provide insights into the design that reflect the important components and actions.

Key Assumptions

The staff reviewed the applicant's approach for identifying key assumptions and sources of uncertainty. The review included an assessment of the assumptions identified by the applicant and a comparison of those assumptions to staff's knowledge and understanding of the components of the APR1400 design that are driving the PRA results and insights.

The staff finds the assumptions identified to be comprehensive. The key assumptions were discussed in detail to the extent needed for a DC application. As a result of questions raised by the staff during the PRA audit, the applicant added an assumption to the list addressing hot leg injection during a large break LOCA in one of the hot legs. This assumption was evaluated and discussed above. The staff, during its review, did not identify any additional assumptions of significance. The staff finds the key assumptions reasonable because the applicant specified the assumptions made in the PRA model and provided the basis for each assumption. The key assumptions are acceptable because they are consistent with the requirements for describing assumptions in SRP Section 19.0.

Conclusion on At-power Internal Events PRA

The staff finds the applicant reasonably evaluated IE analysis, accident sequence analysis, success criteria analysis, system analysis (including system dependencies), data analysis (including special event data) and common cause analysis, human reliability analysis, quantification, and results (including CDF, uncertainty and sensitivity analyses) and insights (including risk insights, significant cutsets and sequences, and key assumptions) for the APR1400 Level 1, at power internal events. The staff finds that the APR1400 Level 1 internal events has been performed in a manner consistent with the SRP Section 19.0 and RG 1.200 guidance, and therefore, it is acceptable.

The staff confirmed that the APR1400 PRA met the SR Capability Category I of the ASME/ANS PRA Standard to the extent possible. The applicant adequately addressed those SRs that do not meet Capability Category I in the DCD and evaluated the potential impact on the PRA results and insights. The staff finds this approach reasonable and acceptable in conformance with the SRP Section 19.0. The staff concludes that the impact from the SRs that were not met on the APR1400 DC application is not significant and the Commission's goals, as discussed in SECY-90-016, "Evolutionary Light-Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements" and SECY-89-102, "Implementation of the Safety Goals," and their related SRMs, remain met.

19.1.4.2 Level 2 Internal Events PRA for Operations at Power

Summary of Application

The Level 2 analysis uses both deterministic and probabilistic analysis tools to follow the progression of the core damage accidents. Computer analysis codes were used to simulate the meltdown of the core, failure of the reactor vessel (RV) due to contact with molten core materials, and transport and interactions of core debris in the containment. Because of the large uncertainties associated with the progression of a core damage accident, these deterministic calculations were supplemented with assessments that considered the potential for phenomena different from or more severe than those treated in the analysis codes. This part of the analysis included an assessment of the potential for a variety of containment failure modes for each type of core damage sequence, and an estimate of the magnitude of the radionuclide release that would be associated with each mode.

Plant Damage State Analysis

The Level 2 analysis begins with the end point of the Level 1 analysis (i.e., core damage). Accident sequences identified from the Level 1 analysis are assigned to one of several plant damage states (PDSs) which allows many sequences exhibiting similar characteristics to be analyzed in detail in the Level 2 analysis, while keeping the total number of PDSs to be analyzed at a manageable level.

The process of the PDS analysis for the APR1400 is as follows:

a. Define the PDS characteristics to identify the physical characteristics and the accident sequence characteristics of the core damage sequences
b. Develop the PDS event tree logic diagram
c. Extend the Level 1 event trees to PDS event trees by questioning the status of functions that can affect containment integrity
d. Group the extended core damage sequences (i.e., the end point of the PDS event trees) into the plant damage states by using a systematic logic diagram.

The applicant used the following PDS grouping parameters as listed in DCD Table 19.1-26, “PDS Grouping Parameters” and described in DCD Section 19.1.4.2.1.1:

a. Containment bypass
b. Containment isolation
c. LOCA or transient
d. RCS pressure at core damage
e. Cavity condition
f. In-vessel injection
g. Release point
h. Containment heat removal
i. SG / feedwater available.

Containment Event Tree Analysis

Containment event trees (CET) are developed to model the containment response during SA progressions. The potential SA progression for each PDS is unique and normally would be represented by a specific CET. However, for most PDSs the potential severe accident progressions are very similar and can be represented by a general CET. Special CETs represent the accident progressions for the rest of PDSs that pertain to containment bypass, containment isolation failure, and containment rupture before core melt. The important phenomena that can affect the containment failure modes and the source term are also addressed as top events in the CETs. The CET probabilistically evaluates the accident progression to calculate the likelihood of various end states ranging from an intact containment to small releases to large releases. The CETs were developed based on potential containment failure modes listed in NUREG-1335, “Individual Plant Examination: Submittal Guidance,” and considered in the NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants.” They include the following.

a. Direct bypass
b. Containment isolation failure
c. Steam explosion
d. Combustion processes
e. Steam over pressurization
f. Molten core concrete interaction (basemat melt-through)
g. Blowdown forces (vessel thrust force or rocket mode failure)
h. Liner melt-through (direct contact of containment shell with fuel debris)
i. Thermal attack on containment building penetrations.

The containment event trees are shown in DCD Figures 19.1-43 through 19.1-47.

Containment Ultimate Pressure Capacity Analysis

DCD Tier 2 Section 19.1.4.2.1.2.4 “Containment Ultimate Pressure Capacity Analysis,” summarizes the evaluation and results of the containment ultimate pressure capacity analysis with respect to a probabilistic assessment of containment performance. DCD Table 19.1-31 “Containment Failure Modes and Results” and Figure 19.1-60 “Total Containment Fragility Curve” present the results of the analysis. The applicant provided the probability density function for each failure mode (including a failure mode based on leakage) and estimated a total fragility curve.

The total fragility curve is determined by combining the results from the individual containment failure modes into a single distribution representing the capacity. The containment pressure fragility is determined from the following failure modes.

- Failure of the membrane
- Failure of the cylindrical wall at the basemat
- Failure of the basemat
- Failure of the equipment hatch
- Failure of the personnel access airlock

- Failure of the personnel emergency exit airlock
- Failure of the fuel transfer tube.

CET Phenomenological Evaluations

The MAAP code was used to support many of the CET phenomenological evaluations. MAAP evaluations included evaluations of core melt, RCS failure, containment pressurization, ex-vessel core-concrete interactions, and releases from the containment. Containment failure due to over pressurization was considered using the results of the containment ultimate capacity evaluation. Many other calculations were performed to support the CET, which are described in the DCD.

Release Category Evaluations

The CET accident sequences are grouped into a representative number of release categories that exhibit similar characteristics. A particular release category consists of a group of CET end points that have similar source term governing characteristics. Once the release categories are determined, various accident sequences are allocated to each category. The APR1400-specific source terms are evaluated using the MAAP computer code for one sequence that best represents the release category. The MAAP cases are used to predict the source term characteristics, including the release fraction and the release timing. A source term category for APR1400 is characterized by the following parameters:

a. The frequency of occurrence
b. The isotopic content and magnitude of the release (release fractions of the fission products)
c. The energy of the release to the environment
d. The time of the release to the environment
e. The duration of the release
f. The location of the release (release point of the release height).

The following process was used to select the representative sequence for each specific release category:

a. Select the PDS with the largest contribution to the release category’s total frequency
b. Choose the dominant sequence for the release category from among the accident sequences corresponding to the PDS. This defines the initiating event and the status of the various plant systems
c. The definitions of the CET sequence (i.e., accident progression sequence) are retrieved to determine if any special phenomenological conditions have to be specified
d. A containment failure pressure, failure time, and failure condition are specified based on the release category definition.

A total of 21 release categories were established. Each category is described in DCD Section 19.1.4.2.1.3.

The summary of the MAAP results for release magnitude and release timing, and the release categorization (i.e., large release, large early release, or not large release) are presented in

DCD Table 19.1-33, ”Summary of Source Term Evaluation” and Table 19.1-34, “Source Term Category Frequencies and Contributions to LRF for Internal Events,” respectively.

Results from Level 2 Internal Events PRA for Operations at Power

The following results of the Level 2 internal events analysis are presented in detail in the DCD:

- risk metrics, including large release frequency and conditional containment failure probability
- contributions of the release categories to the total source term
- significant sequences and cutsets
- significant core damage end states, initiating events, phenomena, and basic events
- key assumptions
- results of sensitivity analysis
- results of uncertainty analysis
- risk insights.

Technical Evaluation

Plant Damage State Analysis

The staff reviewed DCD Section 19.1.4.2.1.1, “Plant Damage State Analysis,” and audited “Full Power Level 2 PRA - PDS Analysis” and “Full Power Level 2 PRA - Quantification Notebook” using SRP Section 19.0 guidance.

DCD Section 19.1.4.2.1.1 states that “[t]he large break LOCA sequences result from a primary system break of greater than 15.24 cm (6 in) diameter. The large break LOCA sequences correspond to sequences that would result in RCS pressure in the low pressure range, less than 17.6 kilogram per square-centimeter (kg/cm^2) (250 pound per square inches absolute [psia]).” The DCD is not clear whether the large-break LOCA identified is a single-ended or double-ended guillotine break. Therefore, in RAI 432-8377, Question 19-58, the staff requested the applicant to clarify. In the response, dated May 31, 2016, the applicant stated that per the definition of APR1400 LOCA, the most conservative break size of LOCA would be a double-ended guillotine break of cold leg (or hot leg), as described in the PDS analysis notebook, APR1400-K-P-NR-013601, Revision 0, Appendix A, Case-A01 for analyzing a large break LOCA sequence (ML16152B041). The staff finds this response acceptable because it clarifies the LOCA break size used in the DCD Chapter 19 analysis. RAI 432-8377, Question 19-58, is resolved and closed.

DCD Table 19.1-27, “Frequency of Dominant PDSs,” shows 14 PDSs contribute to 95.4 percent of the core damage sequences. However, using the DCD Revision 0 mean internal event CDF of 1.9E-6/year, the staff calculated only 78.3 percent of core damage sequences represented by the 14 PDSs listed in the table. Therefore, in a public meeting from April 13 to 15, 2015, the staff pointed out this apparent discrepancy to the applicant (ML15097A368). In a letter dated October 1, 2015, the applicant provided a correction to Table 19.1-27 and committed to revise the DCD (ML15274A284). The applicant revised Table 19.1-27 again in the response to RAI 434-8352, Question 19-92. The staff reviewed the changes and found that the corrected table provides 10 PDSs with 93.5 percent CDF, which is more than the 90 percent CDF discussed in RG 1.216. The staff finds the applicant's response acceptable because it is conservative and consistent with RG 1.216. Based on the review of the DCD to verify that it incorporated the text as shown in the RAI response, the staff has confirmed incorporation of the changes described above; therefore, RAI 434-8352, Question 19-92, is resolved and closed.

The staff review finds the PDS analyses to be consistent with SRP Section 19.0 and RG 1.200 guidance, and therefore, acceptable.

Containment Event Tree Analysis

The staff reviewed DCD Section 19.1.4.2.1.2, "Plant Containment Event Tree Analysis," and audited "Full Power Level 2 PRA - CET/DET Analysis" and "Full Power Level 2 PRA - Quantification Notebook" using SRP Section 19.0 guidance.

DCD Section 19.1.4.2.1.2.1 states that "[t]he containment event trees are shown in Figure 19.1-42 through Figure 19.1-46." Of these, the DCD Revision 0 does not describe Figures 19.1-43 through 19.1-46, which show CETs for SGTR, interfacing system LOCA (ISLOCA), containment isolation failure, and containment failure before vessel breach, respectively, nor describe how the applicant determined the top events of these CETs. Therefore, on March 8, 2016, in RAI 432-8377, Question 19-55, the staff asked the applicant to provide this information (ML16068A097). In the response, dated July 29, 2016, the applicant stated the following (ML16211A404):

> Each PDS end point represents a unique accident progression staring point with respect to the CET. In practice, there will be many commonalities for most accident sequences, except for those PDSs such as containment bypass and containment isolation failure. For example, core damage sequences in which containment is successfully isolated and not bypassed would have different modeling approaches with respect to containment challenges compared with those sequences in which containment bypassed initially. Therefore, to model containment responses for most accident sequences (i.e., PDS 8 through 108), a general CET is developed. Special CETs are developed for the containment bypass sequences (i.e., PDS 1 through 4), the containment isolation failure (i.e., PDS 5 through 6) and the containment failure before core melt (i.e., PDS 7).

The applicant proposed adding two new DCD Sections 19.1.4.2.1.2.2 and 19.1.4.2.1.2.3 to describe the general CET and special CETs. The staff finds that applicant's response provides sufficient information about CETs for SGTR, ISLOCA, containment isolation failure, and containment failure before vessel breach for the staff to understand how the applicant has modeled these and determined the top events. The staff finds it acceptable because it is consistent with SRP Section 19.0 and RG 1.200 guidance. The applicant's response also included corresponding DCD changes. The staff confirmed that the applicant incorporated these changes in DCD Tier 2 Revision 1. Based on the applicant's response, RAI 432-8377, Question 19-55, is resolved and closed.

DCD Section 19.1.4.2.1.2.1 provides a description of the decomposition event trees (DETs), but does not provide a description of the DET analysis. Therefore, in RAI 432-8377, Question 19-54, the staff asked the applicant to provide DET details (ML16068A097). In the response, dated July 29, 2016, the applicant stated that DETs are used for quantification of complex CETs (ML16211A404). The applicant proposed to update DCD Section 19.1.4.2.1.2.5 (revised from Section 19.1.4.2.1.2.3 by the response of RAI 432-8377, Question 19-55) to describe DETs for the general CET and special CETs. The general CET has eight headings (a through h) and special CET has four (i through l) as given below, which are quantified using DETs.

a. Mode of RCS failure before vessel breach (RCSFAIL)
b. In-vessel core melt arrest (MELTSTOP)
c. Dynamical containment failure (DCF)
d. Early containment failure (ECF)
e. Late containment heat removal recovery failure (CSLATE)
f. Ex-vessel debris coolability (DBCOOL)
g. Late containment failure (LCF)
h. Basemat melt-through (BMT)
i. Steam generator tube rupture (SGTR)
j. Interfacing systems loss-of-coolant accident (ISLOCA)
k. Containment isolation failure
l. Containment rupture before core melt.

The staff finds that the applicant's response provides sufficient details on all DETs for the staff to understand how they were used to quantify CETs. The staff finds it acceptable because it is consistent with SRP Section 19.0 and RG 1.200 guidance. The staff confirmed that the applicant incorporated these changes in DCD Tier 2 Revision 1. Based on the applicant's response, RAI 432-8377, Question 19-54, is resolved and closed.

The staff review finds the CET analyses is consistent with SRP Section 19.0 and RG 1.200 guidance, and therefore, is acceptable.

CET Phenomenological Evaluations

The staff reviewed DCD Section 19.1.4.2.1.2.3, "CET Phenomenological Evaluations," and audited "Full Power Level 2 PRA - Quantification Notebook" using SRP Section 19.0 guidance.

Section 19.1.4.2.1.2.3 of the DCD states the following:

> The MAAP code was used to support many of the CET phenomenological evaluations. MAAP evaluations included evaluations of core melt, RCS failure, containment pressurization, ex-vessel core-concrete interactions, and releases from the containment. Containment failure due to over pressurization was considered using the results of the containment ultimate capacity evaluation. Many other calculations were performed to support the CET.

However, details of MAAP runs performed to support the APR1400 CET phenomenological evaluations are not provided in the DCD. Therefore, in RAI 432-8377, Question 19-61, the staff asked the applicant to provide details of MAAP runs performed to support the APR1400 CET phenomenological evaluations (ML16068A097). In the response, dated August 9, 2016, the applicant provided a list of MAAP runs performed and proposed to change DCD Section 19.1.4.2.1.2.3 to include the following uses of MAAP cases for CET phenomenological evaluation (ML16222A928):

a. To review the number of cycling of the POSRV and the main steam safety valve (MSSV) cycles before core damage
b. To review the containment pressurization for the sequences with a dry cavity without containment sprays
c. To review the emergency containment spray backup system (ECSBS) performance for containment depressurization, and
d. To review the maximum adiabatic isochoric complete combustion (AICC) pressure inside the containment.

The staff finds that the applicant's response provides sufficient details for the staff to understand how the applicant used MAAP computer code for the APR1400 CET phenomenological evaluations, and is therefore acceptable. The staff confirmed that the applicant incorporated these changes in DCD Tier 2 Revision 1. Based on applicant's response, RAI 432-8377, Question 19-61, is resolved and closed.

The staff review finds the CET phenomenological evaluations is consistent with SRP Section 19.0 and RG 1.200 guidance, and therefore is acceptable.

Release Category Evaluations

The staff reviewed DCD Section 19.1.4.2.1.3, "Release Category Evaluations," and audited "Full Power Level 2 PRA – Source Tem Category Analysis" and "Full Power Level 2 PRA - Quantification Notebook" using SRP Section 19.0 guidance.

10 CFR 52.47(a)(2) states, in part, that "[i]t is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of radioactive fission products." The MAAP calculations documented in APR1400-K-P-NR-013603, which the staff audited, for Source Term Categories 17 and 21 differ only by the release opening area assumed. Although, Case STC21 assumes an area (1.0 ft^2 [0.093 m^2]), which is ten times larger than that for STC17 (0.1 ft^2 [0.0093 m^2]), the iodine release fraction for STC21 is 400 times larger than for STC17. Therefore, in RAI 432-8377, Question 19-59, the staff asked the applicant to explain the significant variation in releases in two cases compared to the area assumed (ML16068A097).

In the response, dated December 7, 2016, the applicant stated that in addition to larger containment opening, late re-volatilization of cesium-iodide (CsI) in the pressurizer in Case STC-21 has caused a larger release than in Case STC-17 (ML16342C937). The staff determined based on the applicant's response that re-volatilization occurs in the pressurizer of STC-21 because of the heating up of the pressurizer wall from deposited radioisotopes. Depressurization of the containment through large opening in STC-21 makes containment gas less dense, in turn making natural convection cooling of pressurizer wall less effective. The staff reviewed the applicant's response and finds it acceptable because it explains the reasons for variation in releases between STC-17 and STC-21. RAI 432-8377, Question 19-59, is resolved and closed.

The staff review finds the release category evaluations is consistent with SRP Section 19.0 and RG 1.200 guidance, and therefore is acceptable.

Results from Level 2 Internal Events PRA for Operations at Power

The staff reviewed DCD Section 19.1.4.2.2.2, “Internal Events Core Damage Release Category Results,” and audited “Full Power Level 2 PRA – Source Tem Category Analysis” and “Full Power Level 2 PRA - Quantification Notebook” using SRP Section 19.0 guidance.

Key Assumption (b) in DCD Section 19.1.4.2.2.5 states that “[t]he conditional probability of PI-SGTR, given ATWS and MSLB/FWLB sequences without feedwater, is assumed to be 0.027 based on engineering judgment of applicable industry references for this probability.” However, the DCD did not list industry references for this statement. Therefore, in a public meeting, the staff requested the applicant to provide justification for the assumed value. On September 11, 2015, the applicant responded stating that the basis can be found in NUREG/CR-6365 "Steam Generator Tube Failures," April 1996 and committed to update the DCD with this information (ML15254A448). As stated in NUREG/CR-6365, the source of the 0.027 value for PI-SGTR is NUREG-0844, “NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity,” September 1988. A more recent estimate of the probability of PI-SGTR is 0.054 (NUREG-1570, “Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture,” March 1998). In addition, draft NUREG-2195, “Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes,” May 2016, provides an estimate of 0.008. The staff agrees with the applicant’s use of the 0.027 value because it is similar to the staff’s more recently estimated values. Also, PI-SGTR is unlikely to affect the results of the Level 2 PRA because the SOARCA study (NUREG-1935, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report,” Revision 1, November 2012) showed that a spontaneous SGTR (such as a PI-SGTR) is unlikely to lead to core damage due to the nearly two days available until core damage, allowing sufficient time for operator action to mitigate. The staff confirmed that the applicant’s APR1400 DCD Revision 1 reflects the proposed change.

DCD Revision 0, Figure 19.1-50, shows a conditional containment failure of 0.138 for the core damage sequences for internal events. This value is higher than the Commission’s containment performance goal (CPG) in SRP Section 19.0, which states that the conditional containment probability (CCFP) should be less than 0.1 for the composite of all core damage sequences assessed in the PRA. Therefore, in a public meeting on May 28, 2015 (ML18128A032), the staff requested that the applicant explain how APR1400 design meets the Commission’s goal. On September 11, 2015, the applicant responded by stating that the 0.138 value included the frequency of small release due to BMT, which would occur much later than 24 hours after core damage, and therefore is not relevant to the CCFP calculation (ML15254A448). The applicant pointed to SRP Section 19.0, SECY-90-016, and SECY-93-087, which specify that an intact containment for 24 hours for more likely scenarios is appropriate.

In response to RAI 434-8352, Question 19-92, as discussed in Section 19.1.2.4 above, the applicant revised DCD Figure 19.1-50, which shows a CCFP of 0.17 (i.e., LRF and small release frequency contributing to 0.10 and 0.07, respectively). DCD Table 19.1-30, “Source Term Category Frequencies and Contributions to LRF for Internal Events,” shows that STC-21, “Late containment failure with a rupture failure size” consists of a frequency of 0.012 of CDF. Excluding STC 21, which contains late release, from LRF, the staff calculated a frequency of 0.088 of CDF. Therefore, excluding small releases and late releases, the CCFP would stay below the CPG.

The staff concludes that the estimated CCFP for the APR1400 design conforms to the CPG (i.e., 10 percent). Specifically, within the 24-hour period after core damage, which is the focus of the CPG, the probability of containment failure is below the goal. However, the CCFP (0.17) is somewhat higher than the goal (10 percent) when small releases and failures beyond 24 hours are included and the structural integrity definition of failure is used. Furthermore, in the related SRM in response to SECY-90-016, the Commission directed the staff that the CCFP objective of 0.1 should not be imposed as a requirement in and of itself. In view of the approximate nature of the CPG, the recognition that PRA results, particularly bottom-line numbers, contain considerable uncertainties, and the fact that close to 48 percent of containment failures reflected in the CCFP estimate of 0.17 will result in small releases and late releases, the staff finds the applicant’s approach consistent with SRP Section 19.0, SECY-90-016 and SECY-93-087, and therefore, acceptable. Based on the review of the DCD, the staff has confirmed incorporation of the revised Figure 19.1-50; therefore, RAI 434-8352, Question 19-92, is resolved and closed.

One of the Commission’s goals, as stated in the SRM in response to SECY-90-016, is that the LRF should be less than 1E-6 per year. The DCD was not clear on how the APR1400 design meets this goal. Therefore, in a public meeting on April 13-15, 2015, the staff requested that the applicant show how the design meets this Commission’s goal. On September 11, 2015, the applicant responded showing that the LRF summed from all modes of operation was 5.65E-7 per year, which is below 1E-6 per year and therefore meets the Commission’s goal (ML15254A448). The applicant committed to update DCD Tier 2 Section 19.1.8 with this change. The applicant’s response to RAI 434-8352, Question 19-92, resulted in changing the total LRF from all modes of operation to 6.1E-7 per year and included these changes in a DCD markup. The staff reviewed the changes and found that the revised LRF is also less than the Commission’s goal of less than 1E-6 per year, and therefore, acceptable. Based on the review of the DCD, the staff has confirmed incorporation of the changes in DCD Section 19.1.8; therefore, RAI 434-8352, Question 19-92, is resolved and closed.

The staff review finds the results from Level 2 internal events PRA for operations at power is consistent with SRP Section 19.0 and RG 1.200 guidance, and therefore, acceptable.

Containment Ultimate Pressure Capacity Analysis

The staff reviewed DCD Section 19.1.4.2.1.2.2, and audited technical report APR1400-K-P-NR-013605, Rev. 0, “Ultimate Pressure Capacity Analysis,” dated July 31, 2013. As described in Section 4.3 of that report, the fragility is formulated as the product of the median internal pressure capacity at which failure occurs and a random variable that represents uncertainty in the median pressure capacity. Once the potential failure modes are identified, failure criteria are established, from which the median capacities are evaluated by conducting independent limit state analyses using specified failure criteria with the applied loading. The applicant stated that the median capacities are generally evaluated using existing analyses or simplified calculation methods based on equilibrium, compatibility, and basic shell theory. The uncertainties considered in the analysis include modeling and strength uncertainties. Examples of the source of modeling uncertainty include assumptions used to develop internal force redistribution, failure criteria, the use of empirical formulae, variabilities in tendon and reinforcing steel placement, and variability in force-deformation relationships. Examples of sources of strength uncertainties include variability in concrete strength, tendon, and reinforcing steel strength; steel liner strength; and the influence of elevated temperature on material strengths.

The staff's review of the applicant's containment pressure fragility evaluation found the failure modes considered in the evaluation and the determination of controlling failure modes for leakage and rupture to be reasonably established in view of the conclusions in NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories - An Overview," issued July 2006. Further, the staff's review found the resulting median pressures and respective fragilities for the controlling leakage and rupture failure modes and the total containment fragility to be comparable to the median pressures and fragilities presented in NUREG/CR-6906. On this basis, the staff found the applicant's median pressure and fragility estimates to be acceptable.

However, the staff found that DCD Tier 2 Section 19.1.4.2.1.2.2 lacked a description of the applicant's analytical process used to determine pressure fragility, consideration of SA temperature effects on material properties, and consideration of uncertainties. Therefore, on March 8, 2016, the staff issued RAI 433-8363, Question 19-83, to request the applicant provide additional descriptions in the DCD (ML16068A099). The applicant submitted a response on June 23, 2016, which supplemented the DCD with additional information on the containment fragility analysis (ML16176A375). The applicant addressed concerns regarding the SA temperature (i.e., 204 degrees Celsius [400 degrees Fahrenheit], conservatively estimated from a broad spectrum of accident sequences described in DCD Section 19.2.3.3.7.2.1 "Bounding Temperature Environment"), provided the basis for the temperature reduction factors used in its analysis, and described the uncertainties associated with the median capacities and how these were aggregated. The staff reviewed the applicant's response to Question 19-83, and determined that the proposed additional information for the DCD is consistent with the applicant's technical report APR1400-K-P-NR-013605 and provides an adequate level of detail for the DCD pertaining to the containment pressure fragility analysis. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the RAI response. Accordingly, the staff considers RAI 433-8363, Question 19-83, resolved and closed. On May 1, 2018, the applicant supplemented the response to correct Table 19.1-28a to be consistent with DCD Section 3.8.1.4.11. Based on the review of the DCD, the staff has confirmed incorporation of the changes in DCD Table 19.1-28a; therefore, RAI 434-8352, Question 19-92, is resolved and closed.

The containment performance is evaluated in Section 19.2.4 of this report.

19.1.5 Safety Insights from the External Events PRA for Operations at Power

The applicant addressed the following external hazard groups:

- Seismic

- Internal fire
- Internal flood
- Other external events.

19.1.5.1 Seismic Risk Evaluation At-Power Operation

Summary of Application

The APR1400 seismic risk was evaluated using PRA-based SMA. The starting point to perform PRA-based SMA is to select a review level earthquake (RLE). The RLE for the APR1400 is 0.5g which is 1.67 times the safe shutdown earthquake (SSE). The seismic equipment list (SEL) is developed from the IEs PRA model and provides a documented list of SSCs that could be used to respond to an earthquake or mitigate potential plant damage initiated by a seismic event. Not all SSCs included in the IEs PRA are included in the SEL. For example, many balance-of-plant components are not considered in PRA-based SMA because they depend on offsite power, which is considered to be unavailable after a seismic event. Some SSCs such as distribution systems (piping, cable trays, ventilation ducts) and structural items (masonry block walls) are not modeled explicitly in the IEs PRA but are considered in the PRA-based SMA and SEL. Initiating events due to a seismic event are identified based on the IEs PRA as well, with two major differences: seismic events may damage passive plant components and structures that are not explicitly modeled in the IEs PRA and seismic events may simultaneously damage multiple SSCs in the plant. All seismic events are assumed to cause a LOOP. All initiators by a total loss of a supporting system in the IEs model are considered in the seismic event analysis. LOCA events are also considered to be induced by a seismic event. The system models are developed from the IEs PRA model to include random failures and human errors and to identify important accident sequences. The APR1400 used the Conservative Deterministic Failure Margin (CDFM) approach to calculate the structure, system, and component HCLPF capacity. This is done to demonstrate a seismic margin over the design-specific CSDRS. HCLPF capacities are described in DCD Section 19.1.5.1.2.2. The APR1400 used the "min-max" method to evaluate the plant seismic margin as described in NUREG/CR-4482, "Recommendations to the NRC on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants." This method was used since detailed plant-specific data is unavailable at the DC stage. Part 5, "Requirements for Seismic Events at-Power PRA," of the ASME/ANS PRA Standard is used in guiding the APR1400 PRA-based SMA. The applicant demonstrated seismic margin in the design by developing SEL, demonstrating plant level HCLPF, and developing risk insights for core damage and containment failure. The COL applicant will confirm and update APR1400 PRA-based SMA from the site-specific information. Uncertainties were explicitly considered in the fragility development. No sensitivity studies were conducted since PRA-based SMA is primarily qualitative.

Technical Evaluation

The staff recognizes that it is not practical for a DC applicant to complete a seismic PRA because a DC application would not contain site-specific seismic hazard information. As an alternative approach to a seismic PRA, the staff proposed in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, and the Commission approved in the corresponding SRM, dated

July 21, 1993, a PRA-based SMA. The results and insights from the PRA-based SMA are evaluated with respect to the Commission's objectives for new reactor designs described in the SRMs on SECY-89-102, SECY-90-016, and SECY-93-087, including the systematic evaluations of the risk associated with the design.

The staff conducted its review of APR1400 PRA-based SMA using the applicable guidance from the following:

- SECY-93-087 and the related SRM, 1993
- SRP Section 19.0, 2007
- DC/COL-ISG-020, 2010
- "Seismic Probability Risk Assessment Implementation Guide," EPRI 1002989, 2003
- "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," NUREG/CR-4334, 1985
- "Recommendations to the NRC on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants," NUREG/CR-4482, 1986
- "Methodology for Developing Seismic Fragilities," EPRI TR-103959, 1994
- "Requirements for Seismic Events At-Power PRA," Part 5 of ASME/ANS RA-Sa-2009, 2009.

The findings that the staff makes in the seismic-related sections of this SER (19.1.5.1 and 19.1.6.3) are established in accordance with the above guidance. Although several tasks and supporting analyses are necessitated to complete the PRA-based SMA, this SER focuses on the review of key technical areas that drive the results and insights.

The principal objective of PRA-based SMA is to provide an understanding of significant seismic vulnerabilities and insights to demonstrate the robustness of a standard design. In this context, the staff focused on (1) the framework for assessing potential significant failures induced by seismic events (2) the basis for establishing the SEL, which identifies essential equipment and structures to respond and mitigate the seismic accidents, and (3) seismic fragility evaluation for determining sequence-level and plant-level HCLPFs. The staff's review ensures that the technical adequacy of the APR1400 PRA-based SMA is sufficient to justify the plant-level seismic capacity and to identify risk insights that are used to support the DC application.

The staff's initial review of DCD Chapter 19, Revision 0, observed that the applicant did not identify all applicable accident sequences and address all plant operating modes. The applicant limited its seismic assessment to only those scenarios and sequences resulting from seismic events during at-power operation. The seismic evaluation during LPSD modes is entirely excluded from the application. DC/COL-ISG-020 specifies that all at-power, low-power, and shutdown modes should be considered for the selection of seismic sequences leading to core damage or containment failures. Therefore, on September 30, 2015, in RAI 232-7864, Question 19-10, the staff requested KHNP to update the DCD to provide the seismic assessment during LPSD conditions (ML16203A437). On April 11, 2018, the applicant submitted the final response to Question 19-10 (ML18101A794). The staff's review of that response is documented in Section 19.1.6.3 of this report

The staff's review also found the APR1400 at-power PRA-based SMA provided in DCD Revision 0 unacceptable due to missing and unjustifiable technical information. On March 8, 2016, the staff issued RAI 434-8352, Question 19-85 requesting the applicant to address the staff's concerns including the approach taken and related insight from a risk assessment standpoint (ML16068A187). On April 11, 2018, the applicant submitted the final response to Question 19-85 (ML18101A155). The staff's review and its basis for accepting the response to Question 19-85, are provided as follows.

Methodology

DCD Section 19.1.5.1.1 describes the APR1400 PRA-based SMA methodology to estimate the seismic margin and identify vulnerabilities of the design. The staff finds all steps performed and described in this section are reasonable and acceptable in conformance with the SRP Section 19.0. Since certain design details are not available at the DC stage, per SRP guidance, it is acceptable to perform the assessment using conservative and simplifying approaches, to the extent possible. The staff verified during the audit that the approach taken to complete the seismic evaluation satisfied the SRM on SECY-93-087 by providing the plant level HCLPF and seismic-related insights. The applicant's methodology is consistent with standard practice for assessing the seismic margin and identifying risk insights as described in SRP Section 19 and other seismic-related guidance. Based on its review, the staff finds the applicant has established a sufficient methodology for performing PRA-based SMA.

Seismic Hazard Input

In accordance with DC/COL-ISG-020, the seismic fragility calculation should use the response spectrum shape defined as the DC's certified seismic design response spectra (CSDRS). The applicant discussed the CSDRS for the APR1400 and peak ground acceleration (PGA) in the DCD Tier 2 Section 3.7.1, "Seismic Design Parameters," and depicted them in DCD Tier 2 Figures 3.7-1 "Horizontal CSDRS" and 3.7-2 "Vertical CSDRS." The applicant selected a RLE of 1.67 times the CSDRS for the PRA-based SMA. The RLE is anchored to a PGA of 0.5g.

The staff reviewed the selection of the RLE and determined that the spectrum shape described in DCD Tier 2 Section 19.1.5.1.1, "Description of the Seismic Risk Evaluation," satisfies the recommendation of SECY-93-087 and DC/COL-ISG-020. The applicant provided the definition of the response spectrum shape and magnitude used for the fragility analysis of SSCs, accident sequences, and the plant that the staff finds is consistent with DC/COL-ISG-020. Therefore, the staff finds that the seismic input is acceptable.

Seismic Equipment List

The staff acknowledges that the SEL is essential for demonstrating completeness of the seismic initiated sequences and to forming a foundation for the seismic PRA development. This list establishes the basis for the seismic fragility analysis and systems analysis. The staff reviewed the applicant's process and DCD Table 19.1-42, "Seismic Equipment List" and finds that the APR1400 SEL satisfies the key concepts in development of the list as described in SRP Section 19.0 and other seismic-related guidance.

The staff finds that the APR1400 SEL includes all components considered in the IEs PRA model. Furthermore, the applicant properly examined and considered inclusion of SSCs that are not explicitly modeled in the IEs PRA. The applicant properly excluded several balance-of-plant components, such as main feedwater system, condensate system, condensate storage and transfer system, AAC diesel generator, and steam generator blowdown system, since they depend on offsite power, which is expected to be unavailable after a seismic event. Only those systems that can be supported by the onsite emergency AC power sources are considered. The APR1400 SEL also includes passive components whose seismic-induced failure could affect the safety functions modeled in the IEs PRA (i.e., piping, cable trays, tanks, ventilation ducts, cranes, structural items, etc.).

The staff verified that the APR1400 SEL contains those structures that house the PRA and passive components. The applicant completed the SEL by including structural failures that could cause widespread equipment failures, (i.e., reactor containment building, auxiliary building, turbine building, compound building, CCW heat exchanger building, ESW building, emergency diesel generator building, and diesel fuel oil tank building).

The staff compared the SEL compiled for APR1400 PRA-based SMA and other similar designs for completeness and finds the APR1400 list reasonable and acceptable.

Seismic Fragility Evaluation

The seismic fragility analysis characterizes the capacities of SSCs to withstand ground motion due to an earthquake. The fragility is expressed as the conditional probability of failure of an SSC as a function of earthquake size (typically characterized by PGA). The capacity of a component to maintain its safety function during and following strong ground motion must be estimated, taking into account the seismic response at the component's location in a structure. The resulting fragilities are represented by a HCLPF value, which represents a 95-percent confidence that the failure probability is less than 5 percent.

The staff reviewed DCD Tier 2 Section 19.1.5, "Safety Insights from the External Events PRA for Operations at Power," and audited Calculation 1-035-N392-304, "Seismic Fragility Analysis," to determine the acceptability of the applicant's seismic fragility evaluation. The applicant made Calculation 1-035-N392-304 available in response to the staff's April 9, 2015, request for fragility calculations (ML15099A710), and for the basis and justification of assumed HCLPF values (including screened out components). The applicant provided Revision 2, dated March 31, 2016, and Revision 3, dated June 30, 2016 (addressing stability of the EDGB and diesel fuel oil tank room), for staff's audit. Specifically, the applicant provided conservative deterministic failure margin (CDFM) calculations for the following SSCs:

- Building Structures
    - Reactor Containment Building – Containment
    - Reactor Containment Building – Containment Internal Structure
    - Auxiliary Building
    - Emergency Diesel Generator Building
    - Diesel Fuel Oil Tank Building
- Reactor Coolant Systems Components

  - Reactor Pressure Vessel
  - Reactor Vessel Internals
  - Control Element Drive Mechanism
  - Reactor Coolant Pumps
  - Steam Generators
  - Pressurizer
  - Reactor Coolant Loop (RCL) Piping
- Global Stability of Structures.

The staff audited these fragility calculations to assess the validity of the PRA-based SMA. The staff confirmed that the calculation results are consistent with DCD Tier 2 Table 19.1-43, “Seismic Fragility Analysis Results Summary.” The staff’s assessment of the methodology described in the DC is discussed below.

The staff reviewed the applicant’s description of the seismic fragility analysis in DCD Tier 2 Section 19.1.5.1.1.5, “Seismic Fragility Analysis,” Section 19.1.5.1.2.2, “Seismic Fragility Analysis Results,” and the associated results provided in DCD Table 19.1-43. The staff determined that more information was required to explain the methodologies and references used to develop the seismic fragilities. The staff issued RAI 433-8363, Question 19-75, on March 08, 2016, to confirm whether, consistent with DC/COL-ISG-020, the applicant used the CDFM (EPRI NP 6041 “A Methodology for Assessment of Nuclear Power Plant Seismic Margin”) and/or the separation of variables approach (EPRI TR 103959 “Methodology for Developing Seismic Fragilities”) to determine the seismic fragility or to justify an alternative method as applicable (ML16068A099). The staff also requested the applicant describe any generic data or assumptions (e.g., failure modes, capacity and response factors, and associated uncertainties) used to develop the HCLPF capacities for seismic Category I structures.

In the response to RAI 433-8363, Question 19-75, submitted on August 1, 2016, the applicant stated that the CDFM approach is used for the seismic fragility analysis in the DC application (ML16214A075). The applicant revised DCD Tier 2 Section 19.1.5.1.1.5 to clarify the use of the CDFM approach for the seismic fragility analysis. The applicant provided a revised Table 19.1-43, which provides the HCLPF capacities derived using the CDFM approach for the buildings and RCS components.

Additionally, the calculation audited by the staff specified that the HCLPF capacity was calculated using code-specified design capacities and CSDRS demand. Material properties consistent with EPRI NP-6041 were specified for each material. These properties are consistent with those presented in DCD Tier 2 Section 3.8, “Design of Category I Structures.” An inelastic energy absorption factor was applied appropriately. For components (e.g., reactor pressure vessel [RPV], reactor vessel internals, control element drive mechanism, RCPs, SG subcomponents), consistent with EPRI NP-6041, the Level D allowable and Level D stress were used to demonstrate the appropriate HCLPF capacities. The earthquake component was separated from the Level D stress for the CDFM approach. The seismic demands used as input to the design of SSCs resulted from the envelope of eighteen seismic analysis cases. These twenty cases include 9 soil profiles for both cracked and uncracked concrete, as discussed in Section 3.7 of this report.

The staff reviewed the applicability of the CDFM approach and confirmed the use of this approach in the “Seismic Fragility Analysis” calculation. Consistent with DC/COL-ISG-020, the staff finds the use of CDFM (EPRI NP-6041) acceptable for determining the HCLPF capacity of SSCs. RAI 433-8363, Question 19-75, is resolved and closed.

The capacities for the balance of plant (BOP) components and site-specific structures listed below were not derived using the CDFM approach. Instead, HCLPF capacities were assumed for these SSCs.

- BOP Components (listed in Table 19.1-43)
- ESWIS
- CCW Hx Building
- Turbine Building
- Compound Building

The staff’s review of DCD Tier 2 Section 19.1.5 determined that the application lacked a basis for the assumed HCLPF values for the site-specific structures and BOP components. Therefore, on March 8, 2016, the staff issued RAI 433-8363, Questions 19-73 and 19-74, to request the bases and justifications for the assumed HCLPF values (ML16068A099). The staff further discussed the applicant’s justification for the assumed HCLPF values during public calls on June 20, 2016 (ML18128A036), and July 21, 2016 (ML18128A055).

The applicant submitted a response to RAI 433-8363, Questions 19-73 and 19-74, on August 1, 2016 (ML16214A068). Some of the information provided in this response was subsequently revised in the response to RAI 434-8352, Question 19-85 (ML18101A155). The responses revised Item COL 19.1(8) to include separate commitments for the BOP components and the site-specific structures.

With respect to the BOP components, in its response to RAI 433-8363, Question 19-74, the applicant added a description to DCD Section 19.1.5.1.1 “Description of the Seismic Risk Evaluation” and Table 19.1-4, “Risk Insights from PRA Models,” which explains the basis for assuming a HCLPF capacity of 1.67 x CSDRS = 0.5 g is reasonable (ML16214A068). The applicant demonstrated by reference to Appendix E of EPRI NP-1002988 “Seismic Fragility Application Guide” that equipment designed for 0.25 g SSE can have a HCLPF that exceeds 0.5 g by considering conservatism in the design process.

The staff endorsed the reference documents EPRI NP-6041 and EPRI NP-1002988 in DC/COL-ISG-020 as acceptable guidance for conducting a PRA-based SMA. The endorsed reference documents provide justification that the generic fragility data are consistent and applicable to SSCs within the scope of the certified design application. Therefore, the staff concluded that the basis provided by the applicant for assuming the BOP components are able to meet or exceed a HCLPF capacity of 1.67 times the CSDRS is acceptable. As indicated in COL Information Item, COL 19.1(8), the COL applicant will confirm that the DCD PRA-based SMA is bounding for the selected site, and update the PRA-based SMA to include site features, design departures, site-specific SSCs and soil effects (including sliding, overturning liquefaction, and slope failure). Therefore, the commitment to demonstrate that the HCLPF capacity is equal to or exceeds 1.67 times the CSDRS for the BOP components, as described in COL 19.1(8) is acceptable to the staff.

For the site-specific structures (i.e., the ESWIS, CCW Hx building, turbine building, and compound building), the applicant has assumed a HCLPF capacity with a discrete value of 0.5g, which is equivalent to the commitment of 1.67 times the CSDRS. In COL 19.1(8), the applicant states that the COL applicant is to demonstrate that site-specific structures have a HCLPF capacity that is equal to or greater than 1.67 times the ground motion response spectra (GMRS) and will update the PRA-based SMA with the site-specific HCLPF values. Because absent the GMRS, which will be determined at the COL stage, this assumption cannot be justified at this time, the applicant has provided a statement in footnote [4] to Table 19.1-43, that the assumed HCLPF values are for quantification purposes. The staff has determined that the assumption of a HCLPF capacity of 0.5 for the site-specific structures is acceptable on the basis that the assumed HCLPF capacities are not certified values, and that the COL applicant is to evaluate and update the PRA-based SMA and ensure that the results of the PRA-based SMA remain valid and reflect the site-specific effects and plant-specific features.

In the revisions to COL 19.1(8) in response to RAI 433-8363, Questions 19-73 and 19-74 and RAI 434-8352, Question 19-85, the applicant also included the following commitments for the COL applicant:

- To demonstrate that the equipment and relays qualified by testing remain functionally operational within 1.67 times the required response spectra described in DCD Section 19.1.5.1.1.
- To demonstrate that the components identified as inherently rugged in DCD Section 19.1.5.1.1.2 have seismically rugged capacity.
- To demonstrate that the steam generator tube HCLPF is higher than the steam generator nozzle HCLPF.

The staff determined that the assumptions above are acceptable on the basis that the COL applicant is to confirm the applicability of these assumptions and is to evaluate and update the PRA-based SMA and ensure that the results of the PRA-based SMA remain valid and reflect the site-specific effects and plant-specific features. RAI 433-8363, Questions 19-73 and 19-74, are resolved and closed.

Systems and Accident Sequence Analysis

The staff reviewed the systems and accident sequence analysis to ensure that:

- Potential seismic-induced initiating events are properly identified
- Combination of plant system responses for event mitigation are properly analyzed
- Passive components (e.g., tanks, heat exchangers, and piping) and structures are properly modeled
- Simultaneously damage of multiple redundant systems and components are properly considered.

Initiating Events Analysis

Based on the staff's audit of the applicant's models, the staff finds that the applicant properly analyzed and modeled both seismic-induced initiating events and multiple initiating events resulting from a same seismic event. The staff finds that the applicant properly modeled plant transient, such as LOOP, loss of 4kV AC bus, and SBO. The applicant properly identified supporting system failures, (i.e., failures of engineered safety feature - component control system (ESF-CCS), group controller, plant protection system, etc., as potential initiators). The applicant considered all LOCAs of various sizes, in all relevant systems, and at different locations.

The staff compared the APR1400 seismic-induced initiating events against other similar designs, and based on its review, the staff finds the APR1400 initiating events analysis reasonable and acceptable.

Evaluation of Safety Functions

The staff verified that in performing PRA-based SMA, the applicant properly evaluated all key safety functions that must be maintained to prevent core damage and large release. These functions include reactivity control, core heat removal, reactor coolant inventory control, reactor coolant heat removal, and containment integrity as mentioned in the ASME/ANS PRA Standard. For those front-line and support systems that are required to meet the safety functions, the applicant properly considered and modeled only those systems that do not require offsite power. No credit was taken for nonsafety systems because they are not seismic Category I and thus they become unavailable as a consequence of a seismic event. For these reasons, the staff finds the applicant has identified and analyzed the safety functions properly.

Plant Response and Mitigation Systems

The purpose of the plant response analysis is to develop a model that examines the initiating events and other failures resulting from the effects of the seismic hazard that could lead to core damage or large release. From this perspective, the IEs event trees and fault trees are modified to accommodate seismic events. In this way, the random failures and human errors modeled in IEs PRA are captured in the analysis.

The staff reviewed the APR1400 plant response and mitigation analysis and finds it reasonable and consistent with the IE PRA. As illustrated in Figures 19.1-48A, "At-Power Seismic Event Tree," 19.1-48B, "At-Power Seismically Induced Small LOCA Event Tree," and 19.1-48C, "At-Power Seismically Induced LOOP Event Tree," the plant response logics were developed based on the at-power IEs PRA model to incorporate those different aspects because of the seismic hazard's effects. The staff finds that the applicant has properly developed event trees and modified IEs fault trees that were used to generate core damage and large release sequences from seismic event, coupled with the random failures not related to the ground motion and human errors.

The staff finds that the applicant properly developed and used the IEs event trees and fault trees to conform to a seismic event, including the addition of seismic-induced failure events and human errors in its model. These modifications were reasonably performed consistent with the guidance, and therefore acceptable.

Structure Failures

The staff reviewed the APR1400 assessment of structure failures caused by a seismic event and verified the information against other DCD sections. In its analysis, the seismic Category I structures that contain equipment credited in the IEs PRA and Category II structures that could impact other structures that contain equipment credited in the PRA, were examined and added to the SEL. Failure of these structures was assumed to cause widespread equipment failures and result in failure of all components in that building. Consequently, it was conservatively assumed that failure of these structures would lead directly to core damage. Seismic fragilities were calculated or assumed as described in the Seismic Fragility Evaluation section above.

The APR1400 PRA-based SMA identified six structures that failure of any one would directly lead to core damage due to the catastrophic failure of RCS and/or safety-related components, as follows:

- Containment building
- Auxiliary Building
- Turbine Building
- Compound Building
- Emergency Diesel Generator Building
- Diesel Fuel Oil Tank Building.

Based on its review and verification against other DCD sections, the staff finds the applicant has logically identified and modeled structure failures and therefore, it is acceptable.

Correlation between Seismic Failures

The staff finds that the APR1400 PRA-based SMA examined the potential system correlation and dependencies between SSCs and conservatively assumed all redundant systems are 100% correlated; (e.g. if one emergency diesel generator fails, they all do). DCD Table 19.1-42, “Seismic Equipment List,” shows the correlation factors for all SSCs in the SEL. Similar components that are located in the same building and same elevation are treated as fully correlated. To account for the correlation and dependencies, the correlation factor between SSCs was assigned by either zero or 1.0, fully independent or fully dependent, respectively.

The staff acknowledged that, during design stage, the overall state-of-knowledge about the amount of dependency/correlation among seismic-induced structures and component failures is limited. Therefore it is acceptable to assume full response correlation when similar items are co-located and zero correlation whereas if the components are quite different or found in very different locations. The staff finds that the APR1400 assignment of response correlation with either a 1.0 or zero factor is acceptable because it is a conservative approach for addressing correlation.

Key Assumptions

DCD Section 19.1.5.1.1.4.9 provides the key assumptions used in APR1400 PRA-based SMA. In general, these seismic-related assumptions are conservative to cover the uncertainty in design and operation due to limited information available at the DC stage. The staff evaluated the impact of the uncertainties in these assumptions on the resultant risk insights in accordance with SRP Section 19.0 guidance. From the standpoint of risk evaluation, the staff finds these assumptions to be reasonable since they help ensure the risk insights are rationally identified but not underestimated.

Based on its review, the staff finds that the key assumptions used in the APR1400 PRA-based SMA are reasonable for the DC phase. Furthermore, per COL Information Items 19.1(4) and 19.1(8), adequate provisions for these assumptions are to be reevaluated and dispositioned during the COL phases to ensure that the results and insights continue to remain valid.

Sequence-Level and Plant-Level HCLPF Assessment

The staff finds that the applicant chose the "min-max" method for computing sequence-level and plant-level HCLPF values. DC/COL-ISG-020 states that "min-max" is an acceptable method to determine the accident sequence level and plant level fragilities. In the min-max method, input fragilities are combined by using the lowest (minimum) HCLPF value of a group of inputs operating in an OR logic, and by using the highest (maximum) HCLPF value of a group of inputs operating in an AND logic. Random/human failure probabilities are also reported in combination with HCLPFs for each accident sequence.

DCD Tables 19.1-44A through 19.1-44D provide the combination of HCLPFs and random failure probabilities of the analyzed accident sequences. The risk was addressed by showing that there is adequate margin in the plant seismic design to 1.67 times the SSE (i.e., 0.5g PGA). The staff reviewed the accident sequences in the above tables and finds them acceptable. They were generated consistent with the APR1400 fragility evaluation, plant responses, and key assumptions. The staff reviewed these tables and confirmed that no accident sequence has a HCLPF lower than 0.5g. The sequence-level and plant-level HCLPF assessment has provided confidence that the APR1400 design will withstand an earthquake of at least 0.5g intensity and achieve safe shutdown without damage to the reactor core.

Results from At-Power Seismic Risk Evaluation

According to SRP Section 19.0, the staff expects the products of APR1400 PRA-based SMA to include: (1) evaluation of sequence-level and plant-level HCLPFs, (2) identification of the limiting structure/component HCLPFs from the assessment of core damage and containment failure sequences, (3) examination of plant responses and consequences, and (4) development of seismic-related insights. The staff reviewed the response to Question 19-85 and information presented in DCD Chapter 19 and, based on the staff's audit of the applicant's seismic assessment, finds that the applicant fully addressed the above expectations.

The applicant performed its PRA-based SMA using the preferred approach in SECY-93-087 and associated SRM. The applicant demonstrated that the APR1400 design can withstand a review level earthquake of 1.67 times the CSDRS based on the calculated plant HCLPF capacity of

0.5g. Through the PRA-based SMA, the applicant identified significant accident sequences and dominant contributors to both core damage and containment failure in accordance with the Commission's objectives for the DC.

The seismic fragilities of the APR1400 SSCs are based on (1) design-specific seismic capacity calculations for certain structures and RCS components, (2) generic industry information, and (3) assumptions. Based on the estimated fragilities and generated accident sequences, the staff confirmed that the APR1400 plant HCLPF value is at least one and two-thirds the ground motion acceleration. The staff concludes that if all key assumptions remain valid and all seismic-related COL information items are fulfilled, the APR1400 design meets or exceeds the 0.5g HCLPF capacity.

The APR1400 PRA-based SMA identified dominant accident sequences for seismic events, irrespective of their likelihood. Although the APR1400 PRA-based SMA does not support the determination of important contributors to seismic risk in a probabilistic sense, it allows identification of the plant features that are important to the plant level HCLPF value. These insights are documented in DCD Section 19.1.5.1.2.4. The results did not identify any human reliability insights that were not already recognized in the internal events analysis.

Based on its review, the staff finds that the APR1400 plant HCLPF will be at least one and two-thirds times the SSE. All of the sequence-level HCLPF values are verified to exceed the 1.67 times SSE that represents the seismic robustness of the plant. As stated in the COL information items, COL applicants will update the PRA-based SMA to include site features, design departures, site-specific SSCS, and soil effects and ensure that the results of the results of the PRA-based SMA remain valid for the APR1400 design. The capacity of as-built SSCs will be confirmed by a seismic walkdown after construction. The staff also finds the APR1400 seismic-related insights practical and useful for DC and COL stages.

Based on the staff's evaluation described above, the APR1400 design satisfies the expectation of the SECY-93-087 regarding seismic evaluation and capacity. Therefore, the staff concludes that the APR1400 PRA-based SMA is acceptable.

Sensitivities and Uncertainties

Since the PRA-based SMA method does not quantify risk, neither uncertainty analysis nor sensitivity analysis was performed. This is acceptable according to SRP Section 19.0.

Combined License Information

Table 19.1.5-1 below lists the PRA-based SMA-related COL information item numbers and descriptions from DCD Tier 2 Table 1.8-2.

Table 19.1.5-1. Combined License Items Identified in the DCD

Item No.	Description	Section
COL 19.1(4)	The COL applicant and holder are to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA (including PRA inputs to RAP and SAMDA) remain valid with respect to internal events, internal flood and fire events (fire barrier and fire barrier penetrations, routing and locations of pipe, cable, and conduit), and HRA analyses (development of operating procedures, emergency operating procedures, and severe accident management guidelines and training), external events including PRA-based seismic margins, HCLPF fragilities, seismic spatial interactions and LPSD procedures.	19.1.4

Item No.	Description	Section
COL 19.1(8)	The COL applicant will confirm and update from new information from the site, e.g., site features, design departures, etc., that the PRA-based SMA is bounding for the selected site, site-specific SSC and soil effects, including sliding, overturning, liquefaction, and slope failure. The COL applicant is to confirm that the as-built plant has adequate seismic margin and does not exceed the CDF and LRF design targets specified in Subsection 1.2.1.1.1e. The COL applicant is to demonstrate that site-specific structures (the turbine building, compound building, ESW IS and CCW HX buildings) have a HCLPF capacity that is equal to or greater than 1.67 times GMRS and will update the PRA-based SMA with the site-specific structure HCLPF values, accordingly. The COL applicant is to demonstrate that HCLPF capacity is equal to or exceed 1.67 times the CSDRS for BOP components and is to complete the SEL. The COL applicant is to demonstrate that the seismic capacity for equipment and relay qualified by testing should remain functionally operational within 1.67 times the required response spectra (CSDRS-based RRS) in the procurement specification. The COL applicant is to demonstrate that the inherently rugged components identified in DCD Section 19.1.5.1.1.2 have seismically rugged capacity. The COL applicant is to demonstrate that the steam generator tube HCLPF is higher than the HCLPF for the steam generator nozzle.	COL 19.1(8)

The staff finds that the applicant established the above COL information items in accordance with the SRP Section 19.0 and DC/COL-ISG-020, and therefore they are acceptable. The staff concludes that no additional COL information items relevant to APR1400 seismic evaluation are necessitated. Based on the review of the DCD, the staff has confirmed incorporation of all changes on the APR1400 seismic assessment; therefore, RAI 8352, Question 19-85, is resolved and closed.

19.1.5.2 **Internal Fire Risk Evaluation**

<u>Summary of Application</u>

Section 19.1.5.2 of the DCD describes the internal fire risk evaluation and its results. The application includes a description of the internal fire PRA (FPRA) and its results including any risk insights. The DCD states that the FPRA methodology is based on NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," and NUREG/CR-6850 Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements." Since certain design details were not available at the DC stage, the applicant performed the FPRA using conservative and simplifying approaches. The applicant performed the internal fire HRA based on the methodology described in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines."

The applicant reported the following internal fire CDF results:

5 percent value:	1.2E-6 per year
Mean value:	2.3E-6 per year
95 percent value:	4.3E-6 per year

The applicant reported the following internal fire LRF results:

5 percent value:	8.7E-8 per year
Mean value:	1.9E-7 per year
95 percent value:	3.7E-7 per year

<u>Technical Evaluation</u>

In accordance with SRP Section 19.0, the staff conducted its review to determine whether the technical adequacy of the FPRA is sufficient to justify the risk estimation and identification of risk insights that are used to support the DC application. To evaluate the technical adequacy, the staff reviewed the extent to which the applicant's FPRA information is consistent with the applicable methods described in NUREG/CR-6850. The staff recognized that the applicant either did not perform certain tasks or used simpler analyses than suggested in NUREG-6850, since certain design details (e.g., specifics of cable routing, ignition sources, and target locations) and operating procedures are unavailable at the DC stage. The staff focused on ensuring that the internal fire risk compares favorably against the Commission's goals, as described in SRP Section 19.0, and that sufficient risk insights are identified to support the DC application.

Key Assumptions

DCD Section 19.1.5.2.1.2 lists the key assumptions used in the FPRA. In general, the applicant appropriately used conservative assumptions to address the uncertainty in design and operation that is present at the DC stage. For example, the applicant did not perform any fire modeling due to lack of sufficient spatial information related to the ignition sources and their targets, including intervening combustibles. Therefore, for single-compartment fire analyses, the applicant assumed that all unsuppressed fires propagate throughout the entire compartment, damaging all PRA-credited equipment within. The applicant assumed that any fire in any fire compartment is at least sufficient to automatically trip or manually shutdown the plant, resulting in an initiating event (i.e., general transient). Due to the uncertainty in the effectiveness of administrative controls, the analysis also assumed that every fire compartment has, at least, an assigned transient ignition frequency. From the standpoint of risk estimation, the staff finds these assumptions to be acceptable since they help ensure the risk is not underestimated.

The staff evaluated the impact of uncertainties in key assumptions on any risk insights. A conservative bias in one part of the model may mask or distort the significance of another part of the model. The DCD did not provide sufficient evaluation of uncertainties associated with FPRA assumptions. Therefore, on February 23, 2016, the staff issued RAI 418-8348, Question 19-45 requesting the applicant to ensure that the impact of uncertainties in key assumptions is properly assessed in the DCD (ML16054A291). In its response, the applicant proposed a revision to the DCD with an evaluation of the assumption associated with main control room evacuation, which could have a large effect on the risk estimates as well as risk insights. To better understand the impact of this assumption, the applicant performed a sensitivity study decreasing the conditional core damage probability for main control room evacuation by an order of magnitude. The results showed that the effect on risk estimates are not significant, but importantly, turbine building fires that lead to loss of offsite power becomes the most important fire scenario. For other assumptions which may affect the risk results or risk insights, the applicant proposed to perform these evaluations as a part of the PRA maintenance and upgrade process as described in DCD Section 19.1.2.4. The DCD also includes provisions for the COL applicant to describe their PRA maintenance and upgrade program [COL 19.1(6)], and for the COL holder to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA remain valid [COL 19.1(4)]. The staff finds the applicant's approach adequate because it provided a sufficient evaluation of uncertainties associated with FPRA assumptions. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 418-8348, Question 19-45, is resolved and closed.

The FPRA model included equipment whose spurious operation may either cause a fire-induced initiating event or adversely affect the response of systems or operator actions required to respond to a fire. In lieu of detailed circuit analyses, the applicant assumed that fire damage to control cables would result in the worst-case failure mode with respect to system function for the affected component (e.g., failure to operate, or spurious operation). Since this is a conservative approach, the staff finds this acceptable. The applicant also assumed that fire damage to fiber-optic cables would result in a failure to operate the associated component, but would not cause a spurious operation, or prevent normally operating equipment from continuing to operate. Since spurious operation from fire damage to fiber optic cables is considered unlikely, the staff finds this assumption to be acceptable.

The applicant's FPRA assumed that certain cables have fire protection features to prevent damage or spurious operation of related components. However, the staff was unclear how the applicant will ensure that these design features will be reflected in the as-built plant, thereby ensuring that the associated FPRA assumptions remain valid. After a public meeting on May 4-6, 2016 (ML17081A449), the applicant provided additional information (ML17026A469) that identifies the cables in each fire compartment that have been assumed to be either protected, had their circuits rerouted or redesigned to prevent failure, or can be shown through detailed circuit analysis to not result in the modeled failure mode. The applicant established COL 19.1(25) to ensure that the fire protection features required for preventing fire-induced damage of the PRA-credited components will be properly incorporated in the design. The staff finds that the proposed COL information item will ensure that the cable protection assumptions in the FPRA will remain valid for the COL application and operational phases. Based on the review of the DCD, the staff has confirmed incorporation of COL Information Item 19.1(25); therefore, RAI 434-8352, Question 19-92, is resolved and closed.

The applicant estimated the fire ignition frequency for each identified ignition source and each fire compartment using the generic frequencies from EPRI TR-1016735, "Fire PRA Methods Enhancements: Additions, Clarifications, and Refinements to EPRI 1019189." However, the staff notes that NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database," published more recent data on fire ignition frequencies. Therefore, on February 23, 2016, the staff issued RAI 418-8348, Question 19-45, requesting the applicant to address the impact of ignition frequency uncertainty in the DCD (ML16054A291). RAI 418-8348, Question 19-45, was subsequently subsumed into RAI 434-8352, Question 19-92. In its response, the applicant proposed to perform these evaluations as a part of the PRA update process. DCD Section 19.1.2.4 describes the PRA maintenance and upgrade process and includes provisions to ensure the adequacy of the PRA model commensurate with its intended use. The DCD also includes provisions for the COL applicant to describe their PRA maintenance and upgrade program [COL 19.1(6)], and for the COL holder to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA remain valid [COL 19.1(4)]. Based on these considerations, the staff finds the applicant's approach adequate and RAI 418-8348, Question 19-45, is resolved and closed.

The applicant provided a qualitative assessment of the interactions between seismic and fire events. Since as-built plant information and post-seismic or post-fire safe shutdown procedures are unavailable at the DC stage, the applicant performed this assessment using available design and PRA information. The applicant identified Item COL 19.1(4) for the COL holder to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the design features and assumptions used in the PRA remain valid. The applicant noted that all equipment included in the PRA-based seismic margin assessment are located in a seismic Category I structure and the restraints on tanks and piping for flammable fluids and the fire protection system components are seismically rugged. Hence, the applicant concluded that the design of the APR1400 minimizes the potential for seismically induced fires to compromise post-earthquake safe shutdown capability. The applicant established Item COL 19.1(21) to ensure that outage procedures will appropriately consider the seismic ruggedness of temporary equipment and ignition sources and that fire protection equipment are adequate. The applicant also established Item COL 19.1(9) to consider the potential for multiple spurious alarms from photoelectric detectors following a seismic event when developing post-earthquake safe shutdown procedures. The staff finds that the applicant has provided sufficient qualitative assessment in conjunction with associated COL information items to address the potential interactions between seismic and fire events.

The staff reviewed the applicant's HRA performed to support the FPRA. The applicant evaluated the applicable HFEs using the approach described in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines." For FPRA, the applicant modified the HFEs developed for at-power internal events to account for the presence of a fire and made certain assumptions to account for the undeveloped fire procedures at this stage of the design. For example, the applicant assumed that at least one division of instrumentation would be available in the main control room and that procedures would be available to alert the operators of potentially inaccurate indication. Item COL 19.1(16) specifically addresses the need for the COL applicant or holder to develop procedures and operator training for reliance on undamaged instrumentation. In addition, Item COL 19.1(4) provides assurance that these HRA assumptions remain valid for the COL phases. Therefore, the staff finds the applicant's approach acceptable.

For local manual actions that could be affected by the fire, the applicant assumed that actions performed inside the same fire compartment as the postulated fire would fail. In addition, actions that require the operator to travel through the same fire compartment as the postulated fire within the first 8 hours would fail. Based on the uncertainties in design and operation that is present at the DC stage, the staff finds this treatment of operator actions outside the main control room (MCR) acceptable. However, the staff noted an assumption in the PRA audit documentation that stated that the operator action to perform feed and bleed is entirely a MCR action, which is inconsistent with HRA performed for internal events. The applicant had previously responded to RAI 418-8348, Question 19-47, on a related topic discussed below in this SER under "Scope of Fire-induced Initiating Events," and committed to revise this response to address the inconsistency. In its response dated August 25, 2017 (ML17237C042), the applicant proposed a revision to DCD Table 19.1-4 clarifying the need for the operators to access the rooms where the local manual actions take place and that the HRA accounts for potential internal fire or flood impacts on the operator's ability to perform feed and bleed. The staff finds the applicant's approach acceptable. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 418-8348, Question 19-47, is resolved and closed.

Based on the above, the staff finds that the key assumptions used in the FPRA are reasonable for the DC phase. Furthermore, adequate provisions for these key assumptions used in the DC PRA are to be appropriately evaluated and dispositioned during the COL phases to ensure that the PRA results and insights continue to remain valid.

Scope of Fire-induced Initiating Events

The staff reviewed the scope of fire-induced initiating events considered in the FPRA. The internal events PRA is used as a starting point to identify which internal event sequences to include in the FPRA model. The applicant eliminated certain event sequences that can be assumed not to occur as a direct result of a fire, such as LOCAs due to pipe breaks, vessel failure, steam generator tube rupture, and secondary system pipe breaks. The applicant eliminated anticipated transient without scram (ATWS) based on a low-frequency-of-occurrence argument. These eliminations are consistent with NUREG/CR-6850. However, the staff was unclear regarding the basis for the eliminating the spurious safety injection (SI) signal from fire-induced initiating events and accordingly, discussed the issue in a public meeting (ML15292A030). During the meeting, the applicant clarified that for full power operation, the SI pump discharge pressure is below the operating pressure and there would be no injection of fluid into the RCS. Therefore, a fire-induce spurious SI would not cause an initiating event. The staff evaluated this information and finds that the applicant has provided an adequate basis for screening fire-induced spurious SI as an initiating event.

The staff noted in the DCD that the applicant had screened out fire-induced ISLOCAs from the FPRA. The staff reviewed the associated basis in the PRA documentation during an audit and the staff discussed this issue in a public meeting (ML15292A030). In particular, the staff was unclear if the applicant considered all potential ISLOCA scenarios for the analysis. During the meeting, the applicant clarified that all containment penetrations identified for the full power internal events ISLOCA analysis were reexamined to determine if fire impacts would change the outcome of the screening analysis, and subsequent frequency of an ISLOCA in the FPRA. Since the applicant demonstrated that they reviewed all potential ISLOCA pathways for fire-induced spurious operation using appropriate screening criteria, the staff finds the exclusion of fire-induced ISLOCA scenarios acceptable.

The applicant determined that fire-induced spurious opening of the POSRVs is a possible mechanism leading to medium LOCA. However, the applicant determined that fire-induced opening of a POSRV and the associated isolation valve is not credible. On February 23, 2016, the staff issued RAI 418-8348, Question 19-47, asking applicant to provide the basis for excluding the fire-induced opening of POSRVs as an initiating event and to describe the design configuration and operating procedures assumed in the HRA for the feed and bleed operation (ML16054A291). In a response dated April 12, 2016, the applicant stated that the motor operated pilot valves are normally closed and the power to one of two valves for each POSRV is disconnected to prevent spurious operation during plant operation except during refueling and cold shutdown operation (ML16103A569). Manual operator action is needed to reconnect power. The applicant also confirmed that the HRA considered the APR1400 emergency operating guideline (EOG) for feed and bleed operation, including any local manual action. Since the power to the motor is removed the staff finds it acceptable to exclude fire-induced opening of POSRVs as an initiating event from the full-power FPRA.

Separately, the staff noted an assumption in the PRA audit documentation that stated that the operator action to perform feed and bleed, which uses the POSRVs, is entirely a MCR action, which is inconsistent with HRA performed for internal events. During a public meeting on September 27, 2016 (ML16274A253), the applicant committed to revise its response to RAI 418-8348, Question 19-47, to address this inconsistency (ML16103A569). In the response, the applicant proposed a revision to DCD Table 19.1-4 correcting the inconsistency. Specifically, the applicant clarified that the operators would need to access the rooms where the local manual actions take place and that the HRA accounts for potential internal fire or flood impacts on the operator's ability to perform feed and bleed. The staff finds the applicant's approach acceptable because it corrects inconsistencies. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 418-8348, Question 19-47, is resolved and closed.

The DCD identifies fire-induced loss of dc bus A (LODCA) and loss of dc bus B (LODCB) as fire-induced initiating events, but the staff was unclear whether loss of dc bus C or D can still lead to a transient. The staff, in a public meeting (ML15292A030), discussed the treatment of fire-induced loss of dc bus C or D. During the meeting, the applicant clarified that a fire damaging only dc bus C or D does not directly lead to plant trip but is conservatively treated as a fire-induced general transient with fire damage to the dc bus. The staff finds that this approach is conservative and that the applicant has provided adequate basis for excluding fire-induced loss of dc bus C or D as an initiating event from the FPRA.

Based on the analysis above, the staff finds the applicant's identification of fire-induced initiating events acceptable.

Scope of Fire Scenarios

The staff reviewed the scope of fire scenarios considered in the FPRA. The APR1400 FPRA partitioned the plant into 390 discrete fire compartments, based on the fire hazard analysis information. The applicant performed a slightly more detailed analysis on 38 out of 390 fire compartments, including the main control room, reactor containment building, and turbine building. Hence, 352 remaining compartments are full-compartment burnout scenarios, which assumed that every ignition source in a compartment would damage every PRA-credited component and cable within that compartment. For those fire compartments found to be potentially risk-significant, the applicant refined the fire analysis, which considered factors such as compartment size, gross knowledge of ignition source and target locations within the fire compartment, and suppression systems. The applicant's identification of fire scenarios is sufficiently consistent with guidance in NUREG-6850 applicable to DC and SRP Section 19.0.

However, it was unclear to the staff whether the large (96,000 gallons) diesel fuel oil storage tanks (DFOSTs) in the auxiliary building have been evaluated as potential combustible sources that can significantly exacerbate a potential fire scenario occurring in their vicinity. On February 23, 2016, the staff issued RAI 418-8348, Question 19-46, requesting the applicant to address this potential fire scenario from a risk assessment standpoint (ML16054A291). In response dated May 31, 2016, the applicant provided additional information to show that potential fire scenarios involving the DFOSTs are not risk-significant (ML16152B019). The DFOSTs are designed and fabricated in accordance with ASME Section III and the concrete walls of the DFOST rooms are designed to seismic Category I criteria. The applicant confirmed that the room walls and floors are 3-hour fire rated, and estimated that the walls and floors are approximately 4 ft (1.2 m) thick. There are no additional rooms above the DFOST rooms. There are pre-action sprinkler systems in the DFOST rooms covering the diesel generator fuel oil storage tanks and fuel oil transfer pumps areas. Due to physical arrangement, a postulated fire would have to fail multiple fire barriers and automatic fire suppression systems to reach the main control room or the remote shutdown room. A scenario that breached a DFOST room wall or floor may result in damage in the same division but would not result in damage to the other division. Further, the applicant proposed a revision to COL 19.1(4) to also confirm that the fire barrier and fire barrier penetration assumptions in the FPRA remain consistent with as-designed and as-built information. The staff finds that the hazards associated with DFOSTs are sufficiently unlikely that they can be excluded from explicit modeling in the internal fire risk assessment. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the RAI response. Accordingly, the staff considers RAI 418-8348, Question 19-46, resolved and closed.

Based on the above analysis, the staff finds that the scope of fire scenarios considered to be acceptable.

Level 2 At-power Internal Fire PRA

The DCD does not describe the process used to quantify Level 2 PRA event trees for internal fire events. Therefore, in RAI 432-8377, Question 19-60, the staff asked the applicant to update the DCD with this information (ML16068A097). In the response to Question 19-60, dated October 31, 2016, the applicant verified that it evaluated internal fire Level 2 PRA using the same models developed for the at-power Level 2 internal events PRA (ML16305A440). The applicant proposed to revise the DCD as follows and attached the DCD markup to Section 19.1.5.2.1.3 in the response to Question 19-92:

> The at-power fire Level 2 PRA was evaluated using the same Level 2 models and methodology as was used in the at-power internal events Level 2 PRA (Section 19.1.4.2). The Level 1 fire PRA quantification was structured to evaluate each fire scenario (SCA and MCA) using internal events accident sequence top-logic, modified by scenario-specific flag files to fail applicable equipment. The Level 2 fire PRA quantification utilized the same approach, except that rather than solve the model using the core damage fault tree gate, an AND gate was developed between the core damage gate AND the STC fault tree logic. The resulting cutsets contain STC flag events, so each fire scenario is quantified with all of the STC cutsets included. The total at-power fire LRF is the sum of the SCA and MCA calculations for the STCs that are LRF (i.e., STCs 1, 6, 7, 8, 13, 20 and 21). Note that STCs 3 and 4 are LRF in the internal events model, but as they are exclusively ISLOCAs, their frequencies are zero in the at-power fire LRF quantification.

The staff reviewed the applicant's response and found that it described the process used to quantify internal fire Level 2 PRA. This process is acceptable as it uses the same at-power internal events Level 2 internal events PRA models. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 432-8377, Question 19-60, is resolved and closed.

DCD, Section 19.1.5.2.2, states that the applicant performed the internal fire risk evaluation using "the design-specific fire protection features in Chapter 9, Appendix 9A and the internal events PRA model of Subsection 19.1.4." In using the internal events PRA model, it was not clear to the staff how the applicant evaluated the systems and equipment that are important for Level 2 and affected by internal fire. Therefore, in RAI 432-8377, Question 19-57, the staff asked the applicant to explain and revise the DCD accordingly (ML16068A097).

In the response to Question 19-57, dated July 15, 2016, the applicant stated that it handled the impact on Level 2 systems and equipment from fire effects in the same way as it did for the Level 1 analysis (ML16197A401). Specifically, for both Level 1 and Level 2, the applicant assumed that all components credited in the fire PRA that are affected by the fires to fail their safe shutdown function. The applicant proposed changes to DCD Sections 19.1.5.2.1.3 and 19.1.5.2.2 clarifying that the fire PRA is based on both the Level 1 and Level 2 internal events PRA models. The staff finds the applicant's response acceptable because it addresses the staff's concern about how the applicant evaluated the systems and equipment that are important for Level 2 and affected by internal fire and demonstrated that the evaluation is conservative. The staff confirmed that the applicant incorporated these changes in DCD Tier 2 Revision 1. Based on applicant's response, RAI 432-8377, Question 19-57 is resolved and closed.

DCD, Tables 19.1-29 and 19.1-30, provide the summary of source term evaluation and source term category frequencies and contributions to LRF for internal events. However, the DCD does not provide similar information for internal fire and internal flooding. Therefore, in RAI 432-8377, Question 19-56, the staff asked the applicant to update the DCD providing similar tables for internal fire and internal flooding. In response, in a letter, dated July 29, 2016, the applicant clarified that DCD Table 19.1-29, "Summary of Source Term Evaluation," shows the summary of source term evaluation not only for internal events, but also for internal fire and flooding events as follows (ML16211A404):

> In the APR1400 At-power Level 2 PRA, the accident scenarios for internal fire events and internal flooding events are assigned to the same STC (i.e., source term category) grouping logic which is used for internal events. Per the definition of source term category, a particular release category includes the accident scenarios which have similar source term governing characteristics. The source term is the result of the MAAP analysis and presents the release fraction of the initial core inventory which is released to the environment as a function of time. To characterize the source term associated with each release category, a single representative sequence was chosen for each release category by using MAAP code. Therefore, the source term for each STC represent various sequences assigned to each STC resulting from the internal events as well as internal fire & flooding events.

The staff finds the applicant's clarification that the results listed in DCD Table 19.1-29 are applicable to internal events, internal fire, and internal flooding acceptable. The applicant proposed to add Tables 19.1-30a and -30b providing the source term category frequencies and contributions to LRF for internal fire events and for internal flooding events, respectively. The staff reviewed the applicant's response to find it acceptable because it addresses the staff's concerns about source term evaluation and source term category frequencies and contributions to LRF for internal flooding events. The staff confirmed that the applicant incorporated these changes in DCD Tier 2 Revision 1. Based on applicant's response, RAI 432-8377, Question 19-56, is resolved and closed.

Internal Fire Risk Results and Insights

The results of the internal fire risk evaluation show that the most important contributors to internal fire risk involve a MCR evacuation, a partial loss of CCW, a loss of a dc bus, or a loss of offsite power. They contribute over 90 percent to both internal fire CDF and LRF. The contribution of the MCR evacuation scenarios is disproportionately high in part due to the use of a simple conditional core damage probability of 0.1 and conditional large release probability of

0.01 based on lack of procedures governing the use of the remote shutdown console. As discussed previously under key assumptions, the applicant performed a sensitivity study decreasing the conditional core damage probability for main control room evacuation by an order of magnitude. The results showed that the effect on risk estimates are not significant but turbine building fires that lead to loss of offsite power becomes the most important fire scenario.

The staff's review determined that the descriptions for the top internal fire sequences are not readily available in the DCD for the staff to check for reasonableness. The staff issued RAI 418-8348, Question 19-43, dated February 23, 2016, to ensure that the top internal fire sequences are clearly documented in the DCD (ML16054A291). In its response, the applicant proposed a revision to DCD with additional description of the important fire scenarios. The applicant clarified that a large, unsuppressed, turbine building fire impacting the offsite power cable within the turbine building is a large contributor to risk. The scenario is assumed to disable the alternate AC power so an additional common cause failure that renders the EDGs unavailable result in an SBO which cannot be mitigated, leading to core damage.

Another important fire scenario is an unsuppressed fire in Compartment F078-A19B in Quadrant of the auxiliary building at the 78 ft elevation. This fire results in a loss of DC in train B as well as damage to B and D trains of the 1E 4kV switchgear. The fire, combined with the failure of the auxiliary feedwater system and the operator failure to provide feed and bleed cooling, is assumed to lead to core damage.

In addition, the staff confirmed, by audit of the underlying PRA documentation, the important internal fire scenarios, the fire-induced initiators they cause, and the fire impacts on equipment and operator actions. The staff finds the applicant's approach acceptable. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 418-8348, Question 19-43, is resolved and closed.

The applicant's FPRA identified the risk-significant operator actions. The operator action to support safe shutdown using the remote shutdown console is highly risk-significant since the main control room abandonment scenarios are the most risk-significant internal fire scenarios. Other risk-significant operator actions include opening of the safety depressurization valve for feed and bleed operation, transferring the AFW source from the AFW storage tank to raw water tank or condensate storage tank, tripping the RCPs following loss of seal cooling, aligning the startup feedwater pump, aligning for hot well makeup, and operating the auxiliary charging pump for RCP seal injection.

The results of the internal fire risk evaluation reflect the APR1400 design features that minimize the impact from any single fire in the auxiliary building. These design features include for example, the quadrant separation of the redundant safety equipment and use of 3-hour fire rated barriers. Also, the use of fiber-optic cables between the main control room safety console and the equipment minimizes the impact of hot short-induced spurious operation.

For the purposes of the DC, the staff concludes that the applicant's FPRA is sufficiently consistent with the guidance in NUREG-6850 and SRP Section 19.0. Therefore, the staff concludes that the APR1400 internal fire risk compares favorably with the Commission's CDF and LRF goals as described in SRP Section 19.0.

19.1.5.3 **Internal Flooding Risk Evaluation**

Summary of Application

Section 19.1.5.3 of the DCD describes the internal flooding risk evaluation and its results. The application includes a description of the internal flooding PRA and its results including any risk insights. The applicant performed the internal flooding PRA using the following general steps:

a. Identification of flood sources and target equipment
b. Definition of flood areas
c. Accident sequence definition
d. Initiating event analysis
e. Internal flooding human action development
f. Quantification of flooding sequences.

The applicant performed the internal flooding HRA using an approach similar to that used for the internal events HRA (e.g. CBDTM, HCR/ORE, THERP). The applicant calculated the pipe break frequency values using the methodology described in EPRI 1021086, Revision 2, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments."

The applicant reported the following internal flooding CDF results:

5 percent value:	5.8E-8 per year
Mean value:	4.4E-7 per year
95 percent value:	1.4E-6 per year

The applicant reported the following internal flooding LRF results:

5 percent value:	2.8E-9 per year
Mean value:	2.9E-8 per year
95 percent value:	1.0E-7 per year

Technical Evaluation

In accordance with SRP Section 19.0, the staff conducted its review to determine whether the technical adequacy of the internal flooding PRA is sufficient to justify the risk estimation and identification of risk insights that are used to support the DC application. To evaluate the technical adequacy of the applicant's internal flooding PRA, staff reviewed the PRA for consistency with RG 1.200 and SRP Section 19.0. The staff noted in the audit documentation that the applicant subjected the internal flooding PRA to a peer review against the ASME/ANS standard requirements. The staff considered the results of this peer review, which found that the internal flooding PRA generally met the ASME/ANS requirements for at least Capability Category I. The staff's review focused on ensuring that the internal flooding risk compares favorably against the Commission's goals, as described in SRP Section 19.0, and that sufficient risk insights are identified to support the DC application.

Flooding Protection Design Features

The APR1400 design includes a number of design features that provide flood protection to the important SSCs.

- Physical separation of redundant trains of equipment provided by the structural wall barriers
- The lowest elevation contains no doors or passages between the two divisions and the limited penetrations through the divisional wall are sealed
- Each of the two divisions is further compartmentalized into two separate compartments (i.e., quadrants)
- Flood barriers with flood doors provide separation between the quadrants, providing a design that is capable of confining water to one quadrant up to the 64-ft (19.5-m) elevation
- Alarm in the control room provides indication of open flood doors installed between the quadrants
- Floor drainage systems are separated by quadrant and are designed to prevent back flow of water to areas containing safety-related equipment
- Each quadrant contains its own sump equipped with redundant sump pumps
- Emergency overflow lines direct fluid from upper elevations of the auxiliary building to the lowest elevations within the quadrant
- Each quadrant is designed to contain over 600,000 gallons (158.5 cubic meters) of water without impacting equipment in adjoining quadrants
- Component cooling water heat exchangers and service water pumps are located outside the auxiliary building
- Doors between the auxiliary building and the turbine building are located above the maximum turbine building flood elevation
- Circulating water system piping is confined to the turbine building to prevent this flood source from flowing into the auxiliary building.

The staff finds that the results of the internal flooding risk assessment reflect these flood protection design features which minimize the flood hazard propagating from one division to the other division. These design features in conjunction with any related assumptions must be validated in order for the internal flooding risk profile to remain valid. The applicant identified the following Item COL 19.1(4) for the COL holder to review as-designed and as built information and conduct walkdowns as necessary to confirm that the design features and assumptions used in the PRA remain valid:

> The COL holder is to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA (including PRA inputs to RAP and SAMDA) remain valid with respect to internal events, internal flood and fire events (fire barrier and fire barrier penetrations, routings and locations of pipe, cable, and conduit), and HRA analyses (development of operating procedures, emergency operating procedures, and severe accident management guidelines and training), external events including PRA-based seismic margins and HCLPF fragilities, and LPSD procedures.

Key Assumptions

In accordance with SRP Section 19.0, the staff reviewed the applicant's evaluation of uncertainty in the key assumptions used in the PRA. Specifically, the staff evaluated how the uncertainties in key assumptions impact risk quantification and risk insights. The staff found that the DCD did not provide sufficient evaluation of uncertainties associated with internal flooding PRA assumptions. On February 23, 2016, the staff issued RAI 418-8348, Question 19-45, requesting the applicant to ensure that key assumptions for internal flooding PRA are identified and their uncertainties are properly assessed in the DCD (ML16054A291). RAI 418-8348, Question 19-45, was subsequently subsumed into RAI 434-8352, Question 19-92. In its response, the applicant stated that the detailed sensitivity and uncertainty analysis will be performed as a part of the PRA maintenance and upgrade process as described in DCD Section 19.1.2.4. In addition, the staff confirmed, by audit of the underlying PRA documentation, that the applicant identified sources of modeling uncertainty such as equipment failure rates and human error probabilities that should be evaluated as additional information on design and procedures as well as when operating experience becomes available. The DCD also includes provisions for the COL applicant to describe their PRA maintenance and upgrade program [COL 19.1(6)], and for the COL holder to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA remain valid [COL 19.1(4)]. Based on these considerations, the staff finds the applicant's approach adequate and RAI 418-8348, Question 19-45, is resolved and closed.

The staff reviewed the applicant's HRA performed to support the internal flooding PRA, which consisted of two separate analyses. The first analysis determined the operator action basic events for isolation of flood sources and relied upon the same methodologies (e.g., CBDTM, HCR/ORE, THERP) used for the full power internal events analysis. The second analysis determined the appropriate HEPs for operator actions not related to flood mitigation (e.g., opening of POSRVs for feed and bleed operation) and used a simplified approach. For operator actions to be performed in the MCR, the applicant assumed that the baseline HEPs assumed in the internal events PRA are applicable. For operator actions to be performed outside the control room, the applicant assumed the operator action to be a guaranteed failure (i.e., HEP of 1.0).

As discussed previously in Section 19.1.5.2, "Internal Fire Risk Evaluation," of this SER, the staff noted an assumption in the PRA audit documentation that stated that the operator action to perform feed and bleed is entirely a MCR action, which is inconsistent with HRA used for internal events. During a public meeting on September 27, 2016 (ML16274A253), the applicant committed to revise its response to RAI 418-8348, Question 19-47 to address this inconsistency (ML16103A569). In its response dated August 25, 2017 (ML17237C042), the applicant evaluated the potential that operators may need to traverse flooded areas outside the main control room to align for feed and bleed operation. The evaluation considered the potential for specific flooding events to affect operator actions to initiate feed and bleed cooling and concluded that there would be a negligible impact on risk. The applicant also proposed a revision to DCD Table 19.1-4 clarifying the need for the operators to access the rooms where the local manual actions take place and that the HRA accounts for potential internal fire or flood impacts on the operator's ability to perform feed and bleed. The staff finds the applicant's approach acceptable. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 418-8348, Question 19-47, is resolved and closed.

Based on the above, the staff finds that the applicant provided sufficient information to ensure that key assumptions used in the DC PRA will be appropriately evaluated and dispositioned during the COL phases to ensure that the PRA results and insights continue to remain valid.

Internal Flood Source Identification and Screening

The APR1400 has a highly compartmentalized design. For the internal flooding analysis, the applicant initially identified and characterized several hundred potential flood areas spanning the auxiliary building, the turbine building, the emergency diesel generator building, the essential service water building, the component cooling water building, and the compound building. The applicant identified the flood areas accounting for the design-specific physical layouts and separations at the level of individual rooms or combined rooms/halls for which plant design features exist to restrict flooding. The applicant characterized the flood areas, summarizing the interfaces with adjoining areas, including any penetrations and barriers, availability of mitigation equipment, presence of PRA-modeled equipment, and flooding sources. For each of the flood areas, the applicant assessed the potential propagation paths including the barriers' ability to withstand the flooding events. The staff finds that the applicant appropriately identified and characterized the flood areas consistent with the ASME/ANS PRA Standard as clarified by RG 1.200 and the applicable guidance in SRP Section 19.0.

Based on the initial list of flood areas, the applicant screened out certain flood areas as described in Section 19.1.5.3.1.3 of the DCD. For example, the applicant screened out a flood area from the analysis if it contained no flood source and no flood source could propagate into the flood area either through normal pathways or through failure of barriers from other flood areas. A flood area was also screened out if it would not cause an initiating event or an immediate plant shutdown and if it contained no mitigating equipment modeled in the PRA. A flood area may be screened if there is sufficient justification for operator actions to mitigate the flooding event (i.e., flood indication is available in the control room, flood source can be isolated; and actions are procedurally directed, that adequate time is available). The staff reviewed these and other screening criteria used and finds that the flood area screening is consistent with the ASME/ANS PRA Standard as clarified by RG 1.200 and the applicable guidance in SRP Section 19.0.

For each flood area, the applicant identified the potential sources of flooding (systems), associated flooding mechanism, and the characteristics of the release (e.g., type of breach, flow rate, capacity of source, pressure, and temperature of the source). The staff audited the applicant's PRA documentation and confirmed that the applicant performed a systematic evaluation to screen out the flooding areas and flooding sources that do not need to be included in the PRA. Table 19.1-87, "Flood Sources by Flood Area," list the flood areas and flood sources that have been screened in for the analysis. The staff finds that the applicant appropriately identified and characterized the potential flood sources and associated internal flood mechanisms consistent with the ASME/ANS PRA Standard as clarified by RG 1.200 and the applicable guidance in SRP Section 19.0.

Internal Flood Scenario Development and Initiating Event Analysis

Based on the detailed characterization of the flood areas and the flood sources, the applicant developed the flooding scenarios for each flood area, including damage within the area, flood egress from the area, damage to connecting areas and associated flood heights, detection of the flood, potential means of isolation, and potential for un-isolated floods to fill multiple flood areas. The staff finds that the applicant has systematically developed the flood scenarios for each screened in flood area, consistent with the ASME/ANS PRA Standard as clarified by RG 1.200 and the applicable guidance in SRP Section 19.0.

The staff reviewed the applicant's identification and quantification of applicable flood-induced plant initiating event frequencies for each flood scenario that could lead to core damage. The applicant calculated the system pipe break frequencies using the methodology described in EPRI 1021086, "Pipe Rupture Frequencies for Internal Flooding." The staff verified through the audit that the applicant used the isometric and piping arrangement drawings to develop a detailed mapping of pipe segments to flood rooms, for each system included as a potential flood source. The staff also verified during the audit that the applicant properly determined the equivalent break sizes based on the release flow rates that would cause an initiating event (e.g., shutdown per TSs). The equivalent break sizes were used to estimate the pipe break frequency.

The staff noted that the applicant considered flooding initiating events caused by inadvertent operation or erroneous operation of a plant component during maintenance. The applicant concluded that these scenarios do not contribute significantly to the pipe break flooding frequencies during at-power conditions. On February 23, 2016, the staff issued RAI 418-8348, Question 19-49 requesting the applicant to address the potential for flooding events caused by operator error during maintenance activities (ML16054A291). In response dated December 28, 2017, the applicant established COL 19.1(23) for the COL applicant and/or holder to demonstrate that maintenance-induced floods are negligible contributors to flood risk when plant-specific information is available (ML16323A474). The staff finds the proposed approach acceptable because it addresses staff's concern about the potential for flooding events caused by operator error during maintenance activities. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the RAI response. Accordingly, the staff considers RAI 418-8348, Question 19-49, resolved and closed.

Based on above, the staff finds that the applicant appropriately identified and quantified the flood-induced initiating event frequencies consistent with the ASME/ANS PRA Standard as clarified by RG 1.200 and the applicable guidance in SRP Section 19.0.

Level 2 At-power Internal Flooding PRA

The DCD Section 19.1.5.3.1.7 states that for each postulated internal flooding scenario, the flooding initiating events and the flood-induced equipment failures are included in the logic models and that each internal flooding scenario is quantified using the same process used in the internal events PRA quantification to generate the core damage frequency. However, the DCD does not describe the process of PRA quantification for Level 2 event trees. Therefore, in the public meeting on May 28, 2015, in Audit Question SA-12, the staff asked the applicant to provide this information. In the response to Audit Question SA-12, dated October 1, 2015, the applicant stated that the quantification process of Level 2 internal flooding is similar to the Level 2 internal events process and committed to update DCD Section 19.1.5.3.1.7 with this clarification (ML15274A284). As discussed in Section 19.1.5.2 of this SER, in response to RAI 432-8377, Question 19-56, the applicant committed to add Table 19.1-30b, "Source Term Category Frequencies and Contributions to LRF for Internal Flooding Events," to the DCD providing the source term category frequencies and contributions to LRF for internal flooding events. The staff finds acceptable the applicant's use of the Level 2 internal events quantification process for Level 2 internal flooding.

Internal Flooding Risk Results and Insights

The applicant estimated the internal flooding mean CDF and LRF of 4.4E-7/yr and 2.9E-8/yr, respectively, using reasonable assumptions and crediting the key design features of the APR1400 design. Based on the staff evaluation of the applicant's PRA against the ASME/ANS Standard as clarified by RG 1.200 and the applicable guidance in SRP Section 19.0, the staff finds the technical adequacy of the PRA model to be sufficient for DC purposes. Therefore, the staff concludes that the APR1400 internal flooding risk is consistent with the Commission's CDF and LRF goals.

The results show that the estimated CDF and LRF from internal flooding are about an order of magnitude lower than that from internal fires. The results of the internal flooding risk evaluation show that the most important internal flooding accident sequences involve a major or a moderate fire protection system break in the auxiliary building, comprising over 90 percent of the internal flooding CDF. The largest contributor to risk is a large fire protection line break in Corridor 078-A19B of Quadrant B of the auxiliary building. The flooding causes a failure of the train B electrical equipment. Accumulation of water causes a failure of the door between Quadrants B and D and the subsequent surge of water causes loss of the train D electrical equipment. Failure of secondary cooling and failure of equipment needed to support feed and bleed cooling would result in core damage.

The DCD states that most of the other top scenarios follow the same progression. For example, a flooding event occurs and cannot be isolated before barriers to adjoining areas are challenged. Propagation causes flood-induced failure of two trains of electrical power. The resulting hardware failures result in a general transient or require an immediate reactor shutdown per technical specifications. Random hardware failures then preclude operation of secondary cooling and feed and bleed cooling for decay heat removal.

The applicant's internal flooding PRA identified the risk-significant operator actions. The operator action to open the POSRVs to support the feed and bleed operation is risk-significant, since the loss of secondary side cooling scenarios are the most risk-significant internal flooding scenarios. Other risk-significant operator actions include actions to isolate floods due to fire protection line and auxiliary feedwater piping breaks and operation of the essential chilled water (ECW) pumps.

The results of the internal flooding risk evaluation reflect the APR1400 design features that minimize the flood hazard propagating from one division to the other division. These design features include, for example, the quadrant separation of the redundant safety equipment along with elimination of doors and passageways connecting the divisions of safety-related equipment up to the 64-foot (19.5 m) elevation (in the auxiliary building). The emergency overflow lines and floor drains are designed to provide a flow path from upper elevations to the basement. The large volume in the basement of each quadrant and drain sump alarms allow time for the operator to detect and isolate any internal flood before equipment could fail.

Based on the foregoing discussions, the staff finds that the APR1400 internal flooding risk compares favorably with the Commission's goals, as described in SRP Section 19.0, and that sufficient risk insights are identified to support the DC application.

19.1.5.4 **Other External Events Risk Evaluation**

Summary of Application

The applicant identified site-specific attributes for other external events, described in DCD Chapter 2. External events identified in the PRA Standard were screened from further evaluation if they were too infrequent to affect total risk or if their consequences were negligible. Those that were not screened from further review or subsumed within other hazard categories must be addressed by the COL applicant in a site-specific PRA.

Technical Evaluation

The staff's evaluation presented here in this section covers all other external hazards that could affect the safety of APR1400 design and operation. These external hazards are described in DCD Section 19.1.5.4 "Other External Events Risk Evaluation" and can also be found in the ASME/ANS PRA Standard, as endorsed by RG 1.200. The applicant adapted from ASME/ANS PRA Standard, Appendix 6-A, a list of external hazards requiring consideration. DCD Table 19.1-105, "Summary of External Hazard Dispositions," itemizes the external hazards and the associated parameters that a COL applicant referencing the APR1400 design can use to demonstrate that the design is suitable for the proposed site.

In performing its review of the risk initiated by these external hazards, the staff followed the guidance specified in the SRM on SECY-93-087, which provides direction about the treatment of external hazards to support DC applications. The Commission approved the staff's position that advanced LWR vendors should perform bounding analyses of site-specific external hazards likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). Subsequently, when a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped. If the site is enveloped, the COL applicant need not perform further PRA evaluations for these external hazards.

To ensure that the applicant's assessment is in conformance with the above guidance and comprehensive in its scope by addressing all applicable external hazards, the staff reviewed the information provided in DCD Section 19.1.5.4 using the guidance provided in SRP Section 19.0 and also in ASME/ANS PRA Standard, Part 6, "Requirements for Screening and Conservative Analysis of Other External Hazards At-Power," Part 7, "Requirements For High Wind Events

At-Power PRA," Part 8, "Requirements For External Flood Events At-Power PRA," and Part 9, "Requirements For Other External Hazards At-Power PRA."

As specified in the ASME/ANS PRA Standard and staff guidance, all potential natural and man-made external hazards that may affect the nuclear power plant should be considered, and each of them should be either screened out on a defined basis or subjected to additional analysis using a PRA. If an external hazard cannot be screened out, then a demonstrably conservative or bounding analysis, when analyzed together with quantitative screening criteria, can be used to provide a defensible basis for screening out the event.

The staff's review of DCD Section 19.1.5.4 finds that the screening process conducted by the applicant conforms to the staff guidance described above. However, the applicant did not perform any quantitative or bounding analyses specified in the SRM on SECY-93-087. The external hazards, such as transportation accident, dam failure, river flooding, storm surge, high winds, tornadoes, hurricane, tsunami, etc., have not been specifically evaluated. Therefore, on October 22, 2015, in RAI 270-7894, Question 19-14, the staff requested the applicant to provide the bounding results of the probabilistic evaluation of the above hazards, and include the discussion in the DCD (ML15295A265).

On August 7, 2017, the applicant responded to the staff's requests in Question 19-14 (ML17219A182). The applicant stated that the APR1400 external events assessment was performed based on the following two key assumptions:

1. All SSCs that are modeled in the PRA are designed to withstand the design-basis tornado (DBT) and design-basis hurricane (DBH) including all effects, i.e., pressure loading, pressure drop and missile impacts.
2. The non-safety related systems, structures, and components are designed such that they will not collapse on or impact the seismic Category I structures containing SSCs in Item (1) above and will not generate missiles more damaging than the DBT and DBH missiles.

Accordingly, the applicant performed bounding analysis for the following hazards: (1) turbine missile, (2) extreme winds and tornadoes, and (3) external flooding. For the turbine missile hazard, the applicant described that the turbine generator layout for the APR1400 is considered to be a favorable orientation, excluding those important SSCs from low-trajectory turbine missile strikes. APR1400 DCD Section 3.5.1.3 evaluates the potential consequences of turbine missiles that include both direct effects and indirect effects. The applicant concluded that the favorable turbine generator placement and orientation, combined with the design and fabrication processes, the redundant and fail-safe turbine control system, the maintenance and inspection programs, and the overspeed protection systems, all together provide an acceptably small probability of turbine missiles causing damage to essential SSCs. The applicant followed RG 1.115 guidance and estimated the likelihood of an unacceptable damage resulting from turbine failure to be less than 1E-7 per year. The staff reviewed DCD Section 3.5 and finds the probability calculation reasonable, consistent with the guidance provided in RG 1.115, and therefore acceptable. The staff concludes that because of the low damage probability, the turbine missile is an insignificant hazard and can be screened from the detailed design-specific PRA.

With regard to the extreme wind and tornado hazards, DCD Section 3.3 describes the design basis for tornadoes and hurricanes, which was selected as corresponding to Region I in RG 1.76. The maximum windspeed of this DBT is 230 mph as corresponding to a probability of exceedance of 1E-7 per year. The DBH was selected as having a maximum windspeed of 260 mph as corresponding to a probability of exceedance of 1E-7 per year, as specified in RG 1.221. The plant structures (i.e., containment building, containment building internal structures, auxiliary building and emergency generator building, and diesel oil fuel tank) are designed to the maximum of the load effects from the DBT and DBH. The design against DBT also includes the effect of pressure drop as the tornado moves over the building. In addition, DCD Section 3.5 describes how the APR1400 structures are designed against the design basis missile spectrum for the DBT and DBH as specified in RG 1.76 and RG 1.221, respectively. Specifically, the exterior walls and roof slabs of seismic Category I structures are designed to withstand the local and global effects of these missiles.

The staff reviewed the design information and agrees with the applicant's conclusion that because of the robustness of the APR1400 design, the contributions to CDF and LRF from extreme winds (including tornadoes, hurricanes, and thunderstorms) are very low and can be screened from the detailed design-specific PRA. The staff also agrees with the applicant that after a site is chosen, per Item COL 19.1(10), the COL applicant will be performing a comparison of the site characteristics to those assumed in the bounding analyses to ensure that the site is enveloped. The COL applicant will also perform site-specific PRA to address any site-specific hazards that are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.

With regard to the external flooding, in DCD Section 19.1.5.4.4.2, the applicant investigated the risk significance of this hazard on the design. The potential flooding analyzed in that section includes both natural phenomena (i.e., high river or lake water, river flooding, ocean flooding such as from high tides or wind driven storm surges, extreme precipitation, tsunamis, seiches, flooding from landslides) and man-made events (i.e., failures of dams, levees, and dikes).

DCD Table 2.0-1 specifies the APR1400 maximum flood elevation as 1-foot below the plant grade in the vicinity of the SSCs important to safety. Further, DCD Section 2.4 describes the flood analysis for different sources for the COL applicant to perform for the chosen site.

Since the site-specific characteristics are not yet available at the DC stage, the staff agrees with the applicant's position described in COL 19.1(10) that after a site is chosen, the COL applicant will re-evaluate the external flooding hazard in conformance with RG 1.206 guidance and ASME/ANS PRA Standard, Parts 6 and 8 requirements. For these reasons, the staff finds the applicant's approach acceptable and thus RAI 270-7894, Question 19-14, is resolved and closed.

For the other external hazards, with regard to the APR1400 "screening" process, the applicant referenced the five screening criteria provided in ASME/ANS PRA Standard, SR EXT-B1, as the screening basis. In DCD Table 19.1-105, for every screened-out hazard, the applicant specifically identified the associated screening criteria and provided justification for the screening. The applicant also provided treatments for all external hazards listed in this table. Generally, the applicant's screening process separated the external hazards into two groups.

The first group includes those hazards that have been screened from the detailed design-specific PRA using SR EXT-B1 screening criteria, such as avalanche, biological events, coastal erosion, drought, fog, forest fire, frost, hail, high temperature, high tide, hurricane, ice, landslide, lightning, low lake/river water level, low temperature, meteorite/satellite strikes, intense precipitation, river diversion, sandstorm, seiche, snow, soil shrink/swell, storm surge tsunami, turbine generated missiles, volcanic activity, and waves. The second group includes the external hazards that have been considered as site-specific events requiring detailed study by the COL applicant as specified in Item COL 19.1(10). This group comprises the following external hazards: aircraft impacts, external flooding, extreme winds and tornadoes, dam failure, industrial or military facility, internal flooding, pipeline accident, release of chemicals from onsite storage, seismic activity, toxic gas, and transportation accidents.

To ensure that the risk associated with the external hazards is to be fully addressed and dispositioned during the COL phase, the applicant established the following Item COL 19.1(10):

> The COL applicant or holder needs to ensure that screened events do not have site specific susceptibility and do not exceed the CDF and LRF design targets specified in Subsection 1.2.1.1.1e. The COL applicant or holder is to address the following issues with a site-specific risk assessment, as applicable.
>
> - Dam failure
> - Tsunami
> - External flooding
> - Extreme winds and tornadoes
> - Industrial or military facility
> - Pipeline accident
> - Release of chemicals from onsite storage
> - River diversion or flooding
> - Toxic gas
> - Transportation accidents
> - Storm surge
> - Lightning
> - Aircraft crash event.
>
> In addition, the COL applicant or holder is to ensure the site-specific susceptibility is not an outlier for the following issues, as applicable:
>
> - Avalanche
> - Biological events
> - Coastal erosion
> - Drought
> - Forest fire
> - High summer temperature
> - Hurricane
> - Landslide
> - Low lake/river water level
> - Low winter temperature

- Sandstorm
- Volcanic activity.

The staff finds the above COL information item comprehensive, consistent with the SRP Section 19.0, RG 1.206, and ASME/ANS PRA Standard, and therefore is acceptable.

The staff finds that the applicant has fully examined the external events hazards described in PRA Standard and rationally characterized them in the DCD. The staff concludes that, in combination with the COL information items described above, the applicant properly addressed the potential impacts of other external hazards on APR1400 design, in conformance with SRP Section 19.0, RG 1.200, and RG 1.206, therefore, it is acceptable.

19.1.6 Safety Insights from the PRA for Other Modes of Operation

19.1.6.1 Level 1 Internal Events PRA for Low-Power and Shutdown Operations

Summary of Application

The application describes the methodology for quantitative evaluation of internal events for operations at LPSD. The applicant identifies nine technical tasks:

a. Plant Operating State (POS) Development
b. Initiating Events Analysis
c. Accident Sequence Analysis
d. Success Criteria Analysis
e. Systems Analysis
f. Data Analysis
g. Human Reliability Analysis
h. Analysis of Large Early Release
i. Quantification.

In DCD Section 19.1.2, the applicant states, "The LPSD methodology and modeling are state of the art and are designed to meet the requirements of the draft ANS/ASME LPSD PRA Standard." The applicant notes that some of the requirements of the draft LPSD PRA Standard could not be met during the design stage.

The applicant estimates the LPSD risk from internal events as 1.8E-6 per calendar year. This estimate includes the frequency of an outage per calendar year and the average duration in hours of each POS during an outage. Thus, the shutdown CDF can be added to the full power CDF for comparison against the Commission's goals. In the APR1400 accident sequences, core damage is defined as Peak Cladding Temperature (PCT) > 1300 °F (707.4 °C).

The top LPSD CDF initiating event contributor is RCS over draining to reach mid-loop conditions, such that suction to the shutdown cooling system (SCS) pumps, which provide the decay heat removal function is lost. This top initiating event comprises around 49 percent of the LPSD internal events CDF. Loss of offsite power and SBO events together contribute 13 percent of the total LPSD internal events CDF. Failure to control RCS level during reduced inventory conditions such that suction for SCS is lost contributes another 9 percent of the LPSD internal CDF. Unrecoverable LOCA through the chemical volume and control system (CVCS) drain line contributes about 9 percent. Total loss of essential service water (initiating event "TS") plus total loss of component cooling water (initiating event "TC") events, also contributes about 9 percent to the LPSD CDF.

The top four cutsets contribute more than 77 percent of the total CDF. These cutsets initiate by RCS over draining to reach mid-loop conditions or failure to maintain water level at reduced inventory conditions. Common cause failure (CCF) events, which have the highest RAW values, include: CCF of the SI DVI check valves, CCF of IRWST sumps due to plugging, CCF of PPS LC application software, CCF of PPS operating system software, CCF of the ECW components, and CCF of the ESW components. Based on operator action RAW values, considering all POSs, the risk-significant operator actions include the operator failing feed and bleed, and the operator failing to isolate and makeup from overdraining the RCS to reach midloop conditions in POSs 5 and 11, the midloop POSs. The next most risk significant operator actions include the operator failing to restore the decay heat removal function of the SCS during POSs 5 and 11, the midloop POSs.

RCS makeup following a loss of inventory or a loss of suction to the SCS pumps is not automated in this design. However, two trains of SI are required to be operable according to technical specification (TS) limiting condition for operation (LCO) 3.5.3. Two trains of safety injection systems (SIS) shall be OPERABLE and diagonally oriented with respect to the reactor vessel during Modes 4, 5, and Mode 6 with RCS level < 39.7 m (130 ft. 0 in). The APR1400 design has added ultrasonic instrumentation and control room alarms that monitor hot leg level conditions.

The uncertainty results for the Level 1 internal events CDF for LPSD operations are summarized below:

Five percent value:	7.4E-7/calendar year
Mean value:	1.9E-6/calendar year
Ninety-Five percent value:	4.4E-6/calendar year

Uncertainty in the Level 1 shutdown PRA results is quantified with a process similar to that described for full power internal events in DCD Section 19.1.4.1.2.6. Parametric uncertainty was represented by selecting an uncertainty distribution for each parameter type, as described in DCD Section 19.1.4.1.2.6. Modeling uncertainty is not represented in the shutdown model.

Successful SCS operation is the preferred end-state for all shutdown sequences. The applicant states that the COL holder should limit planned maintenance that can potentially impair one or both shutdown cooling (SC) trains during the shutdown modes. The applicant also states that plant shutdown risk can be minimized by appropriate: outage management, administrative controls, procedures, and operator knowledge of the plant configuration. If one train of the SCS is unavailable for any reason during shutdown, the shutdown cooling function is dependent upon the remaining train and its support systems.

The applicant improved decay heat removal requirements in the TS during Mode 5 operation with the RCS loops not filled. Two shutdown cooling trains shall be operable and one shutdown cooling train shall be in operation. In addition, one containment spray pump in the operating shutdown cooling train shall be operable. This containment spray pump can be manually realigned to meet the requirement of a shutdown cooling pump. The applicant improved these TS according to 10 CFR 50.36(c)(2)(ii)(D), LCO selection Criterion 4. The applicant also established Item COL 19.1(11), which states:

> The COL applicant and/or holder is to develop outage management procedures that limit planned maintenance that can potentially impair one or both SC trains during the shutdown modes.

In addition, the applicant established Item COL 19.1(12), which states:

> The COL applicant and/or holder is to develop procedures and a configuration management strategy to address the period of time when one SC train is unexpectedly unavailable (including the termination of any testing or maintenance that can affect the remaining train and restoration of all equipment to its nominal availability).

Technical Evaluation

Plant Operating State Development

Consistent with the trial LPSD PRA Standard and SRP Section 19.0, the staff reviewed the applicant's POS analysis to ensure that it defined a complete set of plant operating states. The staff reviewed the screening and grouping of POSs. The staff also reviewed the applicant's POS frequencies and POS durations associated with the decay heat level and RCS configuration defined for each POS.

The staff required more information regarding the POS definitions used to develop the average shutdown model. Given that shutdown risk may be highly outage-specific, consistent with SRP Section 19.0, the staff reviewed the shutdown PRA insights to confirm that operational assumptions used to develop an average shutdown have been clearly documented in the DCD model (e.g., use of nozzle dams, outage schedule, containment status, procedural requirements). On June 22, 2016, the staff issued RAI 42-7945, Question 19-4 (ML16203A282, proprietary version is ML15174A322), which requested the applicant to provide the following information for each POS in DCD Section 19.1.6 and to update the DCD as required.

1. RCS level
2. Anticipated decay heat level as a function of time after shutdown
3. Size and location of RCS vents
4. Time to RCS boiling given a loss of the decay heat removal function
5. Time to core uncovery
6. Thermal-hydraulic code used to obtain time to core uncovery.

The applicant provided their final response to RAI 42-7945, Question 19-4 on September 30, 2015 (ML15273A319). The applicant documented the requested information in proposed DCD Table 19.1-93, "Summary of Analysis Results for Plant Operating States." Table 19.1-93 was re-numbered as Table 19.1-92b, "Summary of Analysis Results for Plant Operating States," in DCD Tier 2 Revision 1, dated March 2017. The staff confirmed that Table 19.1-92b, "Summary of Analysis Results for Plant Operating States" was added to Revision 1 of the DCD as committed in the response to RAI 42-7945, Question 19-4. Therefore, RAI 42-7945, Question 19-4, is resolved and closed.

Based on the staff's review of: (1) proposed DCD Table 19.1-93, (2) the staff's confirmatory mid-loop MELCOR calculation, and (3) the mid-loop loss of core cooling calculation referenced in the "APR1400 Fukushima Technical Report," Section A.5.3 "Shutdown Condition with SGs not Available," the staff requested additional information be added to the proposed DCD Table 19.1-93. Therefore, on March 16, 2016, the staff issued RAI 446-8535, Question 19-98 (ML16076A030), requesting that the applicant provide the following information.

1. RCP seal leakage rate for each POS
2. Leakage rate from temporary seals used for the incore instrumentation for each applicable POS
3. The definition of hot leg top level (for POS 4 only).
4. The definition of mid-loop operation level (for POS 5 only)
5. Clarification in POS 4A, with the RCS closed except for the open Reactor Coolant System Gas Vent System (RCGVS), whether the RCS is being drained with a cover gas
6. Clarification in each POS where reflux cooling is being credited, the assumed initial SG secondary side level and the number of SGs with secondary inventory
7. Clarification in what POS the vessel head is removed and re-installed.

In the final response to RAI 446-8535, Question 19-98, dated June 30, 2016 (ML16182A575), the applicant documented the additional requested information in proposed DCD Table 19.1-93, "Summary of Analysis Results for Plant Operating States." The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the response to RAI 446-8535, Question 19-98 to include the additions requested above in Table 19.1-92b, "Summary of Analysis Results for Plant Operating States." Therefore, RAI 446-8535, Question 19-98, is resolved and closed**.**

From evaluating the applicant's response to RAI 446-8535, Question 19-98, in POS 04A, the staff understands that the RCS is drained from normal operating levels in the pressurizer with the reactor coolant system gas vent system open, and the low-temperature overpressure protection (LTOP) valves are in automatic protection mode. Instrument air is ported to the RCS through the pressurizer vent path before the pressurizer manway is opened. This alignment prevents the RCS from drawing a vacuum and speeds draining. RCS inventory is drained through the CVCS letdown line.

On October 22, 2015, the staff issued RAI 268-8308, Question 19-13 (ML15295A262). The staff requested a clarification in POS 4B as to whether the pressurizer manway is opened once mid-loop conditions are reached. Mid-loop conditions are defined in Generic Letter (GL) 88-17 as when the RCS water level is below the top of the flow area of the hot legs at the junction with the reactor vessel. The staff also requested the applicant to document in the DCD whether vacuum refill of the RCS is performed from mid-loop conditions.

In the final response to RAI 268-8308, Question 19-13, dated June 10, 2016 (ML16162A789), the applicant updated DCD Table 19.1-81, "LPSD Plant Operating States" to clarify that: (1) instrument air is ported to the RCS through the pressurizer vent path during RCS drain-down in POS 4A, (2) the pressurizer manway is opened before reduced inventory conditions are reached, and (3) vacuum refill of the RCS is administratively prohibited during mid-loop operation. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the response to RAI 268-8308, Question 19-13 to include the additions requested above in Table 19.1-81, “Plant Operating States.” Therefore, RAI 268-8308, Question 19-13, is resolved and closed.

The staff found that the LPSD risk from POS 7 and POS 9 was screened quantitatively from the average shutdown model. In these POSs, the cavity is filled to the level necessary for core alterations. The staff also recognized that installed reactor internals could shorten the time to core boiling given possible limited communication between the RCS inventory around the core and inventory in the refueling cavity. In addition, the staff was concerned about the losses of RCS inventory events caused by operators (valve misalignments). To evaluate the screening of POS 7 and POS 9 from the LPSD PRA quantitatively, on September 30, 2015, the staff issued RAI 232-7864, Question 19-8 (ML16203A437), requesting that the applicant provide the following information:

a. An evaluation documenting the time to core damage given an extended loss of the decay heat removal function with and without installed reactor internals
b. An evaluation that considers all possible drain paths from the refueling cavity including drain rates
c. The availability of instrumentation and alarms to detect and mitigate each potential drain path
d. The likelihood of the operator failing to terminate each potential leak path
e. The availability of pumps and a source of water to restore RCS inventory for each leak path.

The staff also needed clarification in the DCD as to whether temporary fuel racks could be installed in the refueling cavity.

In the final response to RAI 232-7864, Question 19-8 (ML16309A172), dated November 4, 2016, the applicant provided an update to DCD Chapter 19 documenting why POSs 7 and 9 were screened from quantification. In this update, the applicant provided results of RELAP calculations documenting the time to core uncovery with and without reactor vessel internals installed. For both cases, the times to core uncovery exceeds 60 hours. In this update, the applicant also discussed potential drain paths from the cavity. The most rapid refueling cavity drain down scenario is through the SCS back to the IRWST. Draining of the cavity back to the IRWST is normally performed after refueling. There is no automatic function that performs this valve re-alignment. Several hundred thousand gallons must be removed from the cavity before SCS suction is lost. Finally, the IRWST is the preferred source of water for providing makeup to the RCS during Modes 5 and 6.

The applicant also provided updates to DCD Chapter 9. The applicant updated DCD Section 9.1.4.2 to state that there are no temporary fuel racks in the refueling cavity. The applicant also updated DCD Section 9.1.3.5.4 to include a level transmitter installed in the refueling cavity. This transmitter measures level from the bottom to 12 inches (in) (30 centimeters [cm]) below the top of the cavity. Refueling cavity water level is indicated in the

MCR and the remote shutdown room (RSR). The refueling cavity water level transmitter annunciates high water level and low water level in the MCR and RSR.

The applicant's proposed update to the DCD resolves the staff concerns outlined in RAI 232-7864, Question 19-8, regarding the qualitative screening of POSs 7 and 9 from the quantitative evaluation. The staff confirmed that DCD Tier 2, Sections 9.1.4.2, 9.1.3.5.4, and 19.1.6.1 Revision 1, dated March 2017, were revised as committed in the response to RAI 232-7864, Question 19-8. Therefore, RAI 232-7864, Question 19-8, is resolved and closed.

In DCD Section 19.1.6, POS 12B, refill of the RCS after refueling with the pressurizer manway closed, was screened quantitatively from the average shutdown model based on a thermal-hydraulic analysis. The analysis assumes the time to core damage is greater than 24 hours after a loss of shutdown cooling. However, losses of inventory occurring in POS 12B may result in core damage occurring before 24 hours. For the LOCA cases, the applicant performed analyses that conclude that core uncovery does not occur until 23.7 hours, and core damage does not occur within 25 hours. On September 30, 2015, the staff issued RAI 232-7864, Question 19-9 (ML16203A437), requesting the applicant to (1) provide the results of 12B thermal-hydraulic analyses for LOCA and non-LOCA cases and update the DCD accordingly, and (2) document core cooling mitigation strategies for LOCA and non-LOCA cases to ensure that a safe and stable state can be reached.

In the response to RAI 232-7864, Question 19-9 (ML16088A357), dated March 28, 2016, the applicant added proposed DCD Table 19.1-92a, "The Results of Thermal-Hydraulic Analyses for POS 12B." The applicant also added core cooling mitigation strategies for LOCA and non-LOCA cases of POS 12B in DCD Section 19.1.6. The staff confirmed that Table 19.1-92a was incorporated into DCD Tier 2 Revision 1, dated March 2017, as committed in the response to RAI 232-7864 Question 19-9. The DCD addition of thermal-hydraulic analyses results and core cooling mitigating strategies for POS 12B resolves and closes RAI 232-7864, Question 19-9.

Given the incorporation of the information requested above into the DCD Tier 2 Revision 1, dated March 2017, the staff finds the applicant's POS development to be consistent with SRP Chapter 19 and RG 1.200.

Initiating Event Analysis

The staff reviewed how the applicant identified and quantified IEs in each POS or groups of POSs that could lead to core damage. The staff ensured that initiating events were grouped according to the mitigation requirements. The staff also reviewed how the frequencies of the initiating event groups were quantified.

The staff then reviewed DCD Section 5.4.7 for design and operational details that support the frequency of over draining used in the PRA, which is a dominant risk contributor. In DCD Section 5.4.7, it states that the RCS level is maintained higher than the RCS low water level of 8.3 cm (3.28 in) above the loop center, and a SCS flow rate of 14,385 to 15,710 liter per minute (L/min) (3,800 to 4,150 gallons per minute [gpm]) is maintained. DCD Section 19.2.2.2 states:

The shutdown cooling suction lines do not contain loop seals, thereby minimizing the potential to trap gas. The suction piping layout allows self-venting of accumulated gas (or air).

Consistent with 10 CFR 50, Appendix A "General Design Criteria for Nuclear Power Plants," GDC 34 "Residual Heat Removal" and GL 2008-01, the staff needs to conclude that the potential for gas accumulation in the SCS has been reasonably addressed. In addition, the staff needs to conclude that vortexing does not initiate at the specific hot leg levels and SCS flow rates specified above. There was insufficient information in the application to support both conclusions. In accordance with SRP Chapter 19, the operational assumptions used to develop the LPSD PRA model should be clearly documented in the DCD. Thus, on June 22, 2015, the staff issued RAI 42-7945, Question 19-2 (ML16203A282) requesting the applicant to provide the following information and to update the DCD as required.

a. The hot leg level at which the SG nozzle dams are to be installed
b. The highest hot leg level at which the vortexing is expected to initiate in the hot leg given a SCS flow rate of 4150 gpm based on testing and/or analyses
c. The highest anticipated RCS drain rate during reduced inventory operations.

In the response, dated July 28, 2015, to RAI 42-7945, Question 19-2 (ML15212A687) the applicant proposed an ITAAC confirming the piping of the SCS contains no loop seals and maintains a horizontal or downward slope from the RCS to the SC pump (SCP), with exception of the section of piping adjacent to the pump suction flange. The staff then issued a related question, RAI 492-8614, Question 05.04.07-4 (ML16147A594), requesting the applicant to develop an ITAAC for mid-loop conditions, representative of the standard ITAAC for gas entrainment during mid-loop operations. The ITAAC should confirm the decay heat removal function of the SCS will not be impaired by gas entrainment during mid-loop operation while the system is operating at its maximum allowable flow rate and the reactor coolant hot leg level is at the lowest level allowable for decay heat removal.

In the applicant's response, dated July 8, 2016, to RAI 492-8614 Question 05.04.07-4 (ML16190A320), the applicant proposed adding an ITAAC to DCD Tier 1 Table 2.4.1-4, Reactor Coolant System ITAAC, Design Commitment 14, stating "The decay heat removal function of the SCS will not be impaired by gas entrainment during mid-loop operation while the system is operating at its maximum allowable flow rate and the reactor coolant hot leg level is at the lowest level allowable for decay heat removal." The staff confirmed that the DCD Revision 1, dated March 2017, Tier 1 Table 2.4.1-4, was updated to add Design Commitment 15 that specifies "The decay heat removal function of the SCS will not be impaired by gas entrainment during mid-loop operation while the system is operating at its maximum allowable flow rate and the reactor coolant hot leg level is at the lowest level allowable for decay heat removal." Based on the addition of Design Commitment 15 into DCD, Tier 1 Table 2.4.1-4, RAI 42-7945, Question 19-2, is considered resolved and closed.

The staff notes the applicant did not initially provide a discussion on the risk of boron dilution events. As an example, from NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," a LOOP event is postulated and the charging pumps are returned online, powered by the EDGs. If the plant is in startup mode (i.e., dilution and boration in progress), the charging pumps could continue to operate, causing a "slug" of unborated water to collect in the lower plenum of the reactor vessel. If offsite power is restored and the reactor coolant pumps are restarted, then a water slug of unborated water could be injected into the core. In the APR1400 design, the staff noted that when offsite power is restored, the charging pumps are not automatically loaded on the EDGs. On February 22, 2016, the staff issued RAI 409-8325, Question 19-20 (ML16053A015), requesting that the applicant provide the procedures, guidance, or instrumentation that prevents the operator from restarting the CVCS pumps to reduce the risk of boron dilution events.

In the response to RAI 409-8325, Question 19-20, dated August 9, 2016 (ML16222A942), the applicant provided a DCD update to DCD Section 19.1.3.1 to state that the charging pumps will not automatically restart following a LOOP. The charging pump needs to be manually re-started by the operator after bus voltage has been restored. The applicant also provided a DCD update to add Item COL 19.1(7) to require the COL holder to develop procedures that ensure that deboration, following recovery from a LOOP, is not resumed until after at least one reactor coolant pump has been restarted. The applicant proposed an update to Table 19.1-4, "Risk Insights and Key Assumptions," to state that "Boron dilution events at shutdown were screened from the analysis, due to the diverse means of event identification, the availability of procedural recovery actions and the time available for operator response."

The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, Section 19.1.3.1 and Table 19.1-4 were revised as committed in response to RAI 409-8325, Question 19-20. The staff also confirmed that DCD Revision 1, Tier 2 Item COL 19.1(7), was added to DCD Chapter 19 and DCD Table 1.8-2 to require the COL holder to develop procedures that ensure that deboration, following recovery from a LOOP, is not resumed until after at least one reactor coolant pump has been restarted. These DCD updates document why the risk of a postulated boron dilution event is reduced, thereby resolving staff's concerns regarding this issue. Therefore, RAI 409-8325, Question 19-20, is resolved and closed.

The staff then evaluated how ISLOCAs were screened from the LPSD PRA. The CVCS letdown line is directly connected to the RCS and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the letdown nozzle, through the regenerative and letdown heat exchangers, through the letdown orifices, and out of containment through the containment isolation and letdown control valves to the low-pressure sections of the system. The letdown line has a high-pressure alarm that is located downstream of the letdown control valves. This alarm warns the operator when the pressure is approaching the low-pressure system design pressure. When a warning is issued, the control room operator isolates the letdown line to terminate any further pressure reduction. On February 22, 2016, the staff issued RAI 409-8325, Question 19-23 (ML16053A015), requesting that the applicant provide additional information in DCD Section 19.1.6 explaining how the closure of this valve is modeled during any postulated RCS re-pressurization when letdown is operating.

In the response to RAI 409-8325, Question 19-23, dated August 9, 2016 (ML16222A942), the applicant proposed an update to DCD Section 19.1.6.1.1.3, "Initiating Events," to state that the ISLOCA initiating event was retained in the low power and transition modes, using the same frequency as the at-power PRA, and there is a negligible ISLOCA vulnerability once the reactor is depressurized. In the proposed DCD update, the applicant explained that prior to establishing a primary vent, a letdown line rupture or a diversion LOCA was examined for potential containment bypass vulnerability. The applicant concluded the letdown rupture is not a containment bypass vulnerability because there are restricting orifices downstream of the letdown isolation valves within containment, that limit letdown flow even with full RCS pressure during power operation. A rupture upstream of the orifices will occur within containment; a downstream rupture will result in a negligible break flow. Therefore the letdown isolation does not impact potential containment bypass. The staff evaluated the applicant's response and finds that it adequately explains why this design does not have a ISLOCA vulnerability at low power and shutdown, thereby resolving staff's concern. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the response to RAI 409-8325, Question 19-23. Therefore, RAI 409-8325, Question 19-23, is resolved and closed.

Given the discussion above and the incorporation of the information requested into the DCD Tier 2 Revision 1, dated March 2017, the staff finds the applicant's initiating event analysis to be acceptable.

Accidence Sequence Analysis

The staff reviewed the applicant's modeling of the plant response of systems and operator actions to an initiating event. The staff reviewed the applicant's event trees. The staff focused on significant operator actions, mitigating systems, and RCS phenomena such as a postulated RCS re-pressurization that can alter core damage sequences.

The staff found that DCD Section 19.1.6 only included event trees for POS 5, which represents mid-loop conditions. The plant response to a loss of decay heat removal (DHR) is significantly different if the RCS is intact versus an open RCS. Therefore, on October 22, 2015, the staff issued RAI 268-8308, Question 19-11 (ML1515295A262) requesting that the applicant add the event trees for all POSs for the initiating event, loss of the operating train of the SCS, to the DCD. The staff considers these LPSD event trees part of the PRA results.

In the response to RAI 268-8308, Question 19-11, dated June 10, 2016 (ML16162A789), the applicant proposed to add the LPSD event trees for all POSs and all IEs to DCD Chapter 19. The staff reviewed the added event trees and had no additional questions. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the response to RAI 268-8308, Question 19-11. Therefore, RAI 268-8308, Question 19-11, is resolved and closed.

10 CFR 52.47(a)(22) states that the application must include "[t]he information necessary to demonstrate how operating experience insights have been incorporated into the plant design." The staff reviewed the applicant's implementation of the applicable expeditious actions outlined in GL 88-17 to examine whether the applicant is implementing the SG nozzle dam guidance described in GL 88-17 Enclosure Section 2.7, "Nozzles Dams." This section addresses the proper sequence of nozzle dam installation and removal to prevent a sudden loss of RCS inventory given a loss of DHR. This section of GL 88-17 also discusses the need for nozzle dam design pressures not to be exceeded given a postulated loss of DHR and subsequent RCS re-pressurization. Loss of a nozzle dam could lead to rapid reactor vessel voiding.

The staff needed additional information in the DCD on whether the applicant plans to implement the steam generator nozzle dam guidance described in GL 88-17. Therefore, on September 30, 2015, the staff issued RAI 232-7864, Question 19-6 (ML16088A357), requesting this information. This RAI also requested confirmation that a large hot leg path exists whenever cold leg penetrations exist to prevent a rapid loss of RCS inventory as described in Information Notice (IN) 88-36. These configurations are typically not modeled in advanced reactor LPSD PRAs but result in shorter times to core uncovery than from boil-off.

In the final response to RAI 232-7864, Question 19-6, dated July 21, 2016 (ML16203A442), the applicant proposed several DCD updates. The applicant provided a note in TS 3.4.8 that mid-loop operation shall be started more than 4 days after shutdown with the initial hot leg temperature equal to or less than 135 ˚F (57.2 ˚C). The applicant proposed an ITAAC confirming that the nozzle dam withstands its design pressure of 50 psig. The applicant proposed a new risk insight stating:

> The LPSD PRA assumes that Pressurizer manway is opened before mid-loop conditions are reached. In order to assure that the nozzle dam design pressure limit is not exceeded during reduced inventory operations with boiling conditions in the reactor vessel, the APR1400 design includes a requirement that will be imposed to establish a mid-loop vent pathway via the pressurizer manway before operating in reduced inventory. The nozzle dams are installed in the cold legs first and in the hot legs second. The nozzle dams are removed in the hot legs first and in the cold legs second. Before cold leg nozzle dam installation or removal and cold leg maintenance, the steam generator hot leg manway and its associated hot leg pipe should be kept open. When the manway is opened to the containment atmosphere, the surge line provides sufficient venting capacity to prevent RCS pressurization and subsequent nozzle dam failure.

The disposition for this new risk insight is COL 13.5(7), which states: "The COL applicant is to provide a program for developing a shutdown procedure, including the installation and removal order of the pressurizer manway and the nozzle dam.

These proposed DCD updates are consistent with staff guidance in GL 88-17 and IN 88-36 and are therefore acceptable. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, Chapter 16, TS LCO 3.4.8 "APPLICABILITY" was revised to state, "MID-LOOP operation shall be started at least 4 days after shutdown and ≤ 57.2°C (135°F) of initial hot leg temperature." The staff confirmed that DCD Tier 1, Revision 1, dated March 2017, Table 2.4.1-4, "Reactor Coolant System ITAAC," was updated to include Design Commitment 14, "During mid-loop operation, the nozzle dam withstands its design pressure of 50 psig." The staff also confirmed DCD Tier 2 Revision 1, dated March 2017, Chapter 19, Table 19.1-4, Risk Insight 62, was revised as committed in the final response to RAI 232-7864, Question 19-6. Finally, the staff confirmed that proposed COL 13.5(7) was added to DCD Tier 1 as COL 13.5(8). Therefore, RAI 232-7864, Question 19-6, is resolved and closed.

The APR1400 design has incore instrument nozzles installed from the bottom of the vessel. The staff questioned whether temporary seals are used during refueling and/or maintenance, similar to operating PWRs. Therefore, on February 22, 2016, the staff issued RAI 409-8325, Question 19-22 (ML16103A538), requesting information regarding the incore instrument seals. In RAI 409-8325, Question 19-22, the staff also requested information on the design pressure of temporary seals and on the leakage rates from the seals during a postulated RCS re-pressurization.

In the response to RAI 409-8325, Question 19-22, dated August 8, 2016 (ML16222A942), the applicant proposed an update to the DCD Section 19.1.6.1.1.5 to state that temporary seal is not required during refueling since the BM-incore instrumentation (ICI) system uses a fixed ICI system. Since a temporary seal is not required during refueling, the staff's concerns regarding potential leakage from the incore instrument nozzles are resolved. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the response to RAI 409-8325, Question 19-22. Therefore, RAI 409-8325, Question 19-22, is resolved and closed.

Given the discussion above and the incorporation of the information requested into the DCD Tier 2 Revision 1, dated March 2017, the staff finds the applicant's accident sequence analysis to be acceptable.

Success Criteria Analysis

The staff reviewed the applicant's success criteria for critical safety functions such as core makeup requirements, supporting systems, and operator actions necessary to support accident sequence development.

In the APR1400 LPSD accident sequence analysis notebook that was reviewed during the audit, APR1400-K-P-NR-013702, Revision 0, core damage was defined as PCT > 1300 °F (704.4 ˚C). In RAI 409-8325, Question 19-25, dated February 22, 2016 (ML16053A015), the staff requested that the core damage definition be included in DCD Section 19.1.6. In the response to RAI 409-8325, Question 19-25, dated April 12, 2016 (ML16103A538), the applicant included a proposed update to DCD Section 19.1.6 with the definition of core damage. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, Section 19.1.6.1.1.5 was revised as committed in the response to RAI 409-8325, Question 19-25. Therefore, RAI 409-8325, Question 19-25, is resolved and closed.

The staff reviewed the time to boiling and the time to core uncovery for each POS described in proposed DCD Table 19.1-93, "Summary of Analysis Results for Plant Operating States," submitted in the response to RAI 42-7945, Question 19-4 (ML16203A282). Question 19-4 is described in further detail in paragraph titled, "Plant Operating State Development". The staff checked that the success criteria for each POS were referenced to specific thermo-hydraulic calculations that were performed using RELAP 5, Mod 3.3. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the response to RAI 42-7945, Question 19-4, to include Table 19.1-92b, "Summary of Analysis Results for Plant Operating States." The staff found the applicant's description of each POS identified in DCD Table 19.1-92b is consistent with the guidance in SRP Chapter 19 and RG 1.200. Therefore, RAI 42-7945, Question 19-4, is resolved and closed.

Given the discussion above and the incorporation of the information requested into the DCD Tier 2 Revision 1, dated March 2017, the staff finds the applicant's success criteria analysis to be acceptable.

Data Analysis

The staff reviewed the data analysis to check basic event probabilities that are key contributors to the PRA numerical results and insights. The staff used the results of the applicant's importance analyses to guide what basic events needed additional review. The staff has no technical concerns with the applicant's data analysis, so the staff finds the applicant's data analysis to be acceptable.

Human Reliability Analysis

The staff reviewed operator error probabilities that are key contributors to the PRA numerical results and insights. The staff used the results of the applicant's importance analysis to guide what operator error probabilities needed additional review.

The top two cutsets are dominated by over draining of the RCS to reach mid-loop conditions. To mitigate this event, the operators need to initiate RCS injection and recover the shutdown cooling system. To quantify the failure rate of these operator actions, the analyst considers dependence as discussed in the trial use LPSD PRA Standard. The applicant considered dependence in the top two cutsets. However, the staff searched through the LPSD HRA notebook, APR-1400-K-P-NR-013705, Revision 0, and could not find how dependence was calculated or what factors were considered in dependence (e.g. similar alarms and cues). The staff did find the dependence calculations for other LPSD initiators in the LPSD HRA notebooks. Therefore, the staff issued RAI 409-8325, Question 19-27, dated February 22, 2016 (ML16053A015), requesting the applicant to provide additional information on how dependence was calculated for: (1) RCS over draining at reduced inventory operation, and (2) failure to maintain water level during reduced inventory operation.

In the final response to RAI 409-8325, Question 19-27, dated June 27, 2016 (ML16175A691), the applicant proposed to update the DCD Section 19.1.6.1.1.6 to state that some of the top cutsets include operator failures to perform multiple actions including, for example, a failure to restore the Shutdown Cooling System, followed by a failure to initiate feed & bleed operation. The Cause-Based Decision Tree Methodology (CBDTM) methodology, as implemented in the EPRI HRA Calculator (Reference 26), was used to evaluate the dependency of the subsequent operator action. The dependency itself was quantified with NUREG/CR-1278 "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," Table 10-2 equations. The staff found the licensee's methodology for estimating dependence among operator actions to be acceptable because it used the methodology described in NUREG/CR-1278. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, Section 19.1.6.1.1.6 was revised as committed in the response to RAI 409-8325, Question 19-27. Therefore, RAI 409-8325, Question 19-27, is resolved and closed.

The staff reviewed the dependency tables that the applicant proposed to add to the APR1400 PRA notebooks. Dependency was estimated using the following performance shaping factors: same crew, intervening successes, location, and timing. Low dependence was assigned to many of these operator actions due to timing (> 60 minutes of the preceding action). The staff finds the dependency analysis to be acceptable for DC, because it is consistent with the NUREG/CR-6883, “SPAR-H Reliability Analysis Methodology.”

Regarding RCS level instrumentation, the APR1400-E-N-NR-14005, Rev 0, “Shutdown Evaluation Report,” Section 2.8.3.2.1 states:

> “Four unique sets of instruments are provided for the measurement of level during RCS drain down and reduced inventory operations. These instruments make up the refueling water level indication system (RWLIS). The first set of instruments is a pair of wide-range, pressure differential (dP)-based level sensors. These sensors measure the level between the pressurizer (PZR) and the bottom of the hot leg during drain-down operations. Another pair of dP-based level sensors is used to determine RCS water level once it is within the reactor vessel. These narrow-range level sensors measure level between the direct vessel injection (DVI) nozzle and the bottom of the hot leg. The ultrasonic level measurement system measures from 20 percent to 100 percent of the hot leg level.”

During a loss of shutdown cooling at reduced inventory conditions, it was not clear to staff whether the operator reviews all three level indicators. If the RCS is vented via the pressurizer manway, and DHR is lost, RCS heat up and re-pressurization could result in hot leg inventory being entrained in the pressurizer. The wide-range level indication that is tapped into the pressurizer could read erroneously high. The staff issued RAI 268-8308, Question 19-12, dated October 22, 2015 (ML15295A262), requesting that the applicant provide additional information in the DCD on how this condition has been accounted for in the post-initiator HEPs.

The staff also noted that the details of the reduced inventory instrumentation package are referenced in the Shutdown Evaluation Report, APR 1400-E-N-NR-14005, Rev 0, which is not incorporated by reference (IBR) in the DCD and will not be reviewed by the staff. In RAI 268-8308, Question 19-12, the staff requested that the design details of the reduced inventory instrumentation package be added to the DCD Chapter 5 and Chapter 7, as applicable.

In the final response to RAI 268-8308, Question 19-12, dated November 2, 2016 (ML16307A369), the applicant updated DCD Section 5.4.7.2.6 to provide the details of the reduced inventory instrumentation package. This update includes details of the permanent refueling water level indication system (PRWLIS) wide range level indication and narrow range level indication. These details also include a discussion of alarms and where the indication is connected to the RCS. This update also includes details on ultrasonic level measurement system (ULMS). The applicant also added a risk insight to DCD Table 19.1-4, Risk Insights and Key Assumptions, stating, “the core exit thermocouples provide representative indications of the core exit temperature when shutdown cooling system (SCS) is operational, including reduced inventory operations. Continuous, redundant narrow range RCS water level indication is operational during reduced inventory operations.”

Regarding the staff’s concern of the accuracy of the wide range level indication post RCS boiling with a vented pressurizer, the applicant explained that the HRA does not specify the narrow range level transmitters, but implicitly credits procedural compliance in instrument use. In addition, the applicant referenced the APR1400 Emergency Operating Guideline DIAGNOSTIC ACTIONS, Section 6.0 which states "All available indications should be used to aid in evaluating plant conditions since the accident may cause irregularities in a particular instrument reading. Instrumentation readings must be corroborated when one or more confirmatory indications are available." The applicant concluded that guideline ensures that the operators will utilize the narrow range indications during any event that could potentially have an erroneously high wide range level indication.

The applicant's RAI response and proposed DCD updates to include the details of the reduced inventory instrumentation package resolve the staff concerns identified in Question 19-12. The staff confirmed that DCD Revision 1, dated March 2017, contains the changes committed to in the RAI response; therefore, RAI 268-8308, Question 19-12, is considered resolved and closed.

Given the discussion above and the incorporation of the information requested into the DCD Tier 2 Revision 1, dated March 2017, the staff finds the applicant's human reliability analysis to be acceptable.

Quantification

The staff reviewed the numerical results to ensure that significant contributors to CDF and LRF, such as risk significant POSs, initiating events, accident sequences, and basic events (equipment unavailabilities and human failure events) were identified and the CDF was quantified on a POS-by-POS basis.

Basic event probabilities were not included in the LPSD CDF cutsets and the LRF cutsets. Therefore, the staff requested this information in RAI 418-8348, Question 19-43 (previously discussed in this report). These LPSD basic event probabilities were subsequently provided in the response to RAI 434-8352, Question 19-92, on April 10, 2018. Based on the review of the DCD, the staff has confirmed incorporation of the basic event probabilities described above; therefore, RAI 434-8352, Question 19-92, is resolved and closed. The staff finds the applicant's quantification of CDF to be acceptable. The staff finds the applicants quantification of CDF to be consistent with SRP Chapter 19 and RG 1.200, and, thus, acceptable. The applicant's quantification of CDF also meets the Commission's goals described in the SRM to SECY-90-016 of 1E-4 per calendar year with significant margin.

Importance and Sensitivity Analyses

The applicant performed and reported in the DCD importance analysis for initiating events, basic events, CCF events, and operator failures.

In RAI 42-7945, Question 19-3, dated June 22, 2015 (ML16203A282), the staff requested that risk significant initiating events be reported in the DCD Section 19.1.6 according to FV, risk reduction worth (RRW), and RAW, since LPSD initiating events are often caused by operator errors. In the final response to RAI 42-7945, Question 19-3, dated September 30, 2015 (ML15273A319), the applicant added proposed DCD Tables 19.1-106 and 19.1-107, which report LPSD "Internal Events PRA Key Initiating Events" by FV and RAW, respectively. As expected, operator errors are significant. Overdraining of the RCS to reach midloop conditions resulting in a loss of SDC, has a RAW value of 170. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised to add Tables 19.1-105a and 19.1-105b "LPSD Internal Events PRA Key Initiating Events" by FV and RAW as requested by the staff and as committed in the response to RAI 42-7945, Question 19-3. Therefore, RAI 42-7945, Question 19-3, is resolved and closed.

Given the discussion above and the incorporation of the information requested into the DCD Tier 2 Revision 1, dated March 2017, the staff finds the applicant's importance and sensitivity analysis to be acceptable and consistent with SRP Chapter 19 and RG 1.200.

19.1.6.2 Level 2 Internal Events PRA for Low-Power and Shutdown Operations

Summary of Application

The application describes the methodology for quantitative evaluation of large release frequency for internal events for operations at LPSD. The applicant performed five technical tasks:

- Methodology discussion,
- Containment integrity and containment closure evaluation,
- Containment event tree analysis,
- Quantification of LRF, and
- Importance and sensitivity analyses.

As a guideline, the applicant used ANS/ASME-58-22-2014, "Low Power and Shutdown PRA Methodology," (LPSD PRA Standard) which has been issued for trial use and contains guidance for estimating LPSD LRF consistent with RG 1.200. The applicant estimated the total LRF from LPSD internal events to be 1.2E-7/year, which meets the Commission's goals described in the SRM to SECY 90-016 of 1E-6/calendar year. This LRF estimate includes the frequency of an outage per calendar year and the average duration in hours of each POS during an outage. Consistent with the at-power analysis, external reactor vessel cooling (ERVC) is not credited in the baseline LPSD Level 2 analysis.

The applicant estimated the CCFP from all LPSD internal events large release sequences to be 0.066. This estimate meets the Commission's goal of having a Conditional Containment Failure Probability (CCFP) as described in the SRM to SECY-93-087 of approximately 0.1 or lower.

Approximately 41 percent of the LPSD LRF results from POSs 5 and 11 (the mid-loop POSs). These POSs dominate the LPSD CDF (approximately 67 percent of the CDF). The LRF contribution from mid-loop operation reflects the applicant crediting the initiation of SI to arrest core damage in the vessel as a Severe Accident Mitigation Guideline (SAMG). The applicant credited this SAMG action in containment event tree top event, MELTSTOP. However, a key contributor to the LPSD CDF in the mid-loop POSs results from operator failure to initiate SI before core damage. Given the dependence of these two operator actions, the staff requested additional information. In RAI 409-8325, Question 19-28 (ML16053A015), the staff requested additional information on how dependence was calculated. The "Technical Evaluation" section below discusses this issue in detail.

Regarding containment closure, the applicant added a TS LCO 3.6.7, Containment Penetrations, during reduced inventory conditions for Modes 5 and 6. Reduced inventory conditions is defined as reactor vessel level less than 3 ft (0.9 m) below the reactor vessel flange. This TS requires the equipment hatch to be closed and held in place with a minimum of four bolts. One door in each personnel air lock is required to be closed. Penetrations providing direct access from the containment atmosphere to the outside atmosphere are also required to be closed or are required to be capable of being closed through an operable Containment Purge and Exhaust Isolation system. This feature significantly reduces LRF risk compared to operating PWRs.

The next two highest POS contributors to LRF are POS 3A and 3B. POS 3A represents cooldown with shutdown cooling system to 212 ˚F (100 ˚C) during Mode 4. POS 3A contributes 25 percent to the total LPSD LRF. POS 3A is evaluated using the at-power conditional probability of large release (CPLR) (which was rounded up to 0.10), since TS require containment integrity in Mode 4. POS 3B represents cooldown with the shutdown cooling system to 140 ˚F (60 ˚C) during Mode 5. POS 3B uses the full power CPLR but adds the likelihood of the operator failing to close the containment equipment hatch. The containment equipment hatch is allowed to be open in Mode 5 with RCS level above reduced inventory conditions. POS 3B contributes six percent of the total LPSD LRF.

The top six LRF cutsets occur in POS 5 (mid-loop prior to refueling). These cutsets initiate from a purification line rupture (JL sequences) or RCS overdraining to achieve midloop conditions. Post core damage failures include operator action to initiate safety injection as a SAMG action, core debris plugs the sump screens or the containment ruptures due to hydrogen deflagration.

The results of the parametric uncertainty analyses performed on the internal events LRF cutsets for LPSD operations are:

Five percent value:	2.7E-8/year
Mean value:	7.2E-8/year
Ninety-five percent value:	1.5E-7/year

The uncertainty analysis was performed using a Monte Carlo sampling, with a sample size of 5,000.

<u>Technical Evaluation</u>

Methodology

Consistent with the LPSD PRA Standard, SRP Section 19.0, and RG 1.200, the staff reviewed how core damage sequences are grouped into PDSs. The staff then reviewed how the accident progression analyses evaluated the contributors to a large early release. The staff focused on the evaluation of the containment structural capability for those containment challenges that would result in a large release. The staff reviewed the failure of the operator to close containment, when the containment was permitted by TS to be open, before adverse environmental conditions (e.g., temperature, radiation, humidity, and noise) prevent closure. The staff then checked the quantification of different containment failure modes leading to a large release. Finally, the staff checked the importance analyses results for risk insights.

The applicant did not combine Level 1 core damage sequences into PDS similar to the full power Level 2 evaluation. Relevant Level 1 sequence characteristics and system evaluations were incorporated directly in the LPSD CET. The applicant used a top logic fault tree for the CET quantification. This allowed the Level 2 system fault trees and the CETs to be directly linked to the Level 1 system models without the need for PDS.

The applicant developed a single CET model for POSs 4B, 5, 6, 10, 11, and 12A. In each of these POSs, DCD Section 19.1.6.2.1 states that the containment is closed per technical specifications and the pressurizer manway is open. For these POSs, the applicant considered the accident progression to be similar, and the accident progression was assessed with a single CET. Differences in timing are evaluated for HEPs. These differences are evaluated at the fault tree level and do not alter the CET structure. Since a severe accident with the pressurizer manway open results in hydrogen releases from the RCS that is different from the full power, the applicant considered the potential for hydrogen “pocketing.”

For POSs 1, 2, 3A, 13, 14, and 15, containment integrity is required by TS, and the RCS is intact. The LRF for POS 1-4A and 13-15 LRF are estimated using the at-power CPLR. The at-power CPLR is estimated by the applicant to be 0.1. Since the RCS is intact in these POS, the accident progression is assumed by the applicant to be similar to the at-power accident progression, except with lower energy levels. The staff accepts this assumption since it is conservative and simplifies the analysis without masking risk insights.

For POS 3B, cooldown with SCS to 140 °F (60 ˚C), the RCS is intact and the LTOP valves are in auto protection mode. For POS 3B, the applicant used the full power CPLR to estimate LRF with the added likelihood of the operator failing to close the containment equipment hatch. The equipment hatch is permitted to be open in Mode 5 with the RCS level above reduced inventory conditions. Failure to close the containment is assumed to result in a large release.

For POS 4A, RCS system drain down with the pressurizer manway closed, the reactor coolant system gas vent system is open and the LTOP valves are in auto protection mode. For POS 4A, the applicant also used the full power CPLR to estimate LRF with the added likelihood of the operator failing to close the containment equipment hatch. The equipment hatch is permitted to be open in Mode 5 by TS, with RCS level above reduced inventory conditions. Failure to close the containment equipment hatch is assumed to result in a large release.

In summary, the reported Level 2 results focus on the POSs with the RCS open to the containment atmosphere via the pressurizer manway that include POSs 4B, 5, 6, 10, 11, and 12A. The staff found this methodology to be consistent with the high-level requirements in the LPSD PRA Standard, SRP Section 19.0, and RG 1.200.

Containment Integrity and Containment Closure in LPSD

In Mode 4, hot shutdown, an integral containment is required by TS. Once the plant transitions to Mode 5, cold shutdown, an integral containment is no longer required. Therefore, the containment may be opened until reduced inventory conditions are reached. Reduced inventory conditions are defined as when reactor vessel level is less than three feet below the reactor vessel flange. The applicant added TS LCO 3.6.7, “Containment Penetrations” during reduced inventory operations in Modes 5 and 6. The TS requires the equipment hatch to be closed and held in place with a minimum of four bolts. One door in each personnel air lock is required to be closed. Penetrations providing direct access from the containment atmosphere to the outside atmosphere are required to be closed or required to be capable of being closed through an operable Containment Purge and Exhaust Isolation system. This feature reduces LPSD LRF risk compared to operating plants.

During Modes 5 and 6, when the plant is not in reduced inventory operation, the containment hatch may be opened according to TS. If the containment is open at the time of core damage in a severe accident, the applicant assumed the releases are large and unmitigated.

With ac power available (offsite or onsite), the applicant estimates the containment equipment hatch can be closed with at least four bolts in one hour. If no ac power is available (station blackout), the applicant does not credit hatch closure.

The applicant assumes the failure pressure of the equipment hatch is lower when fewer bolts are engaged. The equipment hatch is pressure-seated, meaning that an increase in containment pressure tightens the hatch seal. The at-power containment ultimate failure pressure was modeled as 162.7 psig in DCD Section 19.1.4.2.2.5. For modeling purposes, the LPSD analysis assumes that because of the smaller number of bolts securing the equipment hatch, the failure pressure will be less than half of the at-power ultimate capacity. A failure pressure of 80 psia (65.3 psig) is assumed. The applicant asserts that this assumed failure pressure has little effect on most of the LPSD Level 2 analysis. If the RCS has a slow pressurization and if containment heat removal (CHR) is available, the applicant states that containment pressure is maintained far below this level. The staff performed confirmatory calculations using MELCOR to confirm these assumptions. On March 01, 2016, the staff issued RAI 426-8492, Question 19-53 (ML16061A146), questioning the applicant's LPSD MAAP analyses based on the staff's confirmatory calculations using MELCOR.

On March 08, 2018, the applicant submitted their final response to RAI 426-8492, Question 19-53 (ML18067A847). Regarding RCS and containment pressurization in POS 5, the applicant explained that the steam generated in the RCS causes the pressure in the RCS to rise. As steam escapes the RCS, the containment will pressurize as well, but at a slower rate because of its large volume. The LPSD MAAP analyses were revised to incorporate a more accurate representation of the nozzle dams and releases from the RCS into containment. The analysis includes the potential for the nozzle dams to fail if their design pressure limit is reached, although it was not reached in the analyses. Reviewing the MAAP output files for case POS5, prior to core damage, the maximum primary system pressure is 0.232 MPa (34 psia or 19 psig). The maximum containment pressure is 16.5 psia or 1.8 psig. Additionally, based on the APR 1400 Fukushima Technical Report, APR1400-E-P-NR-14005, Revision 0, Figure 5-4 Containment Pressure for Loss of RHR (Mode 5), following a complete loss of core cooling during POS 5, the assumed LPSD ultimate failure pressure of 80 psia is not reached until roughly 30 hours into the event which allows sufficient time for containment heat removal through the ECSBS to be initiated. Thus, the staff agrees that the assumed containment failure pressure with four bolts (80 psia) has little effect on the LPSD Level 2 analysis. Therefore, the staff's concerns in RAI 426-8492, Question 19-53, are considered resolved and closed.

Regarding the required time for the operator to close the equipment hatch, on September 30, 2015, the staff issued RAI 232-7864, Question 19-5 (ML15273A181), requesting the applicant to justify reliable operator action for hatch closure given the presence of (1) steam, (2) high humidity, (3) low visibility due to fog, and (4) high temperatures. The staff also requested the applicant to clarify in DCD Chapter 19 whether the 160˚F (60 ˚C) upper limit specified in the APR1400 shutdown evaluation report was used to develop the likelihood of the operator failing to re-close the equipment hatch above reduced inventory conditions following a loss of decay heat removal.

In a supplemental RAI to Question 19-5, in RAI 440-8551, Question 19-95, dated March 11, 2016 (ML16074A286), the staff requested that the applicant also resolve the technical inconsistency between TS and DCD Section 19.1.6.2.1.1. According to TS 3.5.7, POS 4B, POS 6, POS 10, and POS 12A may have the containment hatch open. However, DCD Section 19.1.6.2.1.1 states that a single CET model is developed for evaluation of POSs 4B, 5, 6, 10, 11, and 12A. DCD Section 19.1.6.2.1.1 also states that for each of these POSs, "the containment is closed per TS and the pressurizer manway open, the accident progressions considered are similar and can be assessed with a single CET."

On September 6, 2016, the applicant submitted their final response to RAI 440-8551, Question 19-95 (ML16250A866). In the RAI response, the applicant stated that the applicability for TS 3.6.7 will be revised consistent with DCD Section 19.1.6.2.1.1. This revised TS would result in the DCD Chapter 19 being consistent with TS 3.6.7. The issue was subsequently subsumed into RAI 481-8546, Question 16-149 (ML16133A271).

On December 28, 2017, the applicant submitted their final response to RAI 481-8546, Question 16-149 (ML17362A080). The applicability for LCO 3.6.7 was revised to include MODE 5 with any RCS loops not filled and MODE 6 with the water level less than 7.0 m (23 ft) above the top of the reactor vessel flange. In LCO 3.6.7, a NOTE was added that states, “The equipment hatch shall be closed and held in place by a minimum of four bolts before opening the pressurizer manway.” The bases for LCO 3.6.7 was revised to state, “Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, or blind flange.” This revised RAI response results in the DCD Chapter 19 being consistent with LCO 3.6.7 thereby resolving staff’s concerns. Thus, RAI 232-7864, Question 19-5 and RAI 440-8551, Question 19-95 are resolved and closed. The staff finds the applicant’s LPSD containment integrity and containment closure evaluation to be acceptable.

LPSD Containment Event Tree Evaluation

As discussed in the methodology section of this report, the applicant developed a single CET model to evaluate POSs 4B, 5, 6, 10, 11, and 12A. In each of these POSs, the containment is closed per TS and the pressurizer manway is open to containment. For each of these POSs, the applicant considered the accident progression to be similar. Differences in timing are evaluated for HEPs, but only at the fault tree level, and do not alter the CET structure. With the RCS initially open, hydrogen releases from the RCS are assumed to be different from full power scenarios. Thus, hydrogen “pocketing” was considered. The staff reviewed the LPSD CET presented in DCD Figure 19.1-64. The top events are:

CDF Entry from LPSD Level 1 POS 4B-6 and 10-12A

BYPASS Containment not bypassed through the CVCS letdown line

The letdown line has a high-pressure alarm that is located downstream of the letdown control valves and warns the operator when the pressure is approaching the low-pressure system design pressure. When a warning is issued, the control room operator isolates the letdown line to terminate any further pressure. On February 22, 2016, the staff issued RAI 409-8325, Question 19-23 (ML16053A015), requesting the applicant to provide additional information (as discussed in Section 19.1.6.1 of this this SER), explaining how the closure of this valve is modeled during any postulated RCS re-pressurization when letdown is operating.

In the response to RAI 409-8325, Question 19-23, dated August 9, 2016 (ML16222A942), the applicant proposed an update to DCD Section 19.1.6.1.3, "Initiating Events," to state the ISLOCA initiating event was retained in the low power and transition modes, using the same frequency as the at-power PRA, and there is a negligible ISLOCA vulnerability once the reactor is depressurized. In the proposed DCD update, the applicant explained that prior to establishing a primary vent, a letdown line rupture or a diversion LOCA was examined for potential containment bypass vulnerability. The applicant concluded the letdown rupture is not a containment bypass vulnerability because there are restricting orifices downstream of the letdown isolation valves within containment, that limit letdown flow even with full RCS pressure during power operation. A rupture upstream of the orifices will occur within containment; a downstream rupture will result in a negligible break flow. Therefore the letdown isolation does not impact potential containment bypass. The staff accepted this conclusion in the proposed DCD update because it documents the design details that reduce the risk of ISLOCA events.

CISO Containment isolation successful

MELTSTOP The core melt arrested in-vessel

There is a SAMG action to initiate SI when the core exit thermocouples reach 1200 °F given failure to recognize the need for SI earlier in the accident sequence. If the early failure of SI resulted from hardware failures, then no credit is given to this recovery action. ERVC consistent with the full power Level 2 PRA is not credited in the baseline LPSD analysis. Given the dependence of these two operator actions, in RAI 409-8325, Question 19-28, dated February 22, 2016, the staff requested additional information on how dependence was calculated (ML16053A015).

In the applicant's fourth response to RAI 409-8325, Question 19-28, dated September 28, 2017 (ML17271A479), the applicant added COL 19.1(24) stating that the SAMGs are entered when the core exit thermocouples reach 1200F. COL 19.1(24) was added to DCD Chapter 19 and DCD Table 1.8-2

In the RAI response, the applicant acknowledged that LPSD CDF and LRF are highly dependent on the LPSD HEPs, as is expected for a LPSD PRA with many manual actions. The applicant performed an HEP dependency analysis using the same methodology as in the at-power PRA which includes identifying and incorporating dependency between Level 1 and Level 2 LPSD HEPs. If the applicant found the combination of all HEPs within a cutset would result in a total HEP of less than 1E-6 for the cutset, then a joint minimum HEP of 1E-6 was applied.

To evaluate the significant of the use of this minimum joint HEP of 1E-6, a sensitivity evaluation was performed on the LPSD internal events Level 2 PRA. The sensitivity utilized a floor HEP of 1E-5. The result was that the internal events LRF from POS 4B through POS12A increased from 7.0E-8/year to 9.3E-8/year, and the total LPSD internal events LRF (all POSs) increased from 1.2E-7/year to 1.4E-7/year. The applicant concludes the sensitivity demonstrates that the impact on the total LPSD LRF is small and would not alter the conclusions of the DCD. The staff agrees with this conclusion based on the staff review of the revised LPSD LRF cutsets. The applicant proposed revising DCD Section 19.1.6.2.2.7 to discuss this sensitivity calculation. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 409-8325, Question 19-28, is resolved and closed.

DCF No dynamic containment failure

This event is similar to the full power Level 2 CET event DCF except that the RCS pressure will be always low due to the open pressurizer manway.

ECF No early containment failure due to hydrogen burn ex-vessel steam explosion

This event is similar to the full power DET DCF event except that RCS pressure is considered low in all sequences. The conditional probability of containment rupture due to hydrogen detonation, when hydrogen control (passive autocatalytic recombiners [PARs]) is unavailable, is increased to .1 to account for equipment hatch being tightened with less than 40 bolts. The conditional probability of containment rupture is also set to .1 for hydrogen burn when steam is quenched by containment spray and hydrogen control fails.

CSLATE Late containment heat removal availability (containment spray or emergency containment spray backup system)

This event is similar to the full power CET event “CSLATE,” except that the fault trees use an increased probability of sump plugging.

Similar to the full power PRA, the ECSBS is assumed to be unavailable until 24 hours after the event initiates. Containment pressure must be below the failure pressure at 24 hours to allow success.

DBCOOL Molten core debris is cooled ex-vessel

The conditional probability of debris coolability is assumed to be the same in the LPSD as in the full power Level 2 analysis.

LCF No late containment rupture due to over pressurization, hydrogen detonation or hydrogen burn.

This event is similar to the full power DETLCF except all containment failures due to LCF are assumed to be ruptures. In addition, a hydrogen ignition source is always assumed to be available.

BMT No basemat melt-through

Basemat melt-through is modeled consistent with the full power Level 2 CET event, BMT.

Many of the severe accident phenomenological probabilities estimated from the at-power Level 2 analysis were used in the LPSD Level 2 analysis. The staff focused their LPSD review on the CET top events and probabilities that are unique to LPSD.

In the LPSD containment event tree, which describes POSs 4, 5, 6, 10, 11, and 12A, failure of the CVCS purification line (ISLOCA) and containment heat removal (containment sprays) is assumed to result in a large release from the CVCS path.

Regarding containment heat removal, the applicant improved TSs during Mode 5 operation with the RCS loops not filled and Mode 6 operation with low water level to require a containment spray pump to be operable.

Due the assumed lower containment ultimate failure pressure at the equipment hatch with four bolts, the probability of containment failure due to hydrogen detonation is increased to 0.1 and that of containment rupture due to hydrogen burns is estimated to be 0.1 or 0.01. In DCD Section 19.1.6.2.2.5, "Key Assumptions," one of the assumptions (B) states:

> Failure of hydrogen control from PARs and/or igniters is assumed to yield a conditional probability of containment rupture due to hydrogen detonation of 0.1, plus another conditional probability of containment rupture due to hydrogen burn of 0.1 or 0.01. These probabilities are believed to be conservative, but additional calculations are needed for confirmation.

On September 30, 2015, the staff issued RAI 232-7864, Question 19-7 (ML16203A437), requesting that the applicant provide the results of the additional calculations in the DCD documenting the CCFP due to hydrogen. These calculations impact the LPSD LRF. The staff needed to compare total LRF against the Commission's goals for new reactors as directed in the SRP for Chapter 19.

On January 29, 2018, the staff received the applicant's final response to RAI 232-7864, Question 19-7 (ML18029A825). During the audit the staff also reviewed technical report FAI/14-0990, "Analysis of Hydrogen Distribution and Deflagration to Detonation Transition (DDT) Potential in the APR1400. During LPSD Severe Accidents", Revision 0, dated December 2014. The report states that for all cases, using artificial hydrogen "tails" to generate hydrogen equivalent to 100 percent metal water reaction (MWR), the 10 percent hydrogen limited exceedance does not happen if the igniters or PARs are available (with containment sprays actuated). For all cases using mechanistic in-core hydrogen, the 10 percent hydrogen limit is not exceeded in any combination of mitigative features including when none of the mitigative features are available (e.g. PARS, igniters, containment sprays).

In the applicant's final response to RAI 232-7864, Question 19-7, the applicant proposed an update to DCD Section 19.1.6.2.2.5 to state:

> For sequences with low steam concentrations and a significant generation of hydrogen, failure of hydrogen control from PARs and/or igniters is conservatively assumed to yield a conditional probability of containment rupture due to hydrogen detonation of 1.0 in the late containment failure (LCF) decomposition event tree. For similar sequences with success of PARS, a detailed evaluation has shown that hydrogen accumulation does not reach appreciable levels. However, for conservatism, the LPSD Level 2 analysis assumes such sequences have a 10% probability of containment rupture due to hydrogen burns, plus an additional 10% probability of containment rupture due to hydrogen detonation, given no failure due to burns. These probabilities have been demonstrated to be conservative.

The applicant also added risk insight 69 to DCD Table 19.1-4, "Risk Insights," by stating, "The HG design is composed of 30 PARs and 8 igniters. The LPSD Level 2 PRA assumes that 25 percent of the PARs are unavailable due to test and maintenance. The remaining 75 percent of the PARs and all of the igniters are available during LPSD conditions."

The staff confirmed that DCD Tier 2 Revision 2, dated March 2018, was revised as committed in the response to RAI 409-8325, Question 19-7. Based on the staff's review of the technical report and the applicant's proposed DCD updates, the staff finds the applicant's assessment of hydrogen and the impact on the CCFP to be acceptable. Therefore, RAI 409-8325, Question 19-7, is resolved and closed.

The staff finds the applicant's LPSD CET evaluation to be acceptable and consistent with RG 1.200. Based on the review of the DCD to verify that it incorporated the text as shown in the RAI response, the staff has confirmed incorporation of the changes described above; therefore, RAI 409-8325, Question 19-28, is resolved and closed

Quantification of LPSD LRF

Four release categories in the LPSD CET were identified and quantified:

RC-1 Containment is intact and there are no significant releases.

RC-2 The containment ruptures with large and early releases. The main contributors are containment bypass or failure to close the equipment hatch in POS3 and POS 4B. Other contributors include hydrogen detonation, hydrogen burn, and ex-vessel steam explosion.

RC-3 The containment ruptures with large and late releases. The dominant contributors include steam over-pressurization (late CHR failure), late hydrogen detonation, and late hydrogen burn.

RC-4 The category includes those sequences where the containment fails late due to basement melt-through. The releases in this category are assumed to be late and small.

The LPSD LRF is the combined frequencies of RC-2 and RC 3.

When evaluating the LPSD LRF results in the DCD Revision 0, the staff noted that the LPSD LRF for internal events was calculated to be 1.2E-7/year. However, the mean value was estimated to be 6.8E-8/yr. There is a factor of two difference in these values. The staff expected that the state of knowledge correlation would always cause the mean to be greater than a point estimate. In RAI 446-8535, Question 19-100, dated March 16, 2016 (ML1607A030), the staff asked the applicant to address this difference.

In the final response to RAI 446-8535, Question 19-100, dated October 28, 2016 (ML16302A495), the applicant updated the DCD to state that the uncertainty analysis is only performed on the LPSD LRF cutsets generated for POSs 4B through 12A. Since the LRF for the other POSs was conservatively approximated (as described in DCD Section 19.1.6.2.1.1), no LPSD LRF cutsets were generated for those POSs. The applicant also updated the DCD to provide the LRF point estimate for POSs 4B-12A for direct comparison against the mean. The LRF point estimate is calculated to be 7.0E-8/year. The staff finds these DCD updates to be acceptable to resolve staff's questions regarding the uncertainty analysis. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the response to RAI 446-8535, Question 19-100. Therefore, RAI 446-8535, Question 19-100, is resolved and closed.

The applicant did not evaluate modeling uncertainty in the LPSD PRA Level 2 assessment. The staff's review focused on Level 2 modeling issues that could be key sources of uncertainty for POSs 4B through 12A. These POSs cover shutdown operations with the pressurizer manway open. The staff's review specifically focused on top events in the CET for POSs 4B though 12A that: (1) did not use simplifying assumptions from the full power assessment, (2) significantly impact the LPSD LRF, and (3) are unique to LPSD operations. As discussed in previous sections, the staff asked for additional information for the following CET top events: containment bypass scenarios, core damage scenarios arrested in the vessel, and containment heat removal. As discussed in the paragraph titled, "LPSD Containment Event Tree Evaluation", the risk impact of human error dependency modeling between the Level 1 results and the Level 2 results is addressed in RAI 8325, Question 19-28 (ML17271A479). Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 409-8325, Question 19-28, is resolved and closed.

Since POSs 3B and 4A use the Conditional Probability of Large Release (CPLR) from full power and the likelihood of containment equipment hatch closure, the staff's review of modeling uncertainty focused on the modeling of equipment hatch closure. The staff's concerns with the modeling of hatch closure are discussed in the paragraph titled, "Containment Integrity and Containment Closure in LPSD."

Based on the discussion above, the staff finds the applicant's quantification of LPSD LRF to be acceptable.

Importance and Sensitivity Analysis

The applicant performed and reported in its DCD the importance analysis for basic events and operator failures. As discussed previously, in response to RAI 409-8325, Question 19-28, the applicant performed an HEP dependency analysis using the same methodology as in the at-power PRA, which includes identifying and incorporating dependency between Level 1 and Level 2 LPSD HEPs**.** The staff finds the applicant's quantification of LPSD importance and sensitivity analyses to be acceptable.

19.1.6.3 Seismic Risk Evaluation during Low Power and Shutdown Operations

Summary of Application

The applicant used the same methodology to perform the LPSD PRA based SMA as for the full power PRA based SMA. The RLE for the LPSD PRA-based SMA is the same as that for the full power PRA-based SMA as described in DCD Section 19.1.5.1.1. The SEL is developed from the full power SEL. The applicant then reviewed shutdown conditions to identify SSCs which could impact the plant if a seismic event occurs during LPSD conditions. These SSCs were added to the LPSD SEL. The applicant used the same fragility analysis methodology as for the at-power PRA-based SMA as described in DCD Section 19.1.5.1.1(e). The LPSD system model and accident sequence analysis is based on the LPSD IEs PRA. The applicant used the same methodology for evaluating the LPSD plant seismic capacity as for the at-power PRA-based SMA as described in DCD Section 19.1.5.1.1(f).

Technical Evaluation

In RAI 232-7864, Question 19-10, dated September 30, 2015 (ML16203A43), the staff requested the applicant perform a PRA-based SMA for LPSD to determine the seismic capacity of the plant and for each sequence that may lead to core damage or large release. The staff also requested the applicant to update DCD Section 19.1.6 to discuss how the seismic margins approach was applied to low power and shutdown conditions and to provide results and risk insights. In the applicant's final response to RAI 232-7864, Question 19-10, dated April 11, 2018 (ML18101A794), the applicant updated DCD Section 19.1.6.5 to include the LPSD PRA-based SMA with event trees, results, and key risk insights. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 232-7864, Question 19-10, is resolved and closed.

Methodology and LPSD Plant Operating State (POS) Development for LPSD SMA

The staff's review of the applicant's LPSD PRA-based SMA focused on aspects unique to LPSD conditions. The applicant used the same methodology for the at-power PRA-based SMA methodology for LPSD conditions. This method was used to evaluate POSs 3B to 13 which is based on the same POS definitions used in the LPSD IEs PRA, as referenced in Table 19.1-41B, "LPSD SMA Plant Operating States." These POSs include: cooldown with shutdown cooling system (SDC) from 212 F to 140 F with the RCS intact, draining of the RCS with the pressurizer manway closed, draining of the RCS with the pressurizer manway open, installing steam generator nozzle dams, removing the reactor vessel head and raising RCS level for refueling, draining the RCS to re-install the reactor vessel head, draining to remove steam generator nozzle dams and refilling the RCS for restart. This analysis is a margin analysis and not a PRA. POSs 3B to 13 were grouped into two different plant operation conditions depending on whether the RCS is intact or the RCS is not intact/open via a removed pressurizer manway or removed reactor vessel head. POSs 3B and 4A were assessed as POS 3B, which represents cooling the RCS to reach 140°F with the RCS intact. Since the containment hatch is open in POSs 3B and 4A all core damage scenarios in POS 3B and 4A are assumed to go directly to large release (LR).

POSs 4B through 12A were represented as POS 5, midloop conditions before refueling with the RCS vented via an open pressurizer manway. In POSs 4B-12A, the containment is assumed to be closed based on TS LCO 3.6.7. The post refueling POSs, POSs 12B and 13 are very similar to POSs 4A and 3B with normal RCS inventory and sub-cooling. However, the thermal-hydraulic analysis shows that the time to core damage is more than 24 hours after a loss of SCS because the decay heat of POSs 12B and 13 is so low. Therefore, the applicant screened out POSs 12B and 13. The staff finds this screening acceptable since POS 3B is bounding for POSs 12B and 13.

The staff finds the use of the full power methodology to address POSs 3B and POS 5 to generate risk insights from the LPSD PRA-based SMA to be acceptable. POS 3B and POS 5 are the bounding POSs for the RCS intact conditions and RCS open conditions. This methodology is also acceptable since this analysis is a margin analysis not a LPSD seismic PRA, where the risk contribution from every POS has to be evaluated.

Seismic Hazard Input

The seismic hazard input used for the LPSD PRA-based SMA is identical to the seismic hazard input used for the at-power PRA-based SMA described in DCD Section 19.1.5.1.1(A). As described in Section 19.1.5.1 of this SER, the seismic input (i.e. RLE) established for the PRA-based SMA is consistent with DC/COL ISG-020, and therefore is acceptable.

LPSD Seismic Equipment List

The staff reviewed the applicant's SEL for LPSD conditions DCD Table 19.1-42A, "LPSD Seismic Equipment List," which was developed from the at-power SEL. The staff also reviewed the LPSD PRA-based SMA failure modes and effects analysis (FMEA) results submitted in the response to RAI 232-7864, Question 19-10. The applicant searched for SSCs which could impact the plant should a seismic event occur during LPSD conditions. LPSD specific seismic failures identified include seismic induced failure of the containment hatch when it is opened and stowed in POSs 3B and 4A. Failure is assumed to lead to direct core damage and release due to catastrophic RCS failure and loss of containment integrity. Reactor vessel head (removed and stowed) during POSs 6 and 10 is described Table 19.1-43A, “LPSD Seismic Fragility Analysis Results Summary,” as seismic induced failure during reactor vessel head movement. Failure is assumed to lead direct core damage and loss of containment integrity. The polar crane and Jib crane are also included in the LPSD SEL. The staff found the LPSD SEL to be sufficiently complete for DC.

Seismic Fragility Evaluation

The applicant used the same fragility analysis methodology as for the at-power PRA-based SMA as described in DCD Section 19.1.5.1.1(e). Additional assumptions for the LPSD PRA-based SMA fragility analysis are described in DCD Section 19.1.6.5.5, “Seismic Fragility Analysis.” The staff evaluation of the methodology is described in Section 19.1.5.1 of this SER.

Consistent with the description of the SSCs added to the LPSD SEL as described in Section 19.1.5.1 above, the applicant included the following additional SSCs in the LPSD seismic fragility evaluation.

- Containment Hatch
- Reactor Vessel Head
- Polar Crane
- Jib Cranes

The HCLPF capacities are assumed to be equal for or greater than 1.67 times the CSDRS as shown in Table 19.1-43A for these SSCs.

The staff determined that the assumptions above are acceptable for the DC stage on the basis that the COL applicant, as stated in COL 19.1(8), is to confirm the applicability of these assumptions and is to evaluate and update the PRA-based SMA and ensure that the results of the PRA-based SMA remain valid.

LPSD SMA Systems and Accident Sequence Analysis

LPSD system logic model development is same as for the at-power PRA-based SMA which is described DCD Section 19.1.5.1. The staff reviewed the LPSD seismic IEs event tree documented in DCD Figure 19.1-209. Seismic accident sequences were developed from this event tree for POSs 3B and 5 as described previously.

Similar to full power, the first top event is direct core damage (S-DMG) to model seismic failure of structures such as the auxiliary building, the turbine building, compound building, containment, etc. A seismic event that causes significant RCS component failure is assumed to lead to an excessive LOCA which cannot be mitigated by the emergency core cooling system (ECCS). These RCS component failures are assumed to lead to direct core damage. These seismic induced RCS component failures are also included in top event, S-DMG.

Similar to the full power SMA, loss of all instrumentation and control (S-IC) and total loss of component cooling water (S-TLOCCW) are assumed to result in direct core damage. Large and medium LOCAs are also assumed to result in direct core damage due to lower assumed HCLPFs for mitigation, similar to the full power PRA-based SMA.

Seismic induced initiating events based on the internal LPSD PRA model include: seismic induced loss of offsite power including station blackout out (SBO); ISLOCAs, defined as seismic induced failure of components in the SCS line connected to the RCS outside containment; seismic induced small LOCAs (S-SLOCA), defined as seismic induced failure of RCS small piping or tubing lines inside of containment; and non-recoverable LOCA (SNRLOCA), defined as seismic induced failure of the CVCS let-down line outside containment.

The staff then reviewed the seismic LPSD event trees. SBO and ISLOCA events are assumed to lead directly to core damage. Non-recoverable LOCAs credit the operator isolating the letdown line, restoring makeup, and restarting the shutdown cooling system. Non-recoverable LOCA core damage sequences are conservatively assumed to lead to containment failure. No credit is given for isolation of small LOCAs or restart of the shutdown cooling system; thus, manual initiation of safety injection is required to prevent core damage.

The staff found the applicant's seismic induced initiating event analysis, seismic event tree development, and accident sequence development to be technically adequate for DC because it meets the guidance in SECY-93-087.

Key Assumptions for LPSD SMA

DCD Section 19.1.5.1.1.4.9 provides the key assumptions used in APR1400 full power PRA-based SMA. Key assumptions unique to the LPSD SMA include:

1. The applicant assumed that seismic induced loss of the nozzle dams does not result in a loss of the minimum RCS inventory needed to support the shutdown cooling system. However, a seismic event is assumed to lead to a loss of offsite power resulting in a loss/interruption of the operating shutdown cooling system.

2. The containment structure is assumed to have the same HCLPF capacity in LPSD configurations as it does for the at-power fragility calculation. Specifically, collapse of the structure is not affected by whether or not the equipment hatch is removed or installed with four bolts.

Based on its review, the staff finds that the key assumptions used in the APR1400 LPSD PRA-based SMA are reasonable for the DC phase. Furthermore, per COL information items, these assumptions are to be reevaluated and dispositioned during the COL phases to ensure that the results and insights continue to remain valid.

HCLPF Sequence Assessment

This step is identical to that used for at-power PRA-based SMA as described in DCD Section 19.1.5.1 which is acceptable for this LPSD PRA-based SMA.

Results and Risk Insights from LPSD Seismic Risk Evaluation

The plant level HCLPF as supported by the top cutsets provided in DCD Tables 19.1-44E through N is 0.5g. The plant level HCLPF demonstrates the APR1400 design can withstand a review level earthquake of 1.67 times the CSDRS, and the assessment of the seismic capacity of the APR1400 design meets the intent of SECY-93-087.

Seismic induced structure failure and RCS catastrophic failure of the RCS, initiating event, S-DMG, HCLPF is 0.5g. The APR1400 design has three components, the Shutdown Cooling Pump, the Shutdown Cooling Heat Exchangers, and the Shutdown Cooling pump mini-flow heat exchanger, where seismic induced failure leads to S-ISLOCA resulting in core damage. The HCLPF for each component is 0.5g.

Regarding SBO, the HCLPFs for EDG building and diesel fuel oil tank building are higher than that for the auxiliary building and nuclear island (common basemat of the containment building and auxiliary building). The failure of all three buildings would cause the failure of all EDGs, and it would lead to SBO and core damage. The APR1400 design has thirteen components where seismic induced failure of any one component results in a seismic induced SBO leading to core damage. The HCLPF of the each component is 0.5g. The APR1400 instrumentation and control design has four components that failure of any one of these four components would lead to a loss of instrumentation and control resulting in core damage. The HCLPF of the each component is 0.5g.

For non-recoverable LOCAs and Small LOCAs, similar to the LPSD IEs PRA, operator actions are significant. The operator is required to manually initiate safety injection for small LOCAs to prevent core damage. For non-recoverable LOCAs, operator action is required to initiate RCS make-up, isolate the CVCS drain path, and restart the SCS. Failure for the operator to perform these tasks successfully requires manual safety injection to prevent core damage.

The staff reviewed the LPSD seismic cutsets and confirmed that no accident sequence has a HCLPF lower than 0.5g. The sequence-level and plant-level HCLPF assessment has provided confidence that the APR1400 design will withstand an earthquake of at least 0.5g intensity and achieve safe shutdown without damage to the reactor core.

Combined License Information

Table 19.1.5-1, above, documents the PRA-based SMA-related COL information item numbers and descriptions from DCD Tier 2 Table 1.8-2.

The staff finds that the applicant established these COL information items in accordance with the SRP Section 19.0 and DC/COL-ISG-020, and therefore they are acceptable. The staff concludes that no additional COL information items relevant to APR1400 seismic evaluation are necessary.

19.1.6.4 Internal Fire PRA for Low-Power and Shutdown Operations

Summary of Application

Section 19.1.6.3 of the DCD describes the LPSD internal fire risk evaluation and its results. The application includes a description of the LPSD internal FPRA and its results including any risk insights. The DCD states that the LPSD internal FPRA methodology is based on NUREG/CR-7114, "A Framework for Low Power/Shutdown Fire PRA," and NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities." The applicant performed the LPSD internal fire PRA using the full-power FPRA methodology with the exception that it was applied to the LPSD internal events model. As was the case for the full-power FPRA, certain design details were not available at the DC stage. Therefore, the applicant performed the LPSD FPRA using conservative and simplifying approaches precluding the need to perform certain tasks from NUREG-6850. The applicant performed the LPSD fire HRA by applying the methodology described in NUREG/CR-1921 to the LPSD internal events HRA.

The applicant reported the following LPSD internal fire CDF results:

5 percent value:	5.7E-7 per year
Mean value:	1.2E-6 per year
95 percent value:	2.5E-6 per year

The applicant reported the following LPSD internal fire LRF results:

5 percent value:	2.5E-8 per year
Mean value:	6.7E-8 per year
95 percent value:	1.5E-7 per year

Technical Evaluation

In accordance with SRP Section 19.0, the staff conducted its review to determine whether the technical adequacy of the LPSD FPRA is sufficient to justify the risk estimation and identification of risk insights that are used to support the DC application. To evaluate the technical adequacy, the staff reviewed the extent to which the applicant's LPSD FPRA information is consistent with the applicable methods described in NUREG/CR-6850, NUREG/CR-7114, and SRP Section 19.0. NUREG-7114 provides a framework for performing LPSD fire PRA and assumes that the full power fire PRA and LPSD internal events PRA have already been completed. As was the case for the full power FPRA, the staff recognized that the applicant either did not perform certain tasks or used simpler analyses than suggested in NUREG-6850 since certain design details (e.g., specifics of cable routing, ignition sources, target locations) and operating procedures are unavailable at the DC stage. The staff focused on ensuring that the LPSD fire risk compares favorably against the Commission's goals, as described in SRP Section 19.0, and that sufficient risk insights are identified to support the DC application. In particular, the staff's review focused on the changes made to the full-power FPRA and the internal events LPSD PRA models to implement the LPSD FPRA model.

Key Assumptions

DCD Section 19.1.6.3.1.2 lists the key assumptions used in the LPSD FPRA. The applicant applied many of the conservative full-power FPRA assumptions to the LPSD FPRA, which the staff finds reasonable. For example, below are some key assumptions that are common between both the full power FPRA and LPSD FPRA:

- For single-compartment fire analyses, all unsuppressed fires propagate throughout the entire compartment, damaging all PRA-credited equipment within
- Fire compartments, fire barriers and fixed ignition sources from the full-power FPRA are applicable to the LPSD FPRA
- Certain cables were assumed to have fire protection features to prevent damage or spurious operation of related components. The applicant identified Item COL 19.1(25) to ensure that the fire protection features required for preventing fire-induced damage of the PRA-credited components will be properly incorporated in the design
- Fire damage to control cables result in the worst-case failure mode for the affected equipment. Fire damage to fiber-optic cables result in failure to operate the associated equipment but does not cause spurious operations
- Assumptions related to the full-power MCR evacuation analysis are applicable to the LPSD MCR evacuation analysis
- Fire damage to instrumentation is limited to the division in which the fire occurred, and operators will be trained to rely on undamaged instrumentation once the location of the fire is known. The applicant identified Item COL 19.1(16) to ensure development of procedures and operator training to rely on undamaged instrumentation when the location of fire is known.

Some key assumptions specific to LPSD FPRA are listed below:

- Transient ignition frequencies account for conditions representative of an outage, using the NUREG/CR-7114 generic data as well as modified transient influence factors
- Removable walls/floor slabs in the auxiliary building are only removed for major equipment replacement (e.g., large pump or motor) during defueled conditions. The applicant identified Item COL 13.5(7) as providing appropriate provisions in the shutdown procedures
- During maintenance, fire doors and dampers may be propped open to facilitate the passage of temporary hoses or temporary power lines. The applicant assumed that the fire barrier management procedures used during LPSD will provide assurance that breached risk-significant fire barriers can be closed in sufficient time to prevent the spread of fire across the barrier, including the use of a fire watch as appropriate. The applicant identified Item COL 19.1(15) to ensure development of appropriate fire barrier management procedures.

The staff reviewed the applicant's PRA information for acceptability of the fire ignition frequencies assumed in the LPSD internal fire PRA. The applicant estimated the fire ignition frequency for each identified ignition source and each fire compartment using the generic frequencies from NUREG/CR-7114. On February 23, 2016, the staff issued RAI 418-8348, Question 19-45, to ensure that the impact of ignition frequency uncertainty is adequately addressed in the DCD (ML16054A291). RAI 418-8348, Question 19-45, was subsequently subsumed into RAI 434-8352, Question 19.1-92. In its response, the applicant proposed to perform these evaluations as a part of the PRA update process. DCD Section 19.1.2.4 describes the PRA maintenance and upgrade process and includes provisions to ensure the adequacy of the PRA model commensurate with its intended use. The DCD also includes provisions for the COL applicant to describe their PRA maintenance and upgrade program [COL 19.1(6)], and for the COL holder to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA remain valid [COL 19.1(4)]. Based on these considerations, the staff finds the applicant's approach adequate and RAI 418-8348, Question 19-45, is resolved and closed.

The staff reviewed the HRA performed to support the LPSD internal fire PRA. The applicant modified the internal events LPSD HEPs consistent with NUREG-1921 to address the potential effects of the fire on the ability of the operators to execute an action in the context of the fire. The results of the LPSD internal fire risk evaluation show that LPSD internal fire is highly dependent on operator actions with most of the top 100 cutsets only containing operator action failures and very few containing random equipment failures. On February 23, 2016, the staff issued RAI 418-8348, Question 19-45, requesting the applicant to provide an assessment of the uncertainty in the assumed HEPs (ML16054A291). RAI 418-8348, Question 19-45 was subsequently subsumed into RAI 434-8352, Question 19-92. In its response, the applicant indicated that the detailed sensitivity and uncertainty analysis will be performed as part of the PRA maintenance and upgrade process as described in DCD Section 19.1.2.4. The staff confirmed, by audit of the underlying PRA documentation, that the applicant identified sources of modeling uncertainty such as equipment failure rates and human error probabilities, that should be evaluated as additional information on design and procedures as well as operating experience become available. The DCD also includes provisions for the COL applicant to describe their PRA maintenance and upgrade program [COL 19.1(6)], and for the COL holder to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA remain valid [COL 19.1(4)]. Based on these considerations, the staff finds the applicant's approach adequate and RAI 418-8348, Question 19-45, is resolved and closed.

Fire-Induced Risk Model Changes to Support LPSD FPRA

The LPSD FPRA model builds upon the models for the full-power FPRA and the internal events LPSD PRA. Hence, the staff focused on the changes implemented to support the development of the LPSD internal FPRA.

Only a few additions relative to the full power FPRA were necessary to develop the LPSD FPRA component list. The applicant started with the full power FPRA basic events list and added any basic events from the LPSD internal events PRA that were not already contained in the full power FPRA basic events list. The applicant identified 10 additional basic events for cable identification and routing so they can be credited in the LPSD FPRA. The applicant then added the additional basic events to support the LPSD fire-induced risk model, including the logic for fire-induced initiating events.

DCD Section 19.1.6.3.1.3 describes the LPSD fire-induced initiating event analysis. The applicant reviewed the initiating events modeled in the internal events LPSD model for applicability to LPSD FPRA. The applicant identified the following eight potential initiators.

- S2 – Unrecoverable loss of the operating shutdown cooling train
- SL – Loss of level control (POS 5 and 11 only)
- SO – Overdrain events (POS 5 and 11 only)
- JL – CVCS letdown line diversion LOCA
- CC – Loss of operating component cooling train
- KV – Loss of 4 kV bus on operating shutdown cooling train
- LP – Loss of offsite power impacting operating shutdown cooling train
- AS – Fire-induced main control room evacuation (alternate shutdown).

In cases where there is a potential for more than one fire-induced initiator to occur in a given fire compartment, the applicant established a hierarchy of initiating events "wherein perceived worst-case initiators were given preference over lesser initiators." On February 23, 2016, the staff issued RAI 418-8348, Question 19-50 requesting that the applicant provide additional basis in the DCD for establishing this hierarchy and explain how this assumption may impact the PRA (ML16054A291). The applicant provided a response dated August 18, 2016, explaining how each initiator ranks qualitatively with respect to the degree of challenge to the plant, considering factors such as initial inventory, mitigation actions required, and equipment availability (ML16231A501). The applicant's approach reasonably encompasses the risk contribution from all applicable initiating events and, therefore, is acceptable. The staff confirmed that DCD Tier 2 Revision 1, dated March 2017, was revised as committed in the RAI response. Accordingly, the staff considers RAI 418-8348, Question 19-50, resolved and closed.

LPSD Internal Fire Risk Results and Insights

For the purposes of DC, the staff finds that the applicant's LPSD FPRA is sufficiently consistent with the guidance in NUREG/CR-7114, NUREG/CR-6850 and SRP Section 19.0. Therefore, the staff concludes that the APR1400 LPSD internal fire risk is consistent with the Commission's CDF and LRF goals.

The results of the LPSD internal fire risk evaluation show that the most important LPSD internal fire accident sequences involve a fire in a diesel generator room in either quadrant C or D of the auxiliary building (F000-ADGC or F000-ADGD). These sequences contribute about 25 percent to the LPSD internal fire CDF and about 32 percent to the LPSD internal fire LRF. Other risk-significant fire compartments include electrical equipment room F137-ANEA, corridor F078-A19B and general access areas F120-AGAC and F120-AGAD. Fires in the essential service water building (FK-K01) and the component cooling water heat exchanger building (FD-D01A) are also risk-significant. In terms of the POS, the hot mid-loop condition (POS 5) contributes about 8.5E-7 per year or about 68 percent to the LPSD internal fire CDF and about 2.1E-8 per year or about 22 percent to the LPSD internal fire LRF.

Operator actions to restore the SCS and to provide coolant makeup are both risk-significant. In addition, the LPSD internal fire Level 2 analysis results showed that the SAMG action to initiate of safety injection and the action to align the CFS are both risk-significant operator actions.

As is the case with the at-power internal fire analysis, the results of the internal fire risk evaluation reflect the APR1400 design features that minimize the impact from any single fire in the auxiliary building. These design features include, for example, the quadrant separation of the redundant safety equipment and use of 3-hour fire rated barriers. Also, the use of fiber-optic cables between the main control room safety console and the equipment minimizes the impact of fire-induced spurious operation.

Based on the foregoing discussions, the staff finds that the APR1400 internal fire risk compares favorably against the Commission's goals, as described in SRP Section 19.0, and that sufficient risk insights are identified to support the DC application.

19.1.6.5 Internal Flooding PRA for Low-Power and Shutdown Operations

Summary of Application

Section 19.1.6.4 of the DCD describes the LPSD internal flooding risk evaluation and its results. The application includes a description of the LPSD internal flooding PRA, including any risk insights. The DCD discusses the plant operating state development, initiating event analysis, accident sequence analysis, success criteria, operator actions, systems analysis, and quantification. The DCD states that relative to the LPSD internal events model, no changes were made to the success criteria, the human error probabilities, or the systems analysis, for the LPSD internal flooding model. Operator actions to isolate pipe breaks were taken from the at-power internal flooding model.

The applicant reported the following LPSD internal flooding CDF results:

5 percent value:	3.2E-8 per year
Mean value:	8.4E-8 per year
95 percent value:	1.8E-7 per year

Technical Evaluation

In accordance with SRP Section 19.0, the staff conducted its review to determine whether the technical adequacy of the LPSD internal flooding PRA is sufficient to justify the risk estimation and identification of risk insights that are used to support the DC application. The staff reviewed the applicant's LPSD internal flooding PRA to the extent possible, using RG 1.200 and SRP 19.0. The applicant had subjected the at-power internal flooding PRA to a peer review against the ASME/ANS PRA standard requirements. The staff's review considered the results of this peer review in the staff evaluation of LPSD internal flooding analysis since certain tasks from the at-power internal flooding analysis are applicable to the LPSD internal flooding analysis. The peer review generally found that the internal flooding ASME/ANS requirements are met for at least Capability Category I. The staff's review focused on ensuring that the LPSD internal flooding risk is compares favorably against the Commission's goals, as described in SRP Section 19.0, and that sufficient risk insights are identified to support the DC application. Since the LPSD internal flooding analysis builds upon the at-power internal flooding analysis and the internal events LPSD analysis, the staff's review focused on the changes made to those models to implement the LPSD internal flooding model.

Differences Relative to At-Power Internal Flooding PRA

The applicant stated that the at-power internal flooding analysis forms the basis for the LPSD internal flooding analysis. Subjects such as flood area definition, identification of flood sources and propagation paths, and plant operating states are largely unchanged from the at-power internal flooding analysis and the internal events LPSD analysis. A notable difference between the at-power internal flooding PRA and the LPSD internal flooding PRA is that the at-power analysis evaluated those scenarios that resulted in a reactor trip, whereas the LPSD analysis evaluated only those scenarios that lead to a failure of the running train of the SCS. The staff finds this assumption to be reasonable since it is unlikely that a flooding event would cause an inadvertent opening of a valve or other conditions that result in a loss in RCS inventory.

Since the initiating events for the LPSD internal flooding analysis are limited to those events that cause flood damage to the running SCS train, the plant operating states covered only POSs 3 to 6 and 10 to 13, where the SCS provides the decay heat removal function. POSs 1, 2, 14, and 15 rely on the main steam system for decay heat removal and have been screened out for internal flooding. In addition, POSs 7 to 9 have been screened out since either the cavity is flooded or the reactor is defueled.

Another difference in assumption between the at-power flooding analysis and the LPSD flooding analysis was the break flow rate assumed to cause an initiating event. The amount of water to cause a reactor trip for the at-power analysis is different from that needed to cause a submergence failure of the shutdown cooling pumps. The at-power analysis screened out pipe breaks that would only fail the SCPs, because these breaks would not cause a reactor trip. This resulted in the identification of additional pipe breaks and calculation of associated break frequencies specifically for LPSD conditions.

To ensure that the potential flood propagation paths for LPSD conditions have been properly evaluated, on February 23, 2016, the staff issued RAI 418-8348, Question 19-51, requesting the applicant to describe in the DCD how flood barriers assumed to be intact during at-power conditions that may not necessarily be intact during LPSD conditions are addressed in the LPSD analysis (ML16054A291). In the response to Question 19-51, dated April 12, 2016 (ML16103A557), the applicant provided that the propagation analysis developed for the at-power internal flooding analysis would be applicable to the LPSD flooding analysis based on the assumption that flood barriers separating the two divisions of the auxiliary building are maintained. The applicant stated that the outage activities are assumed to be conducted on a train basis (e.g., no maintenance is performed on any train B equipment while work on train A equipment is planned), and that this assumption applies to flood barriers separating the two divisions. The applicant established Item COL 19.1(11), which states:

> The COL applicant and/or holder is to develop outage management procedures that limit planned maintenance that can potentially impair one or both SC trains during the shutdown modes.

In addition, COL 19.1(17) states:

> The COL applicant or holder is to develop a configuration control program requiring that, during Modes 4, 5, and 6, the watertight flood doors and fire doors be maintained closed in at least one quadrant. Furthermore, the COL applicant and/or holder is to incorporate, as part of the aforementioned configuration control program, a provision that if the flood or fire doors to this designated quadrant must be opened for reasons other than normal ingress/egress, a flood or fire watch must be established for the affected doors.

The staff finds that these provisions acceptably address the PRA assumption, which credits the availability of flood barriers separating the two divisions during LPSD conditions. RAI 418-8348, Question 19-51, is resolved and closed.

LPSD Internal Flooding Risk Results and Insights

The results show that the estimated LPSD internal flooding mean CDF of 8.4E-8/yr is roughly an order of magnitude lower than that from LPSD internal fires. The applicant did not explicitly quantify the LPSD internal flooding LRF and estimated the upper bound LRF to be equal to the LPSD internal flooding CDF. The staff finds this approach acceptable since it conservatively estimates the LPSD internal flooding LRF by assuming that all core damage sequences lead to large release.

The results of the LPSD internal flooding risk evaluation identified the following important LPSD internal flooding sequences.

- A major fire protection system break in Room 78-A44B of the auxiliary building which disables both SC pumps contributes about 23 percent to LPSD internal flooding CDF
- A flood from a large fire protection system break in Room 78-A19B which spreads to Quadrant D failing B and D 1E 4kV power and fails the UATs contributes about 19 percent to LPSD internal flooding CDF
- An unisolable break of an IRWST piping in Room 55-A22A, failing SC Pump A and draining the IRWST into Quadrant A of the auxiliary building contributes about 16 percent to LPSD internal flooding CDF
- A major fire protection system flood originating in Room 78-A15D, which spreads to Quadrant B failing B and D 1E 4kV power and fails the UAT contributes about 8 percent to the LPSD flooding CDF.

The applicant conservatively did not credit the containment spray pumps, which could be used as a backup if the flood did not fail them. In terms of plant operating state, POS5 (reduced inventory operation) and POS10 (RCS draindown after refueling), respectively, contribute about 16 percent and 76 percent to the LPSD internal flooding CDF.

In general, operator actions are significant contributors to LPSD flooding risk, since there are no automatic actions for mitigation functions. In particular, operator action to isolate the fire protection line break in less than 20 minutes, and operator actions to restore shutdown cooling and to provide inventory makeup are risk-significant.

The results of the LPSD internal flooding risk evaluation reflect the APR1400 design features that minimize the flooding hazard propagating from one division to the other division. Section 19.1.5.3 of this SER summarizes the built-in flooding protection design features for the APR1400. However, to ensure the validity of the LPSD internal flooding risk profile, the integrity of flood barriers must be maintained. The applicant developed COL information items to address this issue, which the staff evaluated and finds acceptable.

Based on the foregoing discussions, staff finds that the APR1400 LPSD internal flooding risk compares favorably with the Commission's goals, as described in SRP Section 19.0, and that sufficient risk insights are identified to support the DC application.

0.1.7 PRA-Related Input to Other Programs and Processes

0.1.7.1 PRA Input to Design Programs and Processes

Summary of Application

The application states that the APR1400 PRA is an integral part of the design process and has been used to optimize the plant design with respect to safety. The PRA models and results have influenced the selection of design features.

Technical Evaluation

The staff's evaluation of PRA input to the design process is documented in Section 19.1.1.1 of this SER.

~~19.1.7.2~~0.1.7.2 PRA Input to the Maintenance Rule Implementation

Summary of Application

At the DC stage, the PRA is not used to support maintenance rule implementation. Each COL applicant referencing the DCD will describe how it will use PRA to support implementation of the maintenance rule.

Technical Evaluation

The APR1400 DCD established COL information item is needed to enable the staff to assess the use of the PRA to support development and implementation of the maintenance rule program prior to and during the operational phase. Item COL 19.1(18) states:

> The COL applicant is to describe the uses of PRA in support of licensee programs such as Maintenance Rule implementation during the operational phase.

The above COL information item is in conformance with the guidance provided in the SRP and RG 1.206 and therefore, the staff finds it is acceptable.

19.1.7.3 PRA Input to the Reactor Oversight Process

Summary of Application

At the DC stage, the PRA is not used to support the reactor oversight process. Each COL applicant referencing the DCD will describe how it will use PRA to support implementation of the reactor oversight process during the operational phase.

Technical Evaluation

The APR1400 DCD established COL information item is needed to enable the staff to assess use of the PRA to support development and implementation of the reactor oversight process during the operational phase. COL 19.1(19) states:

> The COL applicant is to describe the uses of PRA in support of licensee programs such as the reactor oversight process during the operational phase.

The above COL information item is in conformance with the guidance provided in the SRP and RG 1.206 and therefore, the staff finds it is acceptable.

19.1.7.4 **PRA Input to the Reliability Assurance Program**

Summary of Application

The application states that the PRA is used to identify SSCs that are potentially risk significant, using probabilistic importance measures. These are considered for inclusion within the scope of the reliability assurance program (RAP).

Technical Evaluation

The staff's evaluation of APR1400 RAP is documented in SER Section 17.4.

19.1.7.5 PRA Input to the Regulatory Treatment of Non-Safety-Related Systems Program

Summary of Application

The application states that the APR1400 is an evolutionary design but not a "passive" design. The program for regulatory treatment of non-safety-related systems (RTNSS) is not applicable to the APR1400.

Technical Evaluation

The staff confirmed that the APR1400 is an evolutionary design primarily based on existing LWR technology with no passive backup systems. Therefore, consistent with SRP Section 19.0, the staff concludes that the RTNSS program is not applicable to the APR1400 design.

19.1.7.6 AFWS Reliability Analysis

Summary of Application

The auxiliary feedwater system (AFWS) reliability analysis was performed in accordance with TMI Action Item II.E.1.1 of NUREG-0737, “Clarification of TMI Action Plan Requirements.”

Technical Evaluation

The staff’s evaluation of the AFWS reliability is documented in Section 10.4.9 of this SER, “Auxiliary Feedwater System.”

19.1.7.7 **Combined License Information Items**

The APR1400 DCD is intended to be referenced by an applicant for a combined license. Each COL applicant must supply site-specific information and additional analyses to enable the staff to confirm that the design interface requirements of the APR1400 certified design are bounded. This information is documented in the APR1400 DCD as COL information. Alternatively, a COL applicant may propose and justify departures from the certified design. These must be reflected in the plant-specific PRA based on the design-specific PRA.

The applicant established several COL information items relevant to the APR1400 PRA and tabulated them in DCD, Revision 0, Section 19.1.9. During the course of the review of this chapter, the applicant identified and added more COL items to the COL information list. The bases for these items are discussed in the previous relevant sections of this report. Below is the comprehensive list of APR1400 PRA-related COL information items.

The COL applicants and licensees referencing the certified APR1400 standard design must address and satisfy the commitments identified in the following COL information items. These items do not establish requirements; rather, they identify an acceptable set of information to be included in a plant-specific safety analysis report. An applicant or licensee may deviate from these information items. However, any deviation or omission from these items needs to be clearly identified and justified in the plant-specific safety analysis report. As set forth above, the staff has evaluated all of the COL information items that are applicable to the APR1400 PRA. The staff finds that the applicant has sufficiently established COL information and identified necessary actions and/or analyses to accomplish these items. The staff concludes that the established COL information items associated with APR1400 PRA, as listed below, are reasonable, sufficient, and in conformance with the SRP Section 19.0 guidance. Therefore, the staff concludes they are all acceptable.

Table 19.1.7-1 Combined License Items Identified in the DCD

Item No.	Description	Section
COL 19.1(1)	The COL applicant is to describe the uses of PRA in support of licensee programs, and to identify and describe risk-informed	19.1.1

Item No.	Description	Section
	applications being implemented during the combined license application phase.	
COL 19.1(2)	The COL applicant is to describe the uses of PRA in support of licensee programs, and identify and describe risk-informed applications being implemented during the construction phase.	19.1.2
COL 19.1(3)	The COL applicant is to describe the uses of PRA in support of licensee programs, and identify and describe risk-informed applications being implemented during the operational phase.	19.1.3
COL 19.1(4)	The COL holder is to review as-designed and as-built information and conduct walkdowns as necessary to confirm that the assumptions used in the PRA (including PRA inputs to RAP and SAMDA) remain valid with respect to internal events, internal flood and fire events (fire barrier and fire barrier penetrations, routing and locations of pipe, cable, and conduit), and HRA analyses (development of operating procedures, emergency operating procedures, and severe accident management guidelines and training), external events including PRA-based seismic margins, HCLPF fragilities, seismic spatial interactions and LPSD procedures.	19.1.4
COL 19.1(5)	The COL applicant and/or holder is to conduct a peer review of the PRA relative to the industry PRA Standard prior to use of the PRA to support risk-informed applications, as applicable.	19.1.5
COL 19.1(6)	The COL applicant is to describe the PRA maintenance and upgrade program.	19.1.6
COL 19.1(7)	The COL applicant and/or holder is to develop management procedures for charging pump operation, following recovery from a loss of offsite power (LOOP), to ensure that deboration is not resumed until after at least one Reactor Coolant Pump (RCP) has been restarted.	19.1.7
COL 19.1(8)	The COL applicant will confirm and update from new information from the site, e.g., site features, design departures, etc., that the PRA-based SMA is bounding for the selected site, site-specific SSC and soil effects, including sliding, overturning, liquefaction, and slope failure. The COL applicant is to confirm that the as-built plant has adequate seismic margin and does not exceed the CDF and LRF design targets specified in Subsection 1.2.1.1.1e.	19.1.8

Item No.	Description	Section
	The COL applicant is to demonstrate that site-specific structures (the turbine building, compound building, ESW IS and CCW HX buildings) have a HCLPF capacity that is equal to or greater than 1.67 times GMRS and will update the PRA-based SMA with the site-specific structure HCLPF values, accordingly. The COL applicant is to demonstrate that HCLPF capacity is equal to or exceed 1.67 times the CSDRS for BOP components and is to complete the SEL. The COL applicant is to demonstrate that the seismic capacity for equipment and relay qualified by testing should remain functionally operational within 1.67 times the required response spectra (CSDRS-based RRS) in the procurement specification. The COL applicant is to demonstrate that the inherently rugged components identified in DCD Section 19.1.5.1.1.2 have seismically rugged capacity. The COL applicant is to demonstrate that the steam generator tube HCLPF is higher than the HCLPF for the steam generator nozzle.	
COL 19.1(9)	When developing post-earthquake safe shutdown procedures, the COL applicant and/or holder should consider the potential for multiple spurious alarms from photoelectric detectors following a seismic event.	9.1.9

Item No.	Description	Section
COL 19.1(10)	The COL applicant and/or holder needs to ensure that screened events do not have a site-specific susceptibility and do not exceed the CDF and LRF design targets specified in Subsection 1.2.1.1.1, Item e. The COL applicant and/or holder is to address the following issues with a site-specific risk assessment, as applicable: • Tsunami • Aircraft crash event • External flooding • Extreme winds and tornadoes • Industrial or military facility • Lightning • Pipeline accident • Release of chemicals from onsite storage • River diversion/River flooding • Storm surge • Toxic gas • Transportation accidents. In addition, the COL applicant and/or holder is to ensure the site-specific susceptibility is not an outlier for the following issues, as applicable: • Avalanche • Biological events • Coastal erosion • Drought • Forest fire • High summer temperature • Hurricane • Landslide • Low lake/river water level • Low winter temperature • Sandstorm • Tsunami Volcanic activity.	19.1.10
COL 19.1(11)	The COL applicant and/or holder is to develop outage management procedures that limit planned maintenance that can potentially impair one or both SC trains during the shutdown modes.	19.1.11
COL 19.1(12)	The COL applicant and/or holder is to develop procedures and a configuration management strategy to address the period of time	19.1.12

Item No.	Description	Section
	when one SC train is unexpectedly unavailable (including the termination of any testing or maintenance that can affect the remaining train and restoration of all equipment to its nominal availability). The COL applicant is to ensure operation of the emergency diesel generator sequencer throughout low power and shutdown operations (not including defueled plant operating states).	
COL 19.1(13)	The COL applicant and/or holder is to establish procedures for closing the containment hatch (after being opened during LPSD operations) to promptly re-establish the containment as a barrier to fission product release. This guidance must include steps that allow for sealing of the hatch with four bolts (versus the 40 bolts used to secure the hatch during at-power operation); four bolts are sufficient to secure the hatch so that no visible gap can be seen between the seals and the sealing surface.	19.1.13
COL 19.1(14)	The COL applicant and/or holder is to develop procedures specifying that a fire watch be present when hot work is being performed.	19.1.14
COL 19.1(15)	The COL applicant and/or holder is to develop the fire barrier management procedures that direct the appropriate use of a fire watch and use of the isolation devices with a quick-disconnect mechanism for hose and cables that breach a fire barrier.	19.1.15
COL 19.1(16)	The COL applicant and/or holder is to develop procedures and operator training for reliance (during fire response) on undamaged instrumentation (when the location of the fire is known). The COL applicant is to ensure Alarm Response Procedures related to Loss of all RCP seal cooling including, but not limited to, partial and total loss of the component cooling water system, and partial and total loss of the essential service water system, include steps to trip the reactor followed by immediate trip of the RCPs if RCP seal cooling cannot be re-established within the time limit specified by the manufacturer.	19.1.16
COL 19.1(17)	The COL applicant and/or holder is to develop a configuration control program requiring that, during Modes 4, 5, and 6, the watertight flood doors and fire doors be maintained closed in at least one quadrant. Furthermore, the COL applicant and/or holder is to incorporate, as part of the aforementioned configuration control program, a provision that if the flood or fire doors to this designated quadrant must be opened for reasons other than normal	19.1.17

Item No.	Description	Section
	ingress/egress, a flood or fire watch must be established for the affected doors.	
COL 19.1(18)	The COL applicant is to describe the uses of PRA in support of licensee programs such as Maintenance Rule implementation during the operational phase.	19.1.18
COL 19.1(19)	The COL applicant is to describe the uses of PRA in support of licensee programs such as the reactor oversight process during the operational phase.	19.1.19
COL 19.1(20)	The COL holder is to perform the seismic-fire interactions walkdown to confirm a qualitative seismic-fire interaction assessment.	19.1.20
COL 19.1(21)	The COL applicant and/or holder is to develop outage procedures to ensure that in fire compartments containing post-seismic or post-fire safe shutdown equipment that: (1) the seismic ruggedness of temporary ignition sources is adequate, or that the duration that these temporary ignition sources are in these areas is minimized, (2) the seismic ruggedness of temporary equipment such as scaffolding in fire compartments containing potential seismic-fire ignition sources, or near fire protection equipment is adequate, and (3) either the duration of activities which could impact manual firefighting is minimized, or alternative firefighting equipment (e.g., pre-stage portable smoke removal equipment, prestage additional firefighting equipment, etc.) is supplied.	19.1.21
COL 19.1(22)	The COL applicant and/or holder is to ensure that asymmetric conditions due to modeling simplicity will be addressed or properly accounted for when the PRA is used for decision making.	19.1.22
COL 19.1(23)	The COL holder will demonstrate that maintenance-induced floods are negligible contributors to flood risk when the plant specific data are available.	19.1.23
COL 19.1(24)	SAMGs are entered to initiate SI with the core exit thermocouple indicating 1200^{o}F. For COL 19.1(24), the staff interpreted this as “the COL applicant is to ensure that the APR1400 plant will enter SAMGs when the core exit thermocouples reach 1200F.”	19.1.24
COL 19.1(25)	The COL applicant and/or holder ensures that the fire protection features required for preventing fire-induced damage of the	19.1.25

Item No.	Description	Section
	PRA-credited components will be properly incorporated in the cable design.	
COL 19.1(26)	The COL applicant will ensure the APR1400 thermal-hydraulic (T/H) analysis supporting the application is reflected in the updated PRA model including those design and operational features for mitigation of a hot leg LLOCA.	19.1.26
COL 19.1(27)	The COL applicant will review the Technical Specifications and incorporate logic into the PRA model to ensure cutsets reflect permissible maintenance configurations.	19.1.27

19.1.8 Conclusion and Finding

The staff completed its review of APR1400 design-specific PRA and other PRA-related information provided in DCD Tier 2, Sections 19.0 and 19.1 in accordance with SRP Section 19.0 guidance.

The APR1400 PRA addressed internal events, internal fires, and internal floods initiated from both full-power and LPSD conditions. Table 19.1.10-1 below summarizes the mean CDFs and LRFs reported by the applicant. The applicant employed the PRA-based SMA to evaluate potential vulnerabilities to seismic events in conformance with the SRM on SECY-93-087. The applicant analyzed the challenges to safe operation from other external hazards using the screening approach provided in ASME/ANS PRA Standard as endorsed by RG 1.200.

Table 19.1.10-1. APR1400 Mean CDF and LRF Estimates

	CDF (per year)		**LRF (per year)**	
	At-power	**LPSD**	**At-power**	**LPSD**
Internal Events	1.3E-06	1.9E-06	1.2E-07	7.2E-08
Internal Fires	2.3E-06	1.2E-06	1.9E-07	6.7E-08
Internal Floods	4.4E-07	8.4E-08	2.9E-08	8.4E-08
Total	4.0E-06	3.2E-06	3.4E-07	2.2E-07

Based on the staff's review as discussed in previous sections of this SER and staff's assessment of the mean CDFs and LRFs reported by the applicant as shown in Table 19.1.10-1 above, the staff concludes that the APR1400 risk assessments are consistent with the Commission's CDF and LRF goals of less than 1E-04/yr and 1E-06/yr, respectively, and that sufficient risk insights are identified to support the DC application. The APR1400 PRA is of the appropriate scope, level of detail, and technical adequacy for its identified uses and applications. The APR1400 PRA reasonably reflects the as-designed, as-to-be-built, and as-to-be-operated plant to the extent possible.

The staff confirmed that as compared to currently operating plants, APR1400 is designed with improved levels of redundancy and separation of safety divisions to provide inherent protection against internal and external hazards. The staff finds that the applicant has:

- properly demonstrated that the APR1400 plant-level seismic capacity is sufficient for the seismic hazard represented by the CSDRS.
- reasonably revealed the seismic risk insights to illustrate the robustness and potential weakness of APR1400 standard design for use by COL applicants to optimize plant construction and operation.
- logically addressed the potential risks from other external events by analyzing or performing screening analysis, and established COL information item to ensure the impacts from external hazards are to be fully investigated.
- successfully met the Commission's containment performance goals (e.g., containment integrity be maintained for approximately 24 hours following onset of core damage and CCFP less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA), based on the PRA results.
- practically performed uncertainty analysis and conducted sensitivity studies and importance analysis to identify important equipment and operator actions, as well as risk-related insights for use to prevent and mitigate severe accidents and reduce the risk of these accidents.
- effectively utilized the PRA to assess risk and used PRA as an input to augment final design, TS, ITAAC, SA evaluation, RAP, human factors engineering, physical security, and environmental review.
- fully addressed the Commission's objectives regarding the preparation and use of its PRA in the design phase and certification process. This is consistent with the SRM on SECY-89-102 and the Commission policy statement "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," 60 FR 42622, August 16, 1995.

In conclusion, during the APR1400 PRA safety evaluation, the staff had identified a number of issues that the applicant did not properly address. The applicant has now resolved all of these findings by satisfactorily responding to the staff's RAIs and audit questions, and updating the DCD. The staff finds that the applicant has adequately addressed the Commission's objectives regarding the preparation and use of a PRA. The applicant has satisfied the PRA-related goals and identified the important aspects of the APR1400 design and operation so that particular attention can be placed on these aspects during certification, construction, and operation. The staff concludes that the information provided in APR1400 DCD Sections 19.0 and 19.1 is complete and acceptable.

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19.2. Severe Accident Evaluation

19.2.1 Introduction

This section evaluates the APR1400 features that are designed to prevent and mitigate severe accidents in compliance with the requirements in 10 CFR 52.47(a)(8) and (23). The staff used the guidance provided for specific issues related to accident prevention and mitigation, as identified in Commission papers SECY-90-016 and SECY-93-087, which the Commission approved in related Commission staff requirements memorandum (SRMs) dated June 26, 1990, and July 21, 1993, respectively. Features are included to prevent accidents such as anticipated transient without scram (ATWS), mid-loop operation, station blackout (SBO), fire, and ISLOCA. Mitigation features include hydrogen generation and control, core debris coolability, high pressure core melt ejection, containment performance, dedicated containment vent penetration, and equipment survivability.

19.2.2 Severe Accident Prevention

The application describes features aimed at preventing the onset of a severe accident, including the severe accident precursors identified in SECY-90-016 and SECY-93-087. These precursors include ATWS, mid-loop operation, SBO, fire, and ISLOCA.

*19.2.2.1 **Anticipated Transient without Scram***

Summary of Application

The APR1400 design includes a digital safety system and a diverse protection system (DPS) to minimize the possibility of an ATWS, i.e., an anticipated operational occurrence that is not terminated by automatic or manual reactor shutdown. The DPS design includes reactor trip, turbine trip, auxiliary feedwater actuation, and safety injection actuation functions. The DPS reactor trip provides a simple and diverse mechanism to decrease the risk from ATWS events and mitigates the effects of a postulated software common cause failure (CCF) of the digital computer logic within the digital reactor protection system (RPS) and/or engineered safety feature – component control system (ESF-CCS). The DPS turbine trip is automatically initiated whenever the DPS reactor trip conditions are met (with 3 second time delay). The DPS auxiliary feedwater system actuation provides additional reasonable assurance that an ATWS event could be mitigated if it occurred. The DPS safety injection system actuation assists the mitigation of the effects of a large break LOCA event with a concurrent software CCF within the plant protection system and ESF-CCS.

The reactor trip system is normally available to prevent an ATWS by shutting the reactor down automatically or manually following an anticipated transient. The reactor trip system includes the RPS which generates a trip signal in response to a reactor upset condition, reactor trip switchgear system, and components that perform a reactor trip after receiving a signal from the RPS (either automatically or manually). The DPS initiates a reactor trip signal on high pressurizer pressure to decrease the possibility of an ATWS and provides an auxiliary feedwater actuation signal (backup to the ESF-CCS of the plant protection system) to mitigate an ATWS.

Technical Evaluation

In SECY-90-016, the NRC concluded that evolutionary light water reactor (LWR) designs should provide diverse scram systems to mitigate a potential ATWS and to ensure a safe reactor shutdown. The staff finds that the APR1400 meets the criteria specified in SECY-90-016 through incorporation of digital safety system and a DPS as discussed above.

19.2.2.2 Mid Loop Operation

Summary of Application

DCD Section 19.2.2.2, "Mid-Loop Operation," describes the APR1400 design features that can prevent and mitigate loss of residual heat removal during reduced reactor water inventory operations. These design features include:

a. Instrumentation for shutdown operations

b. Shutdown cooling system design

c. SG nozzle dam integrity

d. Alternate decay heat removal methods.

These issues are also discussed in DCD Section 5.4.7, "Shutdown Cooling System" and DCD Section 19.1.6.1, "Description of Low-Power and Shutdown Operations PRA."

Technical Evaluation

Regarding instrumentation for midloop operations, the staff audited the details provided in the Shutdown Evaluation Report, APR1400-E-N-NR-14005, which has not been incorporated by reference. The staff found that these details were not referenced in DCD Chapter 5 or Chapter 7. This issue is further discussed in Section 19.1.6.1 of this SER under RAI 268-8308, Question 19-12 (ML15295A262). The staff confirmed that DCD Revision 1 contains the changes committed to in the RAI response; therefore, RAI 268-8308, Question 19-12, is resolved and closed.

Regarding improvements in the SCS design, the applicant states that the shutdown cooling suction lines do not contain loop seals, thereby minimizing the potential to trap gas. The suction piping layout allows self-venting of accumulated gas (or air). The potential for gas accumulation in the SCS and the potential for vortexing is further evaluated in Section 19.1.6.1 of this SER under RAI 42-7945, Question 19-2 (ML16203A282). The staff confirmed that the applicant added the Tier 1 Table 2.4.1-4 Design Commitment 15 to DCD Revision 1 that specifies "the decay heat removal function of the SCS will not be impaired by gas entrainment during mid-loop operation while the system is operating at its maximum allowable flow rate and the reactor coolant hot leg level is at the lowest level allowable for decay heat removal"; therefore, RAI 42-7945, Question 19-2, is resolved and closed.

The applicant stated that there are no auto-closure interlocks on the shutdown cooling suction piping valves, minimizing the potential for shutdown cooling isolation events, which is acceptable to the staff.

Regarding the SG nozzle dam integrity, the applicant presented procedural guidance for the nozzle dam installation and removal sequence. Regarding the need for adequate venting during the use of SG nozzle dams, the applicant required that the pressurizer manway be opened so that a hot side vent pathway exists prior to blocking both RCS hot legs with nozzle dams. Procedural guidance for nozzle dam installation and removal is addressed in Section 19.1.6.1 of this SER, specifically, in RAI 232-7864, Question 19-6 discussion (ML16088A357). The staff confirmed that DCD Revision 1 contains the changes committed to in the RAI response; therefore, RAI 232-7864, Question 19-6, is resolved and closed.

Regarding alternate inventory additions and decay heat removal methods, the applicant credited the containment spray (CS) pumps and the SI pumps. If these pumps are unavailable for decay heat removal and inventory control, the applicant stated that a charging pump or a boric acid makeup pump could be used to provide makeup for Modes 5 and 6. If pumped inventory addition is also unavailable, the applicant stated that gravity feed can be performed from the safety injection tanks (SITs).

Regarding alternate inventory addition methods, on February 22, 2016, the staff issued RAI 409-8325, Question 19-21 (ML16053A015), requesting the applicant to address the following information:

1) The applicant to justify how the SITs can keep the core covered assuming the RCS is vented via the pressurizer given possible pressurizer surge line flooding. Surge line flooding following an extended loss of DHR may negate the elevation head necessary for SIT flow. Based on the shutdown evaluation report, the staff understands "[w]ith the earliest nozzle dam installation occurring at 4 days after shutdown, the decay heat present would require approximately 481 L/min (127 gpm)."

2) The applicant to clarify whether a charging pump and a boric acid pump are needed to keep the core covered or if either a single charging pump or a single boric acid pump is sufficient to keep the core covered.

In the response to RAI 409-8325, Question 19-21, dated April 12, 2016 (ML16103A538), the applicant updated DCD Section 19.2.2.2 to state that the flow rate of the charging pump is 150 gpm (567.75 L/min). DCD Section 19.2.2.2 was also updated to state that the minimum makeup flow of 127 gpm (481 L/min) is required to keep the core covered for the loss of DHR during mid-loop operation. Since this proposed DCD update explains that one charging pump can provide sufficient makeup given RCS boil-off, this DCD update is acceptable to the staff. The staff confirmed that Revision 1 of the DCD contains the changes committed to in the RAI response; therefore, RAI 409-8325, Question 19-21, is resolved and closed.

Regarding the use of the SITs to keep the core covered and mitigate loss of residual heat removal during reduced reactor water inventory operations, the applicant provided their final response to RAI 409-8325, Question 19-21 (ML16302A152), on October 28, 2016. Based on calculations reported in the Fukushima Technical Report, APR 1400-E-P-NR-14005 (ML15128A282), the core can be covered by the SITs assuming the RCS is vented via the pressurizer for non-ELAP conditions (extended loss of ac power) if the SITs are maintained between 65 and 75 psia. Revision 1 of DCD Section 19.2.2.2 was revised to state that the SITS can keep the core covered assuming the RCS is vented via the pressurizer for non ELAP conditions, if the SITS are maintained between 65 psia and 75 psia. This DCD update on the need to keep the SITS pressurized between 65 psia and 75 psia resolves the staff's concerns. Accordingly, the staff concludes that RAI 409-8325, Question 19-21, is resolved and closed.

19.2.2.3 Station Blackout

Summary of Application

The APR1400 design provides one alternate alternating current (AAC) source to help mitigate the effects of a station blackout accident. The applicant asserts that the AAC will start and be manually aligned to provide power to a Class 1E 4.16 kV bus in case Class 1E emergency diesel generators (EDG)s fail to start and load during loss of offsite power (LOOP) events.

The applicant makes the following additional key assertions:

- This AAC source is independent and diverse from the Class 1E EDGs
- Successful startup of the AAC together with turbine-driven auxiliary feedwater pumps is sufficient to prevent core damage during SBOs.

The applicant states that major improved design features related to mitigation of an SBO event that were adopted in the APR1400 to reduce or eliminate weaknesses in previous plant designs are (a) design change from two EDGs to four EDGs and (b) extension of 125 Vdc battery life to 16 hours from 8 hours. DCD Section 8.4 documents APR1400 conformance to the requirements of 10 CFR 50.63.

Technical Evaluation

An SBO involves the complete loss of ac electrical power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the on-site emergency ac power system). SBO does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources, nor does it assume a concurrent single failure or design basis accident (DBA).

APR1400 added several design alternatives as stated above to ensure that a plant is able to withstand an SBO for a specified duration and recover. Section 8.4 of this SER documents the acceptability of the APR1400 design relative to the SBO rule.

19.2.2.4 Fire Protection

Summary of Application

The APR1400 design provides fire protection features such as fire detection, automatic and manual fire suppression, and fixed fire barriers to prevent the plant from entering an unrecoverable state resulting from a fire.

Technical Evaluation

Though the fire protection system (FPS) is not defined as a safety-related system, the FPS serves as a preventive feature for severe accidents by reducing or eliminating the possibility of fire events that could induce transients, damage mitigation equipment, and hamper operator responses. SECY-90-016 specifies design criteria for evolutionary advance light water reactors that safe shutdown be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible.

The APR1400 FPS features as described in DCD Section 9.5.1 and Appendix 9.5A are capable of providing assurance that, in the event of fire, the plant will not be subjected to an unrecoverable incident. Section 19.1.5.2 of this report describes the staff review of the applicant's internal fire risk evaluation.

19.2.2.5 Interfacing Systems Loss of Coolant Accident

Summary of Application

In the APR1400 design, the safety injection system, SCS, and chemical volume control system (CVCS) are directly connected to the RCS and are potentially susceptible to one or more ISLOCA (i.e., they have one or more ISLOCA pressurization pathways). The APR1400 design addresses ISLOCA challenges with the following design features:

- Increased design pressure of equipment or systems to 64.3 kg/cm2 ([6.31E6 Pa], 900 psig) for the low-pressure systems connected with the RCS
- Equipment and instrumentation to alert the operator to an ISLOCA challenge, or terminate and limit the scope of an ISLOCA event
- The refueling water tank is located inside containment
- Capability for leak testing pressure isolation valves
- Providing pressure isolation valve position indication and control in the main control room (MCR)
- High-pressure alarms to warn the operator when increasing pressure approaches the design pressure of low-pressure systems.

The DCD Section 19.2.2.5 makes the following key assertion: "…all sections of the systems connected to the RCS and interfaces are designed to withstand full RCS operating pressure, or they have leak-test capabilities, valve position indications in the control room that function even when isolation valve operators are de-energized, and high-pressure alarms to warn operators when pressure is approaching the design pressure."

Technical Evaluation

ISLOCAs are a class of LOCAs in which the RCS pressure boundary is breached and coolant is lost through an interfacing system with a lower design pressure. The breach may occur in portions of piping located outside the primary containment, causing a direct and potentially unisolable discharge from the RCS to the environment. An ISLOCA is of concern because of potential direct releases to the environment, loss of core cooling, and loss of core makeup.

High- or low-pressure interfaces occur on many lines including low-pressure injection lines and the RHR heat exchangers. An ISLOCA occurs when high pressure occurs in a low-pressure system because of valve failure or an inadvertent valve actuation. In either case, the over pressurization can cause the low-pressure system or components to fail. An ISLOCA concurrent with a loss of all core cooling would lead to core damage.

For reducing the possibility of an ISLOCA event, in SECY-90-016, the staff recommended for evolutionary LWRs to design low-pressure systems to withstand full RCS pressure or to provide means of testing pressure isolation valves and indications when pressure isolation valves are not closed. In SRM-SECY-90-016, the commission approved the staff recommendation.

The applicant identified that in the APR1400 design, the SIS, SCS, and CVCS are directly connected to the RCS and are potentially susceptible to one or more ISLOCA events (i.e., one or more ISLOCA pressurization pathways). The applicant stated that both the safety injection system and the SCS designs satisfy the ISLOCA acceptance criteria because all sections of systems and interfaces are designed to withstand full RCS operating pressure or have a leak-test capability. In addition, the valve position indications in the control room function even when the isolation valve operators are de-energized, and high-pressure alarms sound to warn operators when pressure is approaching the design pressure. The CVCS letdown and charging lines are directly connected to the RCS and are primary interfaces through which an ISLOCA event can occur. Each line has a high-pressure alarm to warn the operator when the pressure is approaching the low-pressure system design pressure. Following a warning, the control room operator isolates the line to terminate any further pressure communication downstream of the containment isolation valve.

DCD Section 19.2.2.5 states that "[t]he SCS design satisfies the ISLOCA acceptance criteria because all sections of the system and interfaces are designed to withstand full RCS operating pressure or have a leak-test capability." This section also states that "[d]eletion of the interfaces from the SCS lines eliminates the potential for an ISLOCA without adversely affecting the performance or operations of the SCS."

There appears to be a discrepancy between the first and second sentences as the former implies the presence of interfaces between SCS and RCS while the latter states deletion of interfaces. Therefore, during the public meeting with KHNP from April 13 through 15, 2015 (ML18123A181), the staff requested the applicant to clarify the discrepancy. On July 2, 2015, the applicant responded and agreed to update DCD Section 19.2.2.5 for additional clarification as follows (ML15183A067): “The SCS design satisfies the ISLOCA acceptance criteria because all sections of the system and interfaces are designed to withstand full RCS operating pressure.”

The proposed DCD update, which was incorporated in Revision 1 of the DCD, addresses the staff’s concern by clarifying the text in the DCD. Given the design features described above, which are consistent with commission direction in SRM-SECY-90-016, ISLOCA is not a significant contributor to CDF or LRF in the APR1400 design.

19.2.2.6 Other Severe Accident Preventive Features

Summary of Application

Two independent turbine driven auxiliary feedwater pumps are included in the APR1400 design for removing decay heat when both offsite and onsite ac power are not available.

A reactor feed-and-bleed can remove decay heat when secondary cooling via the steam generators is not available. Feed-and-bleed operation uses the SI pumps to transfer water from the in-containment refueling water storage tank (IRWST) to the reactor vessel and the pilot-operated safety relief valves (POSRV) to vent steam from the reactor vessel to the IRWST. Under these conditions, the containment spray pumps and containment spray heat exchangers can be used as backups for the shutdown cooling (SC) pumps and SC heat exchangers to provide IRWST cooling.

If the containment spray pumps are inoperable during a LOCA event, then the SC pumps can be used as a backup.

Technical Evaluation

In current plants, auxiliary feedwater is used to remove heat from the RCS via the steam generators. If AFW fails, the operators can initiate feed-and-bleed cooling of the primary system via safety injection and the remote manual opening of one or more power-operated relief valves. Both AFW and feed-and-bleed failures have been significant contributors to overall plant risk as indicated in the existing PRAs. The staff finds that the improvements summarized above enhance both secondary heat removal reliability and feed-and-bleed cooling. The staff concludes that APR1400 severe accident preventive features meet 10 CFR 52.47(a)(23) and are consistent with SRP 19.0.

19.2.3 Severe Accident Mitigation

The application states that if a core damage cannot be prevented, other APR1400 features would mitigate the effects of a severe accident. Of particular importance are the containment design and the ability of mitigating equipment to survive severe accident conditions.

19.2.3.1 Overview of the Containment Design

Summary of Application

The APR1400 containment is a pre-stressed concrete structure composed of a right circular cylinder with a hemispherical dome and is founded on a safety-related common basemat. The internal structures of the containment, which are made of reinforced concrete, enclose the reactor vessel and other primary system components. The penetrations in the APR1400 containment include one equipment hatch; two personnel airlocks; containment piping penetration assemblies to provide for the passage of process, service, sampling, and instrumentation pipelines into the containment; electrical penetrations for power, control, and instrumentation; and a fuel transfer tube. A detailed description of the containment design is provided in DCD Chapter 6.

The application states that the APR1400 containment has been designed to meet the containment factored load category (FLC) requirement of ASME Section III, Division 2, Subarticle CC-3720. The applicant asserts that (1) in a severe accident, the containment maintains its role as a reliable, leak-tight barrier by providing reasonable assurance that the FLC requirements are met for a period of approximately 24 hours following the onset of core damage, and (2) following this 24-hour period, the containment continues to provide a barrier against the uncontrolled release of fission products.

Technical Evaluation

SRP Section 19.0 states that the use of applicant's PRA and severe accident evaluation includes the following:

> Demonstrate how the risk associated with the design compares against the Commission's goals[1] of less than $1x10^{-4}$/year for core damage frequency and less than $1x10^{-6}$/year for large release frequency.[2] In addition, compare the design against the Commission's approved use of a containment performance goal, which includes (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and (2) a probabilistic goal that the conditional containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA.

Consistent with staff guidance, the APR1400 consists of design features to mitigate severe accidents. As evaluated under "Surrogate Safety Goals" of this section, APR1400 design meets the Commission's containment performance goal of 0.1 CCFP. As evaluated under Section

[1] These are goals and not regulatory requirements, and applicants should not artificially (or intentionally) increase PRA results associated with one metric simply to meet the goal associated with another metric. Rather, the applicant should compare its plant-specific PRA results and insights against these goals and address how its plant features properly balance severe accident prevention and mitigation, consistent with Item D.
[2] The SRM dated June 26, 1990 in response to SECY-90-016 established the identified goals.

19.2.4 of this SER, the APR1400 design meets the Commision's deterministic containment performance goals.

In APR 1400 DCD Revision 0, the applicant referenced different containment pressure capacities without justification, i.e., 162.7 psig, 123.7 psia, and 158 psig in Sections 19.1, 19.2 and 19.3, respectively. Therefore, in RAI 409-8325, Question 19-26, the staff asked the applicant to justify these differences (ML16053A015). In the responses, dated April 16, 2016 (ML16107A067) and August 4, 2016 (ML16217A306), the applicant stated the following:

- 162.7 psig (177.4 psia) referenced in DCD Section 19.1, "PRA," is the median value of containment pressure evaluated by the PRA containment fragility analysis as described in DCD Section 19.1.4.2.1.2.2.
- 123.7 psia (109.0 psig) referenced in DCD Section 19.2, "Severe Accident Evaluation," is the containment pressure resulting from severe accidents as described in DCD Section 19.2.4.2.1, which is in conformance with RG 1.216 Position 2.
- 158 psig referenced in DCD Section 3.8.1.4.11, which is in conformance with RG 1.216 Position 1.

Also, the applicant proposed to add Table 19.1-28a with these descriptions to the DCD for additional clarification. The staff reviewed the applicant's response and finds that it addresses the staff's concern, and therefore, is acceptable. The staff confirmed that DCD Revision 1 contains Table 19.1-28a as proposed. RAI 409-8325, Question 19-26, is resolved and closed.

19.2.3.2 Severe Accident Progression

Summary of Application

The applicant describes the processes that may occur as a severe accident progresses. Information is presented in the context of how these phenomena affect containment performance. Because of the complex processes involved, the progression of core melt scenarios can vary. The application references previous assessments reported in NUREG/CR-5132, NUREG/CR-5597, and NUREG/CR-5564 that provide generic insights that are also applicable to the APR1400 design.

This section of the application summarizes key phenomena in the progression of severe accidents for the APR1400 design including in-vessel and ex-vessel phases of an accident. The phenomena and processes in the APR1400 that can occur during in-vessel melt progression include:

a. Core heat up resulting from loss of adequate cooling

b. Metal-water reaction and cladding oxidation

c. Eutectic interactions between core materials

d. Melting and relocation of cladding, structural materials, and fuel

e. Formation of blockages near the bottom of the core as relocating molten materials solidify (wet core scenario)

f. Formation of a melt pool, natural circulation heat transfer, crust formation, and crust failure (wet core scenario)

g. Drainage of molten materials to the vessel lower head region (dry core scenario)

h. Formation of a melt pool, natural circulation heat transfer, and crust formation in the lower plenum

i. Reactor vessel breach from a local failure or global creep-rupture.

Early containment failure mechanisms during the ex-vessel phase of an accident include high-pressure melt ejection (HPME) with direct containment heating (DCH) and ex-vessel steam explosions. Contact between molten core debris and concrete in the reactor cavity during the ex-vessel phase of an accident leads to molten core and concrete interaction (MCCI). This interaction decomposes the concrete and can challenge the containment by various mechanisms, including: pressurization from evolved steam and non-condensable gases, which can cause overpressure failure of containment; transport of high-temperature gases and aerosols into the containment, leading to high-temperatures and possibly failure at the containment seals and penetrations; containment basemat melt-through leading to radioisotope release from the containment; reactor support structure melt-through leading to the relocation of the reactor vessel and tearing of containment penetrations; and/or production of combustible gases such as hydrogen and carbon monoxide.

The APR1400 containment uses a containment hydrogen control system to consume hydrogen as it is produced during a severe accident, thereby decreasing the potential for hydrogen combustion events that would challenge containment integrity.

<u>Technical Evaluation</u>

Severe accident progression can be divided into two phases: (1) in-vessel stage and (2) ex- vessel stage. The in-vessel stage generally begins with insufficient decay heat removal and can lead to melt-through of the reactor vessel. The ex-vessel stage involves the release of the core debris from the reactor vessel into the containment resulting in the phenomena such as DCH, steam explosions, and MCCI.

In-vessel melt progression establishes the initial conditions for assessing the thermal and mechanical loads that may ultimately threaten the integrity of the containment. In-vessel melt progression begins with uncovering of the core and initial heat-up, and continues until either (1) the degraded core is stabilized and cooled within the reactor vessel, or (2) the reactor vessel is breached and molten core material is released into the containment. The phenomena and processes for APR1400 during in-vessel melt progression are similar to those for the existing fleet of PWR with a large and dry containment.

The initial response of the containment to ex-vessel severe accident progression is largely a function of the pressure of the RCS at reactor vessel failure and the existence of water within the reactor cavity. If not prevented through design features, risk consequences are usually dominated by early containment failure mechanisms that could result from energetic severe accident phenomena, such as HPME with DCH and ex-vessel steam explosions. The long-term response of the containment from ex-vessel severe accident progression is largely a function of the containment pressure and temperature resulting from MCCI, hydrogen burn, and the availability of containment heat removal mechanisms.

DCD Revision 0, Section 19.2.3.2.1 states that the phenomena and processes in the APR1400 that can occur during in-vessel melt progression include reactor vessel breach from a local failure or global creep-rupture. However, the DCD did not describe the process used to determine vessel failure, modes of vessel failure, and failure size. The staff needed this information to determine how the applicant evaluated the reactor vessel breach. Therefore, in RAI 432-8377, Question 19-68, the staff asked that the applicant update the DCD with this information (ML16068A097).

In the response to Question 19-68, dated May 14, 2016 (ML16135A003), the applicant proposed to update the DCD with details, including the following:

> Five separate vessel failure mechanisms are evaluated using the MAAP code: (1) local ablation of vessel wall by molten jet impingement, (2) melt ingress into a penetration tube and tube wall failure outside the vessel, (3) ejection of a penetration tube, (4) creep rupture of the lower head and (5) attack of the vessel wall by overlying metal layer. For a localized vessel failure for which the failure size is not clearly defined, such as for creep rupture of the lower head, jet ablation of the vessel wall, or overlying metal layer attacking the vessel wall, a radius of 0.01 m was used for the initial failure size. For heatup/failure or ejection of a penetration tube, the radius is used for the initial failure size. The initial failure size is not important because the failure opening size increases rapidly due to ablation by the debris passing through the breach. The failure location is determined by the code as a part of the failure calculation.

The staff finds that the applicant's evaluation of the reactor vessel breach is acceptable because (1) the applicant has used the MAAP code to evaluate possible vessel breach location and the failure opening size increases due to ablation by melt passing through the breach and (2) the staff's "MELCOR Confirmatory Analysis" described below in Section 19.2.3.3 found that the MAAP and MELCOR results are in general agreement. The staff confirmed that the applicant incorporated this change in DCD Tier 2 Revision 1. RAI 432-8377, Question 19-68, is resolved and closed.

DCD Section 19.2.3.2.1 states that the in-vessel core melt progression contains considerable uncertainty relating to the following:

a. Potential for in-vessel steam explosion

b. Interaction between core debris and internal vessel structures

c. Time and mode of vessel failure

d. Composition of the core debris released at vessel failure

e. Amount of in-vessel hydrogen generation

f. In-vessel fission-product release and transport

g. Retention of fission products and other core materials in the RCS.

However, the DCD Revision 0 does not describe how the applicant addressed the above uncertainties. In addition, DCD Section 19.2.3.2.2, "Ex-Vessel Melt Progression," does not list or describe uncertainty relating to the ex-vessel core melt progression. Therefore, in RAI 467-8394, Question 19-102, the staff asked the applicant to update the DCD with this information (ML16113A122). In the response to Question 19-102, dated July 5, 2016 (ML16187A157), the applicant referred to DCD Section 19.2.3.3.5.1.1, which states that the fuel-coolant interaction (FCI) expert review group sponsored by the NRC concluded in NUREG-1116 and NUREG-1524 that probability of in-vessel steam explosion (IVSE) failure was vanishingly small or physically unreasonable. The applicant has also addressed uncertainties in the interaction between core debris and internal vessel structures, vessel failure mode and timing, hydrogen generation, and fission product release and transport by using conservative assumptions. For example, for hydrogen generation, the applicant assumed (a) the cladding oxidation surface area was doubled to account for exposure on both sides after cladding rupture, (b) all core channel steam flows to be available above the location of blocked nodes for oxidation and heat transfer, and (c) the hydrogen generation was artificially extended until achieving the hydrogen equivalent of 100 percent active fuel-cladding oxidation.

The applicant's response to RAI 467-8394, Question 19-102, proposed to update DCD, including Section 19.2.3.2.2 as follows:

> For the ex-vessel steam explosion evaluation, uncertainties associated with the pressure load calculation were assessed (Subsection 19.2.3.3.5.2.2). For the DCH evaluation, the uncertainties in input parameters such as the mass of UO2 in the lower plenum at vessel breach, the fraction of Zirconium (Zr) oxidized and the containment failure pressure were considered by using the Latin Hypercube Sampling (LHS) technique. The sampled inputs and other conservatively estimated inputs such as the RCS pressure at vessel breach, the fraction of corium dispersed from the cavity and the fraction of dispersed corium entering the subcompartment were used to calculate the conditional containment failure probability.

To address the uncertainties related to other ex-vessel core melt progression phenomena, the applicant used two sets of input parameters to conservatively increase (a) ablation depth due to MCCI, and (b) hydrogen generation and containment pressurization. The applicant analyzed a spectrum of sequences, including dominant PRA sequences, and bounding deterministic sequences with the two sets of parameters and selected conservative results.

The applicant proposed to update DCD Sections 19.2.3.2.1 and 19.2.3.2.2 with this information. The staff review finds that the applicant has used conservative assumptions for addressing uncertainties related to in-vessel and ex-vessel phenomena. Therefore, the staff finds that the applicant's response acceptable. The staff confirmed that the applicant incorporated this change in DCD Tier 2 Revision 1. RAI 467-8394, Question 19-102, is resolved and closed.

19.2.3.3 Severe Accident Mitigation Features

The applicant has evaluated severe accident phenomena and processes in the APR1400 that can occur during in-vessel and ex-vessel melt progression. The applicant has designed features to mitigate severe accident phenomena and processes that can take place in the APR1400 containment.

Summary of Application

DCD Tier 2 Section 19.2.3.3 describes various severe accident scenarios and the features designed to mitigate these phenomena. They include: external reactor vessel cooling; hydrogen generation and control; MCCI and core debris coolability; high-pressure melt ejection and DCH; fuel-coolant interactions; and containment bypass.

Specific features have been included in the APR1400 design to mitigate the effects of particular severe accident phenomena. DCD Tier 2 Section 19.2.3.3 provides a description of each severe accident mitigation feature and the analysis of its performance in mitigating specific severe accident phenomena.

The specific severe accident phenomena and issues addressed in the APR1400 DCD are listed below in Table 19.2-1 along with the specific design features that are intended to mitigate the effects of the particular phenomena. Key results and conclusions from the analysis of the design features are also listed in the table.

Table 19.2 1. Summary of Severe Accident Phenomena

Severe Accident Phenomena	Severe Accident Mitigation Features	Analysis Results
External Reactor Vessel Cooling	Shutdown cooling pumps, boric acid makeup pump, dedicated piping and valves.	The APR1400 is designed to allow operators to fill the reactor cavity with water and thereby submerge the reactor vessel in coolant. This may provide sufficient ex-vessel cooling and in-vessel retention. However, in-vessel retention is not credited as a mitigation feature for the APR1400 due to several uncertainties.
Hydrogen Generation and Control	Hydrogen Mitigation System Rapid Depressurization Function Operation of three way valves	The APR1400 design with all of the severe accident mitigation features available is capable of maintaining a well-mixed containment atmosphere and a hydrogen concentration below 10%.

Severe Accident Phenomena	Severe Accident Mitigation Features	Analysis Results
MCCI and Core Debris Coolability	Cavity Flooding System Reactor Cavity Design	The corium in the APR1400 reactor cavity is quenched, and the integrity of containment liners is maintained when the CFS is available. An acceptable stable state can be achieved ex-vessel as long as the CFS has been actuated prior to vessel breach. Having a water-filled reactor cavity initially reduces and ultimately terminates erosion of concrete in the cavity. The cavity floor is free from obstructions and comprises an area available for core debris spreading such that the floor area/reactor thermal power ratio is larger than 0.02 $m^2/MW_{t.}$ Uniform distribution of 100% of the corium debris within the reactor cavity results in a relatively shallow debris bed and consequently, effective debris cooling is expected in the reactor cavity.
Direct containment heating (DCH) High Pressure Melt Ejection (HPME)	Reactor Cavity Design Rapid Depressurization Function	Corium retention in the core debris chamber virtually eliminates the potential for significant DCH-induced containment loadings. Operation of only two POSRVs within a half hour of the plant entering a severe accident is sufficient to decrease the RCS pressure below the DCH cutoff pressure for all sequences considered. The containment failure probability for the APR1400 due to DCH is estimated to be less than 0.01% (0.0001).
In-Vessel Steam Explosion (IVSE)	No mitigation features are provided to prevent or mitigate IVSE.	Because the APR1400 design is not significantly different from current PWRs, the NUREG-1524 conclusions are applicable to the APR1400 design (i.e., probability of containment failure due to IVSE is vanishingly small or physically unreasonable).

Severe Accident Phenomena	Severe Accident Mitigation Features	Analysis Results
Ex-Vessel Steam Explosion (EVSE)	The reactor cavity and RPV column support is designed such that the cavity strength has an adequate capability to withstand the postulated pressure load during a severe accident.	The evaluation of the cavity structural analysis indicates that the reactor cavity integrity is preserved during both static and dynamic EVSE loads.
Containment Bypass Thermally-induced steam generator tube rupture ISLOCA	Pilot operated safety relief valves (POSRV) Design features to address ISLOCA are summarized above in Section 19.2.2.5 of this safety evaluation report.	Manual actuation of POSRVs will reduce pressure in the reactor vessel below 17.6 kg/cm2 (250 psia). Once primary system pressure reaches this level, there is essentially no risk of an induced tube rupture occurring.
Equipment Survivability	Emergency containment spray backup system (ECSBS) The equipment and instrumentation needing to survive the harsh environment produced by a severe accident are summarized in Table 19.2.3-4 of the DCD.	Bounding Environmental Conditions: Temperature–short term:1,200 °K (1,700F) Temperature–long term: 460 °K (368F) Pressure 7.75 kg/cm2 (110 psia) Radiation Dose 4.4×10^{7} rad COL application will evaluate that the likelihood that the instrumentation and equipment required to mitigate a severe accident and achieve a safe stable state can perform their function as intended under severe accident environmental conditions (COL 19.2(1)).

The design features for mitigation of severe accidents are described in the application. The descriptions are summarized below.

Cavity Flooding System

The CFS provides a means of flooding the reactor cavity during a severe accident to cool the core debris in the reactor cavity and to scrub fission products. The CFS takes water from the IRWST and directs it to the reactor cavity. The water flows first into the HVT by way of the two HVT spillways and then into the reactor cavity by way of two reactor cavity spillways. Once actuated, movement of the water from the IRWST source to the cavity occurs passively due to the natural hydraulic driving heads of the system. Flooding of the reactor cavity serves the following purposes in the strategy to mitigate the consequences of a severe accident:

- Minimize or eliminate corium-concrete attack

- Minimize the generation of combustible gases (hydrogen and carbon monoxide) and noncondensable gases
- Scrub fission products released due to corium-concrete interaction
- Remove heat from the core debris.

Hydrogen Mitigation System

The containment hydrogen control system is designed to accommodate the hydrogen generation from a metal-water reaction of 100 percent of the active fuel cladding and limit the average hydrogen concentration in containment to 10 percent consistent with 10 CFR 50.34(f) and 10 CFR 50.44(c) for a degraded core accident. The hydrogen control system consists of a system of passive autocatalytic recombiners (PAR) complemented by glow plug hydrogen igniters (HI) installed within the containment.

Rapid Depressurization Function

The rapid depressurization function is a multi-purpose, dedicated system designed to serve an important role in severe accident mitigation. In the event that a high-pressure meltdown scenario develops and the feed portion of feed and bleed cooling capability cannot be established due to unavailability of the SI pumps, the rapid depressurization function is provided to depressurize the RCS and prevent high-pressure melt ejection following a vessel breach. This function is achieved via remote manual operator control as part of a severe accident management strategy. Upon entering a severe accident condition, the operator starts rapid depressurization by opening the required POSRVs.

Reactor Cavity Design

The reactor cavity is configured to promote retention of, and heat removal from, the postulated core debris during a severe accident. The cavity floor area allows for spreading of the core debris, enhancing its coolability within the reactor cavity region.

Emergency Containment Spray Backup System

The ECSBS provides an alternative to the containment spray system (CSS). The purpose of the ECSBS is to protect the containment integrity against overpressure and prevent the uncontrollable release of radioactive materials into the environment when the CSS is unavailable. The emergency containment spray flow path is from external water sources (i.e., the reactor makeup water tank, demineralized water storage tank, fresh water tank, or the raw water tank), through the fire protection system line via the diesel-driven fire pump, to the ECSBS line emergency connection located at ground level near the auxiliary building.

The applicant asserts that the ECSBS flow rate provides sufficient heat removal to prevent containment pressure from exceeding limits established for severe accidents.

Equipment Survivability

The application describes the scope and the acceptance criteria of the equipment survivability program. Equipment survivability refers to equipment required to mitigate the conditions following a severe accident, the functions which the equipment must accomplish, the environment in which the equipment must function, and the time frame for which the equipment must remain functional. The application describes in detail an approach to the equipment survivability evaluation. The application also states that the COL applicant will be responsible for the actual evaluation.

The equipment survivability program identifies the essential equipment that is needed to mitigate severe accidents, the time frame for which the equipment is required, and the environmental conditions the equipment must function under. These conditions include containment pressure, temperature, and radiation. The equipment must be designed to function in the severe accident environment for as long as needed to perform its mitigative function.

The severe accident environmental conditions are identified by evaluating a spectrum of accident sequences and dominant PRA sequences. Severe accident environmental conditions are defined by how challenging they are – severely, highly, quite, moderately, or nominally. The containment regions associated with each of these temperature assessments are identified in DCD Tier 2 Table 19.2.3-5, "Summary of Temperature Envelopes for Equipment Survivability Assessment." Temperature profiles from DCD Tier 2 Figures 19.2.3-16 through 19.2.3-20 correspond to the temperature envelopes identified in DCD Table 19.2.3-5. These temperatures range from 1,200°K short term and 600°K long term for severely challenging environments, to 460°K indefinitely for nominally challenging environments.

The bounding containment pressure expected, from MAAP results, is 110 psia.

The bounding severe accident radiation environment dose was calculated by MAAP4-DOSE as 4.4E+5 Gray (Gy).

<u>Technical evaluation</u>

External Reactor Vessel Cooling

Section 19.1.3.2 of this report evaluates the external reactor vessel cooling system and determines its acceptability.

Hydrogen Generation and Control

Section 6.2.5 of this report evaluates hydrogen generation and control and determines its acceptability.

MCCI and Core Debris Coolability

The MCCI is a severe accident phenomenon that involves the melting and decomposition of concrete in contact with molten corium. This phenomenon may occur following accident sequences that result in a breach of the reactor vessel due to molten corium and its spreading onto the reactor cavity floor. The thickness of the corium layer within the cavity depends upon the amount of core debris, its spreadability, and the cavity floor area. Once on the cavity floor, the molten corium may react with the concrete and any available water, producing noncondensible gases, water vapor, and heat from exothermic reactions.

MCCI can challenge the containment by various mechanisms including pressurization from noncondensible gases and steam generated, destruction of structural support members, and melt-through of the containment liner. Noncondensible gases, primarily carbon dioxide, carbon monoxide, and hydrogen, are released from the concrete as it decomposes and are formed from reactions between water and metals within the molten corium. The corium and concrete are heated from the combined effects of decay heat and exothermic chemical reactions.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff recommended that the Commission approve the position that both the evolutionary and passive LWR designs meet the following criteria:

1) Provide reactor cavity floor space to enhance debris spreading

2) Provide a means to flood the reactor cavity to assist in the cooling process

3) Protect the containment liner and other structural members with concrete, if necessary

4) Ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or Factored Load Category for concrete containments, for approximately 24 hours. Ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

In its July 21, 1993, SRM, the Commission approved the staff's position.

Regarding SECY-93-087 guidance item (1) above, the APR1400 design consists of two features, reactor cavity design and CFS, to mitigate MCCI. The reactor cavity is designed to maximize the unobstructed floor area available for the spreading of core debris. The cavity floor comprises an area available for core debris spreading such that the floor area/reactor thermal power ratio is larger than 0.02 m2/MWt. The staff found that this is consistent with SECY-93-087 guidance item (1).

Regarding SECY-93-087 guidance items (2) and (3) above, the containment liner plate in reactor cavity area is embedded 0.91 m (3 ft) below from the cavity floor at the minimum and actuation of CFS can provide flooding up to 6.4 m (21 ft) above the reactor cavity floor (elevation 69 ft 0 in) enabling cooling of core debris if spread on the reactor cavity. The staff found that these values are consistent with SECY-93-087 guidance items (2) and (3).

The applicant performed the MCCI analysis for the reactor cavity with the MAAP computer code using model parameters that were adjusted according to the results of the debris coolability code CORQUENCH (DCD Section 19.2.3.3.3.2). However, the DCD Revision 0 does not describe how the model parameters were adjusted. Therefore, in RAI 432-8377, Question 19-67, Part (a), the staff asked the applicant to describe how the model parameters were adjusted. In this question, Part (b) asked applicant to provide MCCI results for a case with no overlying water present in the cavity and Part (c) pointed to an inconsistency in DCD, Figure 19.2.3-7, related to identification of the calculation case (ML16068A097).

The applicant responded to this RAI on November 10, 2016 (ML16309A628). Part (a) of the response described how the applicant used CORQUENCH calculation results to adjust the MAAP model parameters ENT0C and FCHF for the purposes of MCCI analysis. ENT0C is a coefficient multiplier to the total mass of particles stripped from the corium jet when it flows into a deep-water pool. The applicant set the ENT0C to a low value in order to disable particle stripping, and thus the heat transfer between corium and water as corium is relocating out of the vessel and into the pool of water in the reactor cavity. This results in more corium at high temperature reaching the concrete, which conservatively increases the calculated ablation depth. FCHF is the Kutateladze number for corium to water heat transfer, which controls the magnitude of the heat flux. The applicant adjusted FCHF so that the reactor cavity ablation depth predicted by MAAP 4.0.8 was approximately the same as the reactor cavity ablation depth predicted by CORQUENCH 3.03 for a conservative large LOCA sequence with full core relocation into the reactor cavity. In APR1400-E-P-NR-14003, “Severe Accident Analysis Technical Report,” Appendix B, Section 3.2, the applicant states that it selected CORQUENCH 3.03 to calibrate MAAP 4.0.8 code because it has “detailed modeling features for a corium-water interaction and a melt eruption.” The staff finds this approach acceptable because FCHF was adequately adjusted using CORQUENCH, which contains more detailed modeling for MCCI and debris coolability than in MAAP. Therefore, the staff finds that the applicant’s response to Part (a) is acceptable.

In response Part (b), the applicant provided a report that includes MCCI calculation results for a case with a dry cavity for staff audit. The staff audited the calculation and found that the initial and boundary conditions and results are reasonable and therefore acceptable. In Part (c) response, the applicant provided a corrected Figure 19.2.3-7 and proposed to update the DCD with it. The staff reviewed the applicant’s response and finds it acceptable because the correction of discrepancy. The staff confirmed that the applicant incorporated this change in DCD Tier 2 Revision 1. RAI 432-8377, Question 19-67, is resolved and closed.

The largest amount of concrete erosion in the reactor cavity was predicted to occur for the large-break LOCA scenario. This scenario models a large-break LOCA with MAAP predicting early vessel failure and some debris retained in the reactor vessel lower plenum. MAAP predicted an ablation depth of 0.24 m (0.79 ft), which is less than the 0.91 m (3 ft) depth of the containment liner embedded in the reactor cavity. As shown in DCD Figure 19.2.3-12, the predicted containment pressure remains below the 8.7 kg/cm2 (123.7 psia) for 24 hours following the onset of core damage.

The APR1400 design has a reactor cavity sump in which corium may accumulate to a deeper level compared to the rest of the cavity area. The DCD Revision 0 does not provide details of the MCCI analysis performed for the reactor cavity sump. Therefore, in RAI 432-8377, Question 19-65, the staff asked the applicant to provide details (ML16068A097). In its response, dated May 14, 2016, the applicant proposed to update DCD Sections 19.2.3.3.3.2 and 19.2.3.3.3.3.2 summarizing the MCCI analysis performed for the reactor cavity sump (ML16135A003). The applicant used the CORQUENCH computer code for a large-break LOCA sequence for analyzing MCCI for the reactor cavity sump. The update to DCD Tier 2 Section 19.2.3.3.3.2 provided initial conditions and the types of concrete used for the analysis as limestone and LCS (limestone and common sand). The update to DCD Tier 2 Section 19.2.3.3.3.3.2 includes the following:

> The limiting case for MCCI analysis is a LBLOCA with 100 percent core relocation into the reactor cavity resulting in complete spreading into the cavity sump. Approximately 35,000 kg (77,000 pound-mass [lbm]) of debris flows into the sump. The CORQUENCH results for this sequence indicate that the corium in the sump is stabilized in less than 10 hours and the maximum ablation depth of the concrete is approximately 0.44 m (1.44 ft), well short of the containment liner.

The staff reviewed the response to find that the applicant has performed MCCI analysis conservatively and provided DCD update. Therefore, the staff finds that the applicant's response is acceptable. The staff confirmed that the applicant incorporated this change in DCD Tier 2 Revision 1. RAI 432-8377, Question 19-65, is resolved and closed.

The applicant also provided a technical report, "Ex-Vessel Severe Accident Analysis for the APR1400 with the MELTSPREAD and CORQUENCH Codes," dated August 28, 2012, for NRC audit. Table 9 of this report listed calculation results of concrete ablation depth for three different types of concrete for the cavity: siliceous, limestone-common sand, and limestone-limestone. However, Table 9 did not list calculation results of ablation depth for siliceous concrete for the reactor cavity sump. Therefore, during an audit teleconference on May 28, 2015, the staff asked the applicant to clarify the calculation results. In the response, dated October 1, 2015, the applicant stated the following (ML15274A284):

> The corium pool in the reactor cavity sump has a different cross-section area from the pool in the remaining cavity, and the walls and the floor in the sump may be subject to deeper ablation than the remaining cavity walls and floor because of the dimensions of the reactor cavity sump provided above. The siliceous type shows deeper ablation than the other types which are the limestone-limestone and the limestone-common sand. Accordingly, the ablation depth in the reactor cavity sump for the siliceous concrete does not meet the requirement and it is not included as the applicable materials in Table 9.

By having a lower ablation temperature and lower heat of decomposition than limestone-common sand and limestone-limestone concrete, the staff expects siliceous concrete to ablate with lesser thermal energy, and therefore, to have higher ablation under same conditions. For MCCI in the APR1400 cavity, the reactor cavity sump is the critical location as (1) it can accumulate core melt to a deeper depth that results in transferring more thermal energy from core melt to concrete causing it to ablate and (2) the bottom of the sump is closer to the containment liner than the rest of the reactor cavity. The applicant has not chosen siliceous concrete for the APR1400 design based on MCCI analysis for the reactor cavity sump. Therefore, staff finds that the applicant not providing the results of ablation of siliceous concrete in the report provided for audit acceptable.

Regarding SECY-93-087 guidance item (4) above, DCD Tier 2 Revision 1, Figure 19.2.3-12, shows for a large-break LOCA event involving MCCI in the reactor cavity, the containment pressure remains below the FLC pressure of 8.7 kg/cm2 (123.7 psia) for 24 hours following the onset of core damage. SECY-93-087 also recommends that the containment temperature resulting from MCCI should not exceed the FLC temperature within 24 hours. However, the staff could not find in APR1400-E-P-NR-14003 whether the applicant used this recommendation. Therefore, during an audit teleconference on February 1, 2018 (ML18122A341), the staff asked the applicant to clarify. In the response, dated March 8, 2018, the applicant stated the following (ML18067A847):

- The STC 11 in APR1400-K-P-NR-013603 represents the containment fails late by basemat melt-through MCCI. The containment atmosphere temperatures resulting from STC 11 from MCCI were evaluated within 25.5 hours. The maximum temperature was 431 °K (316 °F).

- Containment wall temperature resulting from MCCI sequences in APR1400-E-P-NR-14003, "Severe Accident Analysis Report," Appendix B, are indicated in Calculation Note (1-035-N389-501, Rev.4). The highest peak temperature among these sequences is 322 ℉ as discussed in response to RAI [199-]8223, Question 03.08.01-10 Rev.4, dated March 20, 2017 (ML17079A083).

- SECY-93-087 recommends to ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed the FLC for concrete containments, for approximately 24 hours and RG 1.216 refer this SECY-93-087. As a conservative approach, the constant temperature of 350℉ was selected for the liner temperature. This temperature was then applied in the structural analysis as stated in response to RAI [199-]8223, Question 03.08.01-8 Rev.1, dated July 25, 2016 (ML16207A127).

The applicant stated that in summary, the conservative selection of the temperature resulting from MCCI is made in containment integrity assessment, and the results show that the containment is capable of providing a barrier against the uncontrolled release of fission products for the more likely severe accident challenges, in accordance with RG 1.216 and SECY-93-087.

The staff evaluations applicant's responses to RAI 199-8223, Questions 03.08.01-8 and 03.08.01-10 mentioned above are provided in Section 3.8.1 of this report. The staff reviewed the applicant's response to find that it clarified that the applicant has used the SECY-93-087 guidance on environmental conditions (pressure and temperature). As such the staff found that the APR1400 design is consistent with SECY-93-087 guidance item (4).

The staff concludes the APR1400 design addresses MCCI and core debris coolability consistent with the guidance provided in SECY-93-087.

Direct Containment Heating and High Pressure Melt Ejection

Under conditions of high RCS pressure at the time of reactor vessel failure, a potential exists for the rapid ejection of molten core debris into the containment atmosphere, leading to rapid oxidation, hydrogen combustion, and convective energy transfer. This process is known as direct containment heating (DCH), which could lead to a rapid pressure increase in the containment, and potentially early containment failure. SECY-93-087 recommends that the evolutionary reactors include a depressurization system and cavity design features to reduce the RCS pressure and contain the ejected core melt. Consistent with this recommendation, the APR1400 design includes a rapid depressurization function and reactor cavity design to reduce the risk of HPME and DCH.

As stated in DCD Section 19.2.3.3.4.2.2, the applicant used NUREG/CR-6338 methodology for the DCH/HPME evaluation. This methodology combines the two-cell equilibrium model and Latin hypercube sampling methods. The applicant provided HPME/DCH analysis in APR1400-E-P-NR-14003, “Severe Accident Analysis Report,” Revision 0, Appendix C-1, “Severe Accident Analysis Report for HPME/DCH” (ML15009A225). However, APR1400-E-P-NR-14003 does not provide probability density curves used for the initial mass of UO2 in melt at vessel breach, the fraction of Zr oxidized, and variations in the coherence ratio used as input to the DCH/HPME evaluation. In addition, APR1400-E-P-NR-14003 gives a list of 26 input parameter values used in the analysis but does not provide their values. Therefore, in RAI 432-8377, Question 19-66, the staff requested the applicant to provide the missing information (ML16068A097).

In the response to Question 19-66, dated June 24, 2016 (ML16176A378), the applicant provided the missing probability density curves and their bases and listed input values used for 26 input parameters. The staff reviewed the applicant’s response and finds that the probability density curves and input parameters are realistic or conservative, and therefore, acceptable. RAI 432-8377, Question 19-66, is resolved and closed.

Based on the availability of the rapid depressurization function and reactor cavity design which provide mitigation and the applicant’s analysis, the staff concludes that the APR1400 design has addressed HPME/DCH consistent with the guidance provided in SECY-93-087, and therefore, it is acceptable.

Fuel Coolant Interactions: In- and Ex-Vessel Steam Explosions

The containment function can be challenged by energetic or rapid energy releases. One such energetic or rapid energy release is an FCI that results in a steam explosion. The term "steam explosion" refers to a phenomenon in which molten fuel fragments and transfers its energy to the coolant, resulting in rapid steam generation, shock waves, and possible mechanical damage. To be a safety concern, the interaction must be rapid and must involve a large fraction of the core mass. Steam explosions can occur either in-vessel or ex-vessel.

As stated in DCD Tier 2 Revision 0, Section 19.2.3.3.5.2.1, the FCI expert review group, sponsored by the NRC, concluded in NUREG-1116 and NUREG-1524 that the probability of this failure was vanishingly small or physically unreasonable (References 17 and 18). The FCI experts in OECD/NEA FCI specialist meeting confirmed this conclusion in May 1997 (Reference 19). The applicant performed an IVSE analysis to confirm the applicability of the experts’ conclusions to the APR1400 design; however, the DCD does not summarize the analysis performed. Therefore, in RAI 432-8377, Question 19-62 the staff requested the applicant to provide the summary (ML16068A097). In the response to Question 19-62, dated July 11, 2016 (MI16193A689), the applicant proposed new DCD Tier 2, Sections 19.2.3.3.5.1.1 and 19.2.3.3.5.2.1, summarizing the analysis performed. The staff reviewed the applicant’s response and found that it is consistent with the industry practice, and therefore, acceptable.

The staff confirmed that the applicant incorporated these changes in DCD Tier 2 Revision 1. RAI 432-8377, Question 19-62, is resolved and closed.

The applicant documented the IVSE analysis in APR1400-E-P-NR-14003, "Severe Accident Analysis Report," Revision 0, Appendix D, "Severe Accident Analysis Report for FCI." The applicant used TEXAS-V computer code to calculate steam explosion loading. Since the TEXAS computer code is a one-dimensional calculation, the selection of optimal nodal area would drive the results. The APR1400-E-P-NR-14003 report does not describe how the nodal area was chosen. Furthermore, Section 3.4.1 of APR1400-E-P-NR-14003 states that "[t]he penetration velocity profile [in Figure 3-2(a)] shows the typical corium penetration behavior in TEXAS where the corium jet is injected with the initial velocity and rapidly decelerated where the initial jet break-up occurs and start accelerating again." Nonetheless, the report does not describe the reasons for a second deceleration and subsequent acceleration of the jet. In addition, the report does not provide the value of initial void fraction assumed for the melt jet and the information on steam explosion triggering time. Therefore, in RAI 432-8377, Question 19-63, the staff requested the applicant to provide the following information (ML16068A097).

a. Explanation and justification for using one-dimensional TEXAS-V computer code for in-vessel steam explosion analysis,

b. Reasons for second deceleration and subsequent acceleration of the melt jet shown in APR1400-E-P-NR-14003, Figure 3-2(a),

c. Initial void fraction assumed for the melt jet, and

d. Justification for the triggering time assumed.

The applicant responded to Question 19-63 on March 27, 2017 (ML17058A238). In the response to Part (a) of the question, the applicant stated the following.

> The TEXAS-V code is a transient, one-dimensional model capable of simulating fuel-coolant interactions. And to maximize the fuel mass participates in the explosion, the external trigger when the jet touches the bottom of the lower head. Thus the constant cross-sectional area nodes system for the lower head zone is employed in IVSE analysis rather than considering the hemi-spherical shape of the lower head. TEXAS-V, therefore, can provide more conservative estimation of IVSE loading at the given initial conditions by selecting optimal cross-section area based on energy index . . .

The applicant chose the nodal area for the calculations (i.e., ARIY) to give a maximum energy load based on the energy index concept, e.g., when the ratio of the given melt's initial thermal energy and the coolant energy occurs in the optimal range, the explosion pressure is maximized. The applicant provided a figure with graphs of variations of peak explosion pressure and maximum impulse load with respective energy index. The applicant conservatively used the ARIY value corresponding to the energy index resulting in the highest pressure and impulse load as shown in the figure, which the staff finds acceptable. The staff finds that the applicant's use of one-dimensional TEXAS computer code and its selection of nodal area are conservative and acceptable.

In the response to Part (b) of the RAI question asking for melt jet deceleration/acceleration behavior, the applicant stated the following.

> TEXAS-V models LaGrangian particle filed for the melt as discrete material volumes or 'master particles' within Eulerian control volume for coolant vapor and liquid. In TEXAS-V code only discrete fuel masses and the leading edge may undergo hydrodynamic fragmentation. In addition, the TEXAS-V models the fuel jet as a collection of master particles and the jet breakup is attributed to Rayleigh-Taylor instabilities at the jet leading edge. As an approximation of the actual coherent jet, this jet is taken to be composed of a series of discrete 'blobs' or master particles that enter the coolant sequentially with the jet leading edge found by the relative position of the first unfragmented master particle 'blob' compared to the position of the master particles preceding it.
>
> Therefore, for the velocity profile given in Figure 3-2(a), the deceleration zone represents the influence of the fragmentation of the first leading master particle. After the completion of the first master particle fragmentation, the preceding (or the second) master particle then has a leading position and will have the hydrodynamic fragmentation which leads to the second deceleration.

The staff finds that the applicant's response explains the melt jet deceleration/acceleration behavior shown in the figure, and therefore, acceptable.

In response to Part (c) of the RAI question asking for the assumed initial melt void fraction, the applicant stated that the initial void fraction of the melt jet is set to be zero, which the staff finds conservative and acceptable.

In response to Part (d) of the RAI question asking for justification of IVSE triggering time, the applicant stated that it assumed an artificial trigger to occur by the corium jet contact at the bottom of RPV. The staff finds that the explosion triggering process that occurs when the melt contacts the bottom of the RPV is conservative, because triggering earlier or later would result in less melt involved in the explosion, generating less energy. Triggering earlier would involve less melt mass because the melt continues to be poured in. Triggering later would involve less melt mass to be in the vicinity of water because voiding would increase with time. Therefore, the staff finds that the applicant's response to Part (d) of the RAI acceptable. RAI 432-8377, Question 19-63, is resolved and closed.

If core debris and water come into contact after vessel breach, fuel-coolant interactions can cause the containment pressure to increase. In certain circumstances, steam explosions could occur, leading to a highly energetic pressure rise. SECY-93-087 states that any dynamic forces resulting from ex-vessel FCI on the integrity of the containment should be evaluated. The applicant provides an EVSE analysis in APR1400-E-P-NR-14003, "Severe Accident Analysis Report," Revision 0, Appendix D, "Severe Accident Analysis Report for FCI." However, APR1400-E-P-NR-14003, Revision 0 does not provide a justification for using the TEXAS-V computer code to calculate EVSE loading nor justification for the cross-section area assumed for the calculations. The initial void fraction of the melt jet and the steam explosion triggering time, which affects the steam explosion energetics, are not provided. APR1400-E-P-NR-14003, Revision 0, Table 4-17, shows the cavity structural analysis results, lists number of cracks as

47,073 with a maximum crack width of 0.027 in., and a remark of having considerable concrete damage. However, Table 5-1 of the report remarks that ex-vessel steam explosion causes “no threat” to the APR1400 design. Therefore, in RAI 432-8377, Question 19-64, the staff asked the applicant to provide the following information (ML16068A097):

a. Explanation and justification for using one-dimensional TEXAS-V computer code for analyzing ex-vessel steam explosion in the reactor cavity with a large cross-sectional area,

b. Justification for using one-dimensional analysis considering mixing of a 0.2 m2 melt jet in a significantly larger mixing area of 7 m2,

c. The initial void fraction of the melt jet assumed,

d. Justification for the triggering time assumed, and

e. Clarification for that a possible concrete damage with 47,073 cracks would not cause a threat to the APR1400 cavity design.

The applicant responded on January 23, 2017 to address the requested information (ML17023A329). In the responses to Parts (a) and (b) of the RAI question, the applicant stated that it chose the nodal area for the calculations (i.e., ARIY) to give a maximum energy load based on the energy index concept, i.e., when the ratio of the given melt’s initial thermal energy and the coolant energy occurs in the optimal range, the explosion pressure is maximized. The applicant provided a figure with graphs of variations of peak explosion pressure and maximum impulse load with respective energy index. However, the applicant did not use the ARIY value corresponding to the energy index resulting in highest pressure and impulse load shown on the figure. The applicant’s response provided the following explanation for choosing a lower than optimal ARIY value:

> It is seen from this figure that the explosion energy increases along the energy index, and it begins to fluctuate as it reaches a transition region. After the transition region, the explosion energy decreases abruptly. As the index increases, the total vapor fraction in the cavity coolant also increases, leading to higher energetics. However, after the index exceeds a certain value (the optimal value), the vapor fraction increases much faster and the explosion energetics are reduced. This indicates that the vapor fraction and the energy index have a non-linear relationship, reflecting the jet break-up and several other explosion dynamical phenomena. If the vapor fraction increases rapidly, the explosion energy decreases quickly. As mentioned above, the calculated explosion energy fluctuates substantially in the transition region, due to the vapor fraction intermittently exceeding a certain threshold value. In this region, the area effect is minor, and the explosion energy is driven by the vapor fraction in accordance with the axial dynamic effects. Hence, the selection of the energy index value from the region that precedes the transition region appears to be a reasonable way to achieve a stable, converged solution.

As stated in APR1400-E-P-NR-14003, the applicant used the structural analysis performed for a reference plant, Shin-Kori Units 3&4, for the reactor cavity under steam explosion loading. The applicant proposed to update DCD Tier 2 Section 19.2.3.3.5.2.2, with the following clarification:

> The structural assessment of reactor cavity under EVSE loading was performed in the reference plant project. The results of reactor cavity structural assessment of [the] reference plant are applicable to the APR1400 because the design parameters such as geometry, material properties, rebar arrangement, and design codes are the same between the reference plant and the APR1400. In addition, the EVSE pressure time history curve obtained from the APR1400 is almost identical to that of the reference plant with small perturbation after peak pressure. It is noted that this difference is negligible because the dynamic structural response depends on the peak value and its time.

The applicant also clarified that it did not take credit for the In-Vessel Retention and External Reactor Vessel Cooling (IVR-ERVC) strategy. The evaluation of the steam explosion load and consequential structural integrity assessment under the IVR-ERVC situation will be performed as a COL item and the applicant proposed to update DCD Section 19.2.7 and Table 1.8-2 to state the following:

> [t]he COL applicant and/or holder is to develop and submit an accident management plan including the evaluation of the effect of higher water level in the cavity on steam explosion loading when using In-Vessel Retention and External Reactor Vessel Cooling for accident management.

After completing its review of Parts (a) and (b) of the applicant's response, the staff raised the following concerns at a public meeting with the applicant on April 11, 2017 (ML18122A351):

1. AIRY value - Although "the calculated explosion energy fluctuates substantially in the transition region," the graph of peak pressure versus explosion energy shows an upward trend with a peak corresponding to an optimal ARIY value. Therefore, the applicant's choosing of a lower than optimal ARIY value to obtain a stable, converged solution was not justified.

2. Equation for attenuating pressure loading - As shown in Equation 4.1 in APR1400-E-P-NR-14003, "Severe Accident Analysis Report," Revision 0, Appendix D, "Severe Accident Analysis Report for FCI," the applicant used the following equation to calculate the attenuated pressure loading on cavity structure, ΔP_m:

$$\Delta P_m = \Delta P_{max} \left(\frac{1}{r}\right)^{\propto}$$

where ΔP_{max} is the maximum peak explosion pressure calculated pressure calculated using TEXAS computer code, r is the radial distance from the axis of explosion zone modeled, and α is the attenuation constant. The staff found that the above equation is dimensionally incorrect and does not contain a variable representing the radius of mixing boundary, R_{mix}. which the predicted pressure is assumed to be constant for the one-dimensional TEXAS computer code calculation. The staff determined that the correct equation should be given as follows:

$$\Delta P_m = \Delta P_{max}\left(\frac{R_{mix}}{r}\right)^{\propto}$$

In addition, APR1400-E-P-NR-14003, Revision 0, Figure 4-17 shows that the applicant has used Equation 4.1 to attenuate pressure in vertical direction resulting in non-conservatively lower pressures rather than using the predicted pressures from TEXAS computer code for the same locations.

3. In-Vessel Retention and External Reactor Vessel Cooling (IVR-ERVC) strategy - The APR1400 reactor cavity can be flooded to cover the bottom of RPV in order to cool the core melt in-vessel thus eliminate a possibility of occurring ex-vessel steam explosions. However, as analyzed for the certified AP600 design, when using IVR-ERVC strategy a vessel failure may be postulated to occur by a global creep rupture leading to "unzipping" of the lower head at or near the transition between the hemispherical lower head and cylindrical vessel structure. Such a failure mode may generate a larger melt jet than that was assumed for analysis. In addition, having a deeper water pool than analyzed may provide longer time for fragmentation of melt causing more melt to participate in steam explosions. Both a larger melt jet and a deeper water pool would generate higher steam explosion energetics than analyzed. The staff raised this concern during the APR1400 Chapter 19 audit. In the response, dated January 23, 2017, the applicant stated the following:

> [W]ith regarding In-Vessel Retention and External Reactor Vessel Cooling (IVR-ERVC) strategy, we did not give credit [for] IVR-ERVC system at present. The adoption of this strategy is related with Accident Management (AM). Severe Accident Management Guideline (SAMG) continent to activation of IVR-ERVC is also constructed in AM procedure. Therefore the evaluation of the steam explosion load and consequential structural integrity assessment under the IVR-ERVC situation will be performed as COL item.

However, the staff's review finds that IVR/ERVC is a design-specific system, and thus the COL applicant should not be responsible for performing the structural integrity assessment.

On August 28, 2017, the applicant submitted a supplement to its previous RAI 432-8377, Question 19-64, response to address the above concerns (ML17240A414). The staff evaluation of this response is provided below.

1. Regarding the AIRY value, the applicant conservatively selected the optimal value for ARIY from the graph of peak pressure versus explosion energy for the

TEXAS calculation and proposed to update APR1400-E-P-NR-14003 with this change. This addressed the staff concern on choosing nodal area for the TEXAS calculation.

2. Regarding the equation for attenuating pressure loading, the applicant agreed with the staff on correcting the equation used for attenuating pressure loading on the structures and proposed to update APR1400-E-P-NR-14003. The applicant conservatively imposed the TEXAS computer code predicted peak pressure without attenuation on the near cavity wall regions, i.e., in Zones 2 and 3 in Figure 4-20 of APR1400-E-P-NR-14003 (DCD Figure 4-18). The applicant used Equation 4.1 only for attenuating pressure in Zone 1 corresponding conservatively to the shortest distance from explosion location in that zone.

In response to the staff concern on attenuating pressure in vertical direction, the applicant proposed to update APR1400-E-P-NR-14003 report as follows:

> In terms of the vertical wall segments of Zone A to Zone F illustrated in Figure 4-18, the pressure profile for zones under water (Zone A, B, and C) is selected from the highest explosion pressure curves predicted by TEXAS-V at corresponding nodes (Figure 4-8 (A)). For example, Zone A covers the 6 nodes of TEXAS-V. The highest pressure among these 6 nodes calculated by TEXAS-V is chosen and applied to input load for the whole Zone A. The Zone B and C has the same approach. For the air-exposed cavity wall zones (Zone D, E, and F), we assume the same pressure of Zone C, conservatively.

The staff finds that the applicant's response acceptable because in addition to correcting Equation 4.1 the applicant has conservatively limited its use for attenuating pressure explosion pressure loading on structure walls.

3. Regarding the IVR-ERVC strategy, in justification for leaving the structural integrity assessment for IVR/ERVC strategy for the COL applicant, the applicant stated the following:

> [The] applicant of APR1400 plants in United Arab Emirates (UAE) does not select IVR-ERVC strategy. In the case of Korean domestic APR1400 plants such as Shin Kori Units 3&4 the applicant selects the IVRERVC as SAMG. Then the studies for performance of IVR-ERVC and consequences from IVR-ERVC failure were done during Operating License Stage because they are highly related with Operator actions, ESF's possibility, and so on.

Considering that the IVR-ERVC strategy will be covered under accident management requiring operator action, the staff concludes that allocating the structural integrity assessment for IVR/ERVC strategy to the COL applicant is acceptable.

As discussed above, the staff finds the applicant's responses to Question 19-64, Parts (a) and (b) acceptable.

The applicant's responses to Question 19-64, Parts (c) and (d) are the same as the responses to RAI 432-8377, Question 19-63, which the staff evaluated and found acceptable for the reasons explained in the discussion of Question 19-63, above.

In the response to Question 19-64, Part (e), concerning the reactor cavity concrete cracks described in APR1400-E-P-NR-14003, Revision 0, the applicant stated there should be no through cracks in concrete, and therefore, concrete damage would not cause a threat to the cavity design. Furthermore, the applicant described in regards to the liner integrity that the maximum stress and strain in the containment liner under steam explosion loading are below the liner failure limits. Regarding the concrete strength, as described in the APR1400-E-P-NR-14003 report, the applicant conservatively did not credit a concrete strength increase factor related to American Concrete Institute 349 code requirements for concrete quality, mixing, and placing. Conservatively, the applicant did not use a concrete strength increase factor in its analysis. Further, the applicant's evaluation used a liner failure criteria acceptable to the staff as described in RG 1.217.

Based on the revised pressure loading for the reactor cavity, as described above, the applicant reevaluated the structural integrity of the reactor cavity. The respective updated response including revisions to the DCD and APR1400-E-P-NR-14003 were provided in the applicant's letter dated August 28, 2017. The revised analysis showed additional concrete cracking and increased stresses and strains in the liner. However, as described and demonstrated in the applicant's response, such revised results would not impact the aforementioned conclusions. The concrete damage does not cause a threat to the reactor cavity and the maximum stress and strain in the containment liner remained below the liner failure limits. On this basis and based on the use of conservative pressure loading, conservative material properties, and failure criteria that is acceptable to the staff, as described above, the staff concludes that the applicant's analysis reasonably demonstrates that the integrity of the reactor cavity and containment is maintained when subjected to the EVSE pressure loads. The staff confirmed that DCD Revision 2 reflects applicant's proposed changes. RAI 432-8377, Question 19-64, is resolved and closed.

Containment Bypass

In SECY-90-016, the staff concluded that a special effort should be made to eliminate or further reduce the likelihood of a sequence that could bypass the containment. SECY-93-087 states that vendors should make reasonable efforts to minimize the possibility of bypass leakage and should account, in their containment designs, for a certain amount of bypass leakage. Two types of accident scenarios would lead to containment bypass, SGTR and ISLOCA. ISLOCA is evaluated in Section 19.2.2.5 above.

DCD Section 19.2.3.3.6.1 states the following.

> A thermally induced [SGTR] can occur in severe accident sequences where the primary system is at high pressure during core melt. This condition leads to creep rupture of the steam generator tubes due to the high-pressure and high-temperature conditions. The APR1400 design mitigates this possibility by operator actuation of the required POSRVs. This system is capable of reducing pressure in the reactor

vessel below 17.58 kg/cm2 (250 psia). Once primary system pressure reaches this level, there is essentially no risk of an induced tube rupture occurring.

DCD Section 19.2.3.3.3.1.2 states that "[t]ypical indications of core uncovery include (1) core-exit thermocouple (CET) temperatures in excess of 922.04 K (1,200 °F), (2) reactor vessel level monitoring system (RVLMS) readings indicative of no liquid above the fuel alignment plate, and (3) significant changes in readings of self-powered neutron detectors (SPND)." It was not clear to the staff that after receiving an indication of core uncovery whether the operator would have sufficient time to actuate POSRVs to prevent a thermally induced SGTR from occurring. Therefore, in RAI 432-8377, Question 19-69, the staff asked the applicant to confirm this operator action (ML16068A097). In the response to Question 19-69, dated May 14, 2016, the applicant provided the rapid depressurization analysis and its results showing that if the operator manually opens four POSRVs regardless of timing, the RCS pressure would be decreased below 250 psia within one hour (ML16135A003). Based on these results the applicant expects that in the APR1400 design the operator can have sufficient time to actuate POSRVs for mitigating thermally induced SGTR (ML16135A003). The staff finds that applicant's response is acceptable because the results confirm that the operator would have sufficient time to actuate POSRVs for mitigating thermally induced SGTR. RAI 432-8377, Question 19-69, is resolved and closed.

SECY-93-087 concludes that containment bypass resulting from SGTRs can be a significant challenge to containment integrity and that the plant designer should consider design features that would reduce or eliminate containment bypass leakage in such a scenario. In the SRM to SECY-93-087, the Commission approved the staff's recommendation that the applicant for DC for a passive or evolutionary PWR assess design features to mitigate the amount of containment bypass leakage that could result from steam generator tube ruptures. However, the applicant did not adequately describe in its DCD Revision 0, how the design features of the APR1400 would reduce or eliminate containment bypass resulting from SGTRs. Therefore, in RAI 467-8394, Question 19-103, the staff asked the applicant to describe these features (ML16113A122).

In the response to Question 19-103, dated May 31, 2016 (ML16152B009), the applicant stated that the operator actions after an SGTR event normally include minimization of the break flow rate and control of primary and secondary pressures and levels by using plant components and systems according to the action steps of emergency operating procedure. The operator will cool the primary system to prevent lifting the main steam safety valves using the steam bypass control system. Choices available for the operator in the APR1400 design to reduce RCS pressure include operation of the main or auxiliary spray pressurizer, operation of charging and letdown, throttling of the safety injection pumps, and the RCS depressurization function of the RCGVS. After identifying the affected SG, the operator will isolate the affected steam generator.

For handling a multiple steam generator tube rupture event that is beyond the design basis accident, the applicant stated the following:

> The operator can actuate emergency blowdown (EBD) to reduce the SG water level using the periodical high-capacity blowdown (HCBD) valve and piping. The SG blowdown system (SGBDS) is designed to assist in maintaining the chemical

> characteristics of the secondary side water within permissible limits during normal operation and anticipated operational occurrences (AOOs) such as a main condenser tube leak or SG primary-to-secondary tube leakage. The SGBDS is also designed to remove impurities concentrated in SGs by continuous blowdown (CBD), HCBD, and EBD.
>
> A detailed description of the SGBD system and operator action for SGTR is included in DCD Sections 10.4.8 and 15.6.3, respectively.

The staff finds the applicant's response acceptable because it fully described APR1400 design features that would reduce or eliminate containment bypass resulting from SGTRs, conforming to the SECY-93-087. RAI 467-8394, Question 19-103, is resolved and closed.

Given the design features described above and evaluated in Section 19.2.2.5 of this report that are consistent with SECY-90-016 recommendations, the staff concludes that the containment bypass is not a significant contributor to severe accidents for the APR1400 design.

Equipment Survivability

Equipment Survivability is evaluated below and the results are also discussed in Section 6.2.5 of this report.

To identify the equipment and instrumentation required to mitigate a severe accident, several accident sequences were selected from the most probable core damage sequences from the Level 1 PRA, and, also from several representative LOCAs. The selected initiating events are: large break LOCA, medium break LOCA, small break LOCA, total loss of feedwater, and station blackout. The required equipment for severe accident mitigation is grouped according to function: RCS inventory control; RCS heat removal; reactivity control; and, containment integrity. Specific equipment required to achieve each function is then identified in DCD Tier 2 Table 19.2.3-4, "Systems and Equipment/Instrumentation Required for Equipment Survivability Assessments." Staff reviewed this process, and concurs that the equipment identified by the applicant in Table 19.2.3-4 should be the basis of the equipment survivability program. The staff, however, also identified additional equipment which should be added to the list. Therefore, on October 22, 2015, the staff issued RAI 264-8243, Question 6.2.5-6, requesting the applicant to address this issue (ML15296A022). The staff also requested that the containment piping penetration assemblies and the emergency containment spray backup system inside containment isolation check valve, ECSBS-V1014, be added to the list, as these are both credited for severe accident mitigation. The applicant agreed to add these items in the RAI response (ML16200A230). The staff confirmed that the applicant incorporated these changes in DCD Tier 2 Revision 1. The staff finds that the identity of the equipment in Table 19.2.3-4 required for severe accident mitigation meets the acceptance criteria of 10 CFR 50.44(c)(3).

To determine the severe accident temperature conditions, the applicant selected the bounding temperatures of the burning of hydrogen generated by the release in containment of the equivalent of a 100 percent fuel clad reaction. This is required under 10 CFR 50.44(c)(3). These temperature profiles were calculated by simulating a broad spectrum of accident sequences from severe accidents using the MAAP4 containment model. The temperature environments were categorized as challenging, i.e., either severely, highly, quite, moderately, or nominally. The temperature profile of each category is provided, and the resulting temperature profile is assigned to the severe accident mitigative equipment in each node. The results are found in DCD Tier 2 Table 19.2.3-5, "Summary of Temperature Envelopes for Equipment Survivability Assessment" and the associated DCD Tier 2 Figure 19.2.3-16 through Figure 19.2.3-20, "ES Curves for Challenging Environments." The staff has reviewed this process and found it acceptable as the results are based on a broad selection of severe accident scenarios involving hydrogen combustion in containment.

To confirm the temperature profiles from DCD Tier 2 Figures 19.2.3-16 through 19.2.3-20, the staff has performed calculations for two different scenarios, LLOCA and TLOFW, using MELCOR, a fully integrated, engineering-level computer code produced for the NRC used to model the progression of severe accidents in nuclear power plants. Staff has generally confirmed the temperature envelopes, finding higher short term peak temperatures and significantly lower long term temperatures for the LLOCA scenario, and lower short term and long term temperatures for the TLOFW scenario. The results of temperatures in various locations in containment during large scale combustion from the LLOCA analysis is found in ERI/NRC 16-208 report (ML16314E431 (non-publicly available)), "Assessment of Combustible Gas Control during Severe Accidents in APR1400," Figure 4.1 through Figure 4.6. The results of the TLOFW analysis can be found in ERI/NRC 16-208 report, Figure 4.8, "Temperature of Selected CVs during Delayed Rapid Depressurization." Since the only significant difference is the short term peak temperatures for LLOCA, and the period of high temperature is so brief, between 60 and 100 seconds, staff finds the temperature profiles in DCD Tier 2 Figures 19.2.3-16 through 19.2.3-20 representative of the environmental conditions created by the burning of hydrogen equivalent to that generated from the 100 percent oxidation of the fuel cladding, as required by of 10 CFR 50.44(c)(3).

To determine the severe accident pressure conditions, the applicant selected the bounding pressure of the burning of hydrogen generated by the release in containment of the equivalent of a 100 percent fuel clad coolant reaction. The applicant performed an adiabatic isochoric complete combustion (AICC) analysis to determine the peak containment pressure when combustible gases generated during the course of a severe accident burn all at once. The applicant selected the maximum pressure scenario. The severe accident bounding pressure from the MAAP4 analysis results as 110 psia.

The staff calculated the AICC pressure for all five base cases (LLOCA, MLOCA, SLOCA, SBO, and TLOFW) and from several sensitivity cases. The results range from 45.6 to 88.6 psia. The staff also compared results to the corresponding containment pressure of 68 psia from the LLOCA scenario performed for the equipment survivability AICC temperature profiles described above and in ERI/NRC 16-208 report. Since all of these pressures are bounded by the applicant's AICC pressure of 123.7 psia, the staff finds the containment pressure meets the equipment survivability acceptance criteria of 10 CFR 50.44(c)(3).

To determine the severe accident radiation conditions, the applicant selected the bounding radiation dose of 4.4E+5 Gy (4.4E+7 rad) that equipment in containment is expected to receive. This dose was calculated using MAAP4 output as input to DOSE and found to occur in the steam generator compartment for the loss of feedwater (LOFW) sequence. The applicant calculated the cumulative dose for three separate scenarios, i.e., SBO, LLOCA, and TLOFW.

Results from these calculations at 24 hours after the accident range from 3.0E+5 to 4.4E+5 Gy (3.0E+7 to 4.4E+7 rad), depending on the location in containment. The applicant then selected the scenario with the bounding dose.

The staff compared the results to those found for the AP1000, which is an advanced light water reactor of similar size, fuel type, containment design, with similarly evaluated severe accident scenarios, and found the containment doses comparable.

The equipment survivability analysis methodology is described in the DCD Tier 2 Section 19.2.3.3.7.3. The approach includes: relying on equipment qualification results for actual plant equipment; relying on results from research or experiments performed on similar equipment; and, relying on a thermal lag analysis to account for the real time for the temperature of the critical component to rise to the actual accident conditions. The DCD application provides further description and insight into the specific expected performance of the equipment identified. The staff has reviewed this methodology and the equipment identified as necessary to mitigate a beyond design basis accident, and found the approach acceptable for the COL applicant to use in their evaluation for site specific equipment. During the audit, staff noted that Item COL 19.2(1) should be revised to state that the COL applicant and/or holder is to perform and submit site specific equipment survivability assessment in accordance with 10 CFR 50.34(f) and 10 CFR 50.44, which reflects the equipment identified and the containment atmospheric assessments of temperature, pressure and radiation described in DCD Section 19.2.3.3.7. The applicant made this change in its revised response to RAI 264-8243, Question 6.2.5-6, dated November 18, 2016 (ML16323A488). The staff confirmed that the applicant incorporated these changes in DCD Tier 2 Revision 1. RAI 264-8243, Question 6.2.5-6, is resolved and closed.

APR1400-E-P-NR-14003, Rev. 0, Section 3.5 states that the ECSBS is actuated 24 hours after the onset of core damage and the ECSBS is capable of controlling the containment pressure for a period of 48 hours after 24 hours following the onset of core damage. The staff was concerned that adding water into the containment during the operating of ECSBS may cause flooding in the containment affecting the equipment, which is relied upon during severe accidents. Therefore, the staff raised this concern during the APR1400 Chapter 19 audit public teleconference on June 14, 2017 (ML18123A316). In response, the applicant provided the results of flooding evaluation when using ECSBS during severe accidents for staff audit. The applicant had performed this evaluation for a reference plant case, Shin-Kori 3&4. In RAI 551-8962, Question 19.02-1, the staff asked the applicant to revise the DCD summarizing the flooding evaluation.

In the response, dated August 28, 2017, the applicant provided results of equipment evaluation for containment flooding performed for the reference plant during ECSBS operation (ML17240A419). The applicant identified the following equipment and instrumentation needed beyond the 24 hours after the onset of core damage when ECSBS will be used if needed:

- Hydrogen Mitigation System
- Containment penetrations
- Containment Spray System

- SCS pumps as a backup to containment spray pumps
- ECSBS
- Hydrogen monitors
- High level radiation monitors
- Containment Monitoring System

The applicant summarized the results of flooding evaluation for the above equipment and instrumentation and concluded that the essential function of each equipment and instrumentations required to perform in the beyond 24 hours after onset of core damage can survive on the containment flooding. The applicant agreed to update APR1400-E-P-NR-14003 with the results of the flooding evaluation. In addition, recognizing that the final location of each equipment and instrumentations can be modified and finalized at the detailed design stage, except a few equipment, the applicant proposed to revise COL 19.2(1) to state that the site specific equipment survivability assessment including flooding effect shall be performed by COL applicant. The staff review finds that the applicant's response is consistent with SRP Section 19, and therefore, acceptable. The staff confirmed that DCD Revision 2 reflects the above change to COL 19.2(1). RAI 551-8962, Question 19.02-1, is resolved and closed.

The staff finds that the methodology, identification of equipment required for accident mitigation, and environmental conditions provided by the applicant to address equipment survivability establishes sufficient guidance and input for the COL applicant to demonstrate compliance with the acceptance criteria of 10 CFR 50.44(c)(3) and 10 CFR 50.34(f).

MELCOR Confirmatory Analysis

Summary of Application

NRC regulations 10 CFR 52.47(a)(23) and 10 CFR 52.47(a)(27) require that the applicant perform a PRA and an analysis of design features for the prevention and mitigation of severe accidents. The applicant's PRA analysis of severe accident design features relied on a MAAP model of the APR1400 design. The applicant developed and applied this MAAP model to analyze thermal hydraulics, accident progression, and source term for a range of severe accident scenarios. The applicant's MAAP analysis is described in References 11 and 44 through 48.

Technical Evaluation

The applicable regulatory requirements for analyzing plant response to severe accident scenarios are described in 10 CFR 52.47(a)(23) and 10 CFR 52.47(a)(27). The staff reviewed the applicant's simulation of the accident progression, analysis methodology, and interpretations of its analyses of the reactor, containment, and system response to severe accidents. The staff's review included performing independent assessment of the plant response to selected severe accident scenarios using MELCOR.

The staff examined accident scenarios from the applicant's PRA and severe accident mitigation features analysis. The staff selected scenarios based on risk importance and performed independent MELCOR confirmatory calculations for the selected scenarios. The staff then compared the results of its MELCOR calculations with the applicant's results and engaged with the applicant to resolve the differences in the results. The staff's examination is described below.

The staff's review of the applicant's MAAP analysis identified about eighty MAAP calculations which serve as the basis for the APR1400 Level 2 PRA and SA analysis in DCD Chapter 19. Nineteen of the calculations were run to estimate source terms for the range of severe accident scenarios for the APR1400 design. The remainder were run to examine additional variations of these scenarios needed to estimate branch point probabilities for the PRA and for the analysis of severe accident design features.

To enable the selection of accident scenarios based on frequency, consequence, and dominant risk, the staff grouped the MAAP calculations to estimate source terms by containment failure mode as shown in Table 19.2-2. The individual MAAP case run for each source term category is shown in Table 19.2-3.

Table 19.2-2. Summary of MAAP Source Term Analysis

Containment failure mode	Source term categories (Reference 44)	Scenario analyzed with MAAP	Internal events release frequency (per year)	Fraction of cesium released
Bypass	1, 2	SGTR	7E-8	Up to 0.17
Bypass	3, 4	ISLOCA	1E-10	Up to 0.43
Isolation failure	5, 6	LOOP, LOCCW	3E-9	Up to 0.03
Before vessel breach	7, 8	MLOCA	2E-8	Up to 0.19
No containment failure (core damage arrested in-vessel)	9	MLOCA	3E-7	0.0000063
No containment failure (core damage arrested ex-vessel)	10	LOCCW	8E-7	0.000015
After vessel breach	11 – 21	LOCCW	1E-7	Up to 0.04
Abbreviations: ISLOCA - interfacing systems loss of coolant accident; LOCCW - loss of component cooling water; LOOP - loss of offsite power; MAAP - modular accident analysis program; MLOCA - medium loss of coolant accident; SGTR - steam generator tube rupture				

Table 19.2-3. MAAP Case Run for Each Source Term Category

MAAP case	Characteristics of source term category					Scenario analyzed with MAAP	Severe accident mitigation systems modeled with MAAP		
	Containment failure time	Containment leak	Containment rupture	Containment spray	Wet cavity		Rapid depressurization	CFS	ECSBS
STC01	Bypass					SGTR			
STC02	"					SGTR – scrubbed			
STC03	"					ISLOCA			
STC04	"					ISLOCA – scrubbed			
STC05	Isolation failure			✓		LOOP			✓
STC06	"					LOCCW	✓		
STC07	CF < VB	✓				MLOCA			
STC08	"		✓			"			
STC09	None					"			
STC10	None					LOCCW	✓	✓	✓
STC11	Basemat					"	✓		✓
STC12	Early	✓				No sequences			
STC13	"		✓			LOCCW	✓	✓	
STC14	Late	✓		✓		"	✓		✓
STC15	"	✓		✓	✓	No sequences			
STC16	"	✓				LOCCW	✓		
STC17	"	✓			✓	"	✓	✓	
STC18	"		✓	✓		"	✓		✓
STC19	"		✓	✓	✓	"	✓	✓	✓
STC20	"		✓			"	✓		
STC21	"		✓		✓	"	✓	✓	

Abbreviations: CF - containment failure; CFS – cavity flooding system; ECSBS - emergency containment spray backup system; ISLOCA - interfacing systems loss of coolant accident; LOCCW - loss of component cooling water; LOOP - loss of offsite power; MAAP - modular accident analysis program; MLOCA - medium loss of coolant accident; SGTR - steam generator tube rupture; STC - source term category; VB - vessel breach

The APR1400 design includes severe accident systems intended to provide improved mitigation for new reactor designs. These systems include the (a) RCS depressurization system, (b) CFS, (c) ECSBS, and (d) hydrogen control system.

RCS depressurization capability is provided by pilot-operated safety relief valves on the pressurizer. The pressurizer has four pilot-operated safety relief valves and two of them are designated for use as the rapid depressurization system which is intended to reduce RCS pressure sufficiently to prevent high pressure melt ejection. Accident management procedures would direct the operators to open these two valves when the core exit thermocouple temperature reaches 1,200°F (649 C).

Cavity flooding capability is provided by motor-operated valves in piping connecting the IRWST with the cavity. Opening these valves gravity feeds IRWST water into the cavity up to the bottom surface of the reactor vessel. Accident management procedures would direct the operators to open these valves when the core exit thermocouple temperature reaches 1,200°F (649 C).

The emergency containment backup spray is provided by the fire protection system or a fire truck. This provides a source of external water to the containment through a dedicated spray header.

Hydrogen control is provided through passive autocatalytic recombiners and igniters. In addition, three-way valves are located downstream of the pressurizer's pilot-operated safety relief valves described above. Accident management procedures would direct the operators to open the three-way valves when the core exit thermocouple temperature reaches 1,200 °F (649 ˚C) to direct releases from the pilot-operated safety relief valves into the steam generator cubicles, instead of into the IRWST, to prevent buildup of hydrogen in the IRWST.

The staff selected the following source term category (STC) cases for the independent analysis:

- STC10 because it has the highest frequency. STC10 is an LOCCW scenario which includes a loss of primary, secondary, and containment heat removal and, as a result, is a station-blackout-like scenario (i.e., a boil-off scenario). STC10 includes the severe accident mitigation systems described above.

- STC11 to examine a case where the CFS did not operate. STC11 uses the same boundary conditions as STC10, but with no operation of the CFS.

- STC16 to examine a case where both the CFS and the ECSBS do not operate. STC16 uses the same boundary conditions as STC10 but with no operation of the CFS, no operation of the ECSBS, and with an assumed containment leak at 24 hours. STC16 also assumes no operation of igniters or PARs.

- Q03 to examine the additional variation of the rapid depressurization system also not functioning. The applicant's objective in running Q03 was not to assess the source term, but instead to assess containment pressure when the cavity was dry and without containment spray. Q03 uses the same boundary conditions as STC10 but with no operation of the CFS, no operation of ECSBS, and no operation of the rapid depressurization system. Q03, which is a station blackout, also assumes no operation of igniters.

The staff also considered containment bypass scenarios. The staff did not select ISLOCA for independent confirmatory analysis with MELCOR due to its low likelihood.

The staff considered steam generator tube rupture scenarios in two categories. The first category involves a tube rupture occurring before core damage. The tube rupture is either a spontaneous failure of a tube or is caused by depressurization of the secondary system. State-of-the-art reactor consequence analysis (SOARCA) in Reference 49 showed that such a scenario is unlikely to lead to core damage because ample time and resources would be available to depressurize and cool down the reactor.

The second category involves a tube rupture occurring during core damage as a result of high-temperature creep rupture. This type of rupture (severe accident-induced tube rupture) has been analyzed in earlier NRC studies (e.g., Reference 53). More recently, the SOARCA study (Reference 49) showed this scenario to be of less risk importance than non-bypass scenarios due to more realistic accounting for physical phenomena that would occur in such a scenario including aerosol deposition in the steam generator (Aerosol Trapping in a Steam Generator Test (ARTIST) test program) and subsequent severe accident-induced hot leg creep rupture resulting in redirecting the release into containment where it can deposit instead of going out through the ruptured tube into the environment. As a result of the above considerations together with the low likelihood of a steam generator tube rupture scenario for APR1400, the staff did not select an SGTR scenario for independent confirmatory analysis.

In addition to the at-power cases, the staff selected a shutdown accident occurring during mid-loop operations corresponding to MAAP case POS 5, because of the low RCS water level at this time in the refueling operation. Table 19.2-4 summarizes the scenarios selected by the staff for independent confirmatory analysis.

Table 19.2-4. MAAP Cases Selected for Confirmatory Analysis

MAAP case	Objective of MAAP case	Accident initiator	Severe accident mitigation systems activated		
			Rapid depressurization	CFS	ECSBS at 24 hours
STC10	Source term for “no containment failure” – leakage only	LOCCW	✓	✓	✓
STC11	Source term for basemat melt-through	LOCCW	✓		✓
STC16	Source term for containment leak at 24 hours	LOCCW	✓		
Q03	Containment pressure for dry cavity	SBO			
POS5	Time to core damage, lower head failure	Loss of SDC and injection			
Abbreviations: CFS – cavity flooding system; ECSBS - emergency containment spray backup system; LOCCW - loss of component cooling water; MAAP - modular accident analysis program; POS - plant operating state; SBO - station blackout; SDC - shutdown cooling; STC - source term category					

The staff developed a MELCOR model for APR1400 using plant design data provided by the applicant. The model is described in References 50 and 51. The staff applied the model to each of the scenarios in Table 19.2-4 using best-estimate assumptions regarding systems, operator actions, and phenomenology. The staff compared the results of each MELCOR calculation with the results for the corresponding MAAP calculation in Table 19.2-4. The detailed MAAP results provided by the applicant in References 55 and 56 were used for the comparison.

Differences in the MELCOR/MAAP results comparison included assumed availability and timing of the operation of mitigation systems (e.g., SIT injection, POSRVs and associated 3-way valves, and emergency containment backup spray system), and phenomenological assumptions (e.g., treatment of RCP seal leakage and severe accident-induced hot leg rupture). The staff engaged with the applicant at a December 16-17, 2015, public meeting (ML17025A047), to begin the effort to understand the differences in the studies.

On March 3, 2016, the staff issued RAI 426-8492, Question 19-53, to request the applicant address the differences seen in the MELCOR/MAAP comparisons performed by the staff (ML16061A146).

On July 15, 2016, the applicant provided its partial response to RAI 426-8492, Question 19-53 (ML16167A268). The partial response addressed the staff's questions associated with the MAAP at-power cases the staff examined (i.e., STC10, STC11, STC16, and Q03). The applicant stated that the systems availability assumptions were based on the systems availability modeled in the Level 1 PRA. The response also stated that, for the MAAP runs to estimate source terms, the assumed containment failure times and modes were chosen based on the individual source term category definitions. The response included MAAP sensitivity cases for STC10, STC11, STC16, and Q03 using alternative assumptions to examine the impact of the alternative assumptions on the results.

The objective of MAAP Cases STC10, STC11, and STC16 was to evaluate source terms for a) estimating large release frequency for comparison to the surrogate safety goals and b) estimating benefits for cost-benefits analysis in the environmental assessment used to demonstrate no cost-beneficial severe accident mitigation design alternatives (SAMDAs). For STC10, STC11, and STC16, the applicant concluded that differences between the MAAP calculations documented in the DCD and the sensitivity calculations in the RAI response were insignificant, because the releases for these cases did not change from small to large.

The objective of MAAP Case Q03 was to evaluate containment overpressure and over-temperature failure. For Q03, the applicant concluded that the pressure differences between the MAAP calculations documented in the DCD and the sensitivity calculations in the RAI response were insignificant relative to the containment failure pressure.

On October 23, 2017, the applicant provided a revised response to RAI 426-8492, Question 19-53 (ML17296A136). The revised response included additional information addressing the staff's questions associated with the MAAP shut-down case the staff examined – POS 5. POS 5 refers to the plant operating state with the steam generators isolated from the rest of the RCS, nozzle dams installed in RCS piping, the pressurizer manway removed, and the water level at the mid-plane of the hot legs. Regarding the MAAP POS 5 analysis, the RAI requested information regarding the MAAP modeling of the configuration of the RCS during mid-loop operation and the treatment of physical phenomena related to fission product deposition in the RCS and containment. As a result of the RAI, the applicant found and corrected issues with the MAAP analysis for POS 5 and provided updated FSAR pages to reflect the results of the revised analysis. The staff concludes that the applicant's response to RAI 426-8492, Question 19-53, adequately addressed the issues raised. Based on the review of the DCD to verify that it incorporated the text as shown in the RAI response, the staff has confirmed incorporation of the changes described above; therefore, RAI 426-8492, Question 19-53, is resolved and closed.

Effect of Applicant MAAP Assumptions on other MAAP Calculations Performed for Chapter 19

The staff extrapolated the results of the MAAP sensitivity calculations summarized in the revised RAI response to RAI 426-8492, Question 19-53 (ML17296A136) for STC10, STC11, and

STC16, and Q03 to the other MAAP calculations performed for Chapter 19 to gauge the potential impact.

Surrogate Safety Goals

The DCD uses the estimates of LRF from the APR1400 PRA to show that the LRF goal of less than 1E-6/year and the CCFP goal of less than 0.1 are met. The CCFP is estimated as the ratio of LRF to CDF. Scaling the cesium releases by the differences between Cases STC10, STC11, and STC16 in the DCD and the Cases STC10, STC11, and STC16 sensitivities in the RAI response does not result in changing the release from small to large for any source term category. Therefore, the applicant's surrogate safety goal comparisons (to the LRF and CCFP surrogate safety goals) are not likely to be affected.

Severe Accident Mitigation Design Alternatives (SAMDA) analysis

For at-power internal, fire, and flood events, the total expected person-rem per year is 1.29 and the total expected property costs is $3,512. Seventy percent of the total expected person-rem per year and 76 percent of the total expected property costs are from STC01, which is the source term category for unscrubbed SGTR. Scaling the cesium releases in the DCD by the differences between Cases STC10, STC11, and STC16 in the DCD and the Cases STC10, STC11, and STC16 sensitivities in the RAI response is unlikely to change the overall consequences significantly, considering the margins in the SAMDA analysis. Also, any increase in overall costs from increased releases for other source term categories could be offset by a more realistic analysis for STC01. STC01 has a 17 percent cesium release which is likely to be an overestimate considering the SOARCA study's (Reference 49) predicted cesium release of 0.4 percent for unscrubbed SGTR. Therefore, the differences between MAAP calculations in the FSAR and the sensitivity MAAP calculations in the RAI response are not likely to affect the SAMDA analysis.

CET quantification

The objective of MAAP Case Q03 was to evaluate containment overpressure and over-temperature failure for CET quantification. As a result of the staff's MELCOR confirmatory calculation for Case Q03 (which included SITs injection), the staff identified that the applicant's MAAP analysis did not include SITs injection. To address this issue, the applicant performed MAAP sensitivity calculations for Case Q03 which included SITs injection. The main effect seen in the MAAP sensitivity calculations for Case Q03 was that the predicted containment pressure was higher than in the MAAP base case. The higher pressure was a result of the additional steam generated from the water in the SITs going into the RCS and becoming vaporized.

Cases A01 through A15 (Reference 45) were performed to predict RCS pressure at time of core damage. Because the RCS pressure at the time of core damage is generally governed by whether there is an opening in the RCS (e.g., POSRVs open), assumptions regarding whether SITs function would not significantly impact these cases.

Cases T01 through T04 (Reference 46) were performed to predict the number of POSRV and main steam safety valve cycles before core damage for high pressure scenarios. Because SITs do not inject for high pressure scenarios, there is no impact of assumptions regarding whether SITs function.

Cases S01 through S04 (Reference 46) were performed to evaluate containment pressure with and without ECSBS actuation at 24 hours to confirm that ECSBS actuation at 24 hours would prevent containment over-pressurization when the containment spray system failed. Cases S01 and S02 were run without ECSBS, and S03 and S04 are the corresponding cases with ECSBS. These MAAP calculations showed that ECSBS operation halts containment pressurization preventing containment overpressure failure. The pressure difference between Case Q03 in the DCD and the Case Q03 sensitivities in the RAI response would not affect the conclusion that ECSBS prevents containment overpressure failure. Therefore, there is no significant impact of assumptions regarding whether SITs function.

Cases Q01 through Q03 (Reference 46) were performed to evaluate containment overpressure and over-temperature failure with a dry cavity. MAAP predicts containment pressures are below the containment median ultimate failure pressure of 162.7 psig. The pressure difference between Case Q03 in the DCD and the Case Q03 sensitivities in the RAI response would not affect the conclusion that the containment will not fail from over-pressurization within a day. Therefore, there is no significant impact of assumptions regarding whether SITs function.

Cases R01 through R10 (Reference 46) were performed to predict the maximum containment pressure during a station blackout accident for ten cases with hydrogen combustion. The MAAP-predicted containment pressure following combustion for each case was used to estimate containment failure probability. The containment failure probabilities were then used to quantify an event tree. It is not clear from the documentation (Reference 46) whether SITs injection was included in Cases R01 through R10. However, the pressure difference between Case Q03 in the DCD and the Case Q03 sensitivities in the RAI response is unlikely to impact failure probabilities for the APR1400 containment which has an ultimate failure pressure of 162.7 psig which is significantly higher. Also, the overall impact of the higher pressures predicted from including SITs injection in the MAAP analysis on the surrogate safety goal comparisons (to the LRF and CCFP surrogate safety goals) and the SAMDA analysis is likely to be small, because these analyses are dominated by bypass accidents.

Severe Accident Analysis Report

MAAP cases were analyzed in Appendix A of the Severe Accident Analysis Technical Report (Reference 11) to examine the effect of hydrogen combustion on containment integrity. The staff's review of the results of the MAAP cases in Appendix A indicates that the pressure difference between Case Q03 in the DCD and the Case Q03 sensitivities in the RAI response would not impact the applicant's analysis of containment integrity.

MAAP cases were analyzed in Appendix B of the Severe Accident Analysis Technical Report (Reference 11) to examine the core-concrete interaction issue. Because these calculations assume SITs injection is available, there would be no impact of the assumption of SITs unavailable found in the audited MAAP calculations.

MAAP cases were analyzed in Appendix F of the Severe Accident Analysis Technical Report (Reference 11) to establish the equipment survivability environmental envelope. Because these MAAP cases include cases with and without SITs, there would be no impact of the assumption of SITs unavailable found in the audited MAAP calculations.

Comparison of MELCOR calculations with MAAP sensitivity calculations

The staff requested the MAAP results for sensitivity Cases STC-10a and STC10-all to compare against the staff's independent MELCOR calculation for STC10. These cases were requested because the alternative assumptions used by the applicant in these cases align more closely with the assumptions used in the independent MELCOR calculation for STC10 (e.g., SITs injects water into the RCS). The applicant provided the requested MAAP results on October 27, 2016 (ML16309A031). The staff's comparisons show improved agreement between MELCOR and MAAP results when SITs injection is included in the MAAP analysis. The staff's comparisons are documented in Reference 54.

The staff also compared its MELCOR-predicted cesium releases for STC10, STC11, and STC16 with the applicant's MAAP cesium hydroxide releases. The comparison indicates that the MAAP and MELCOR results are in general agreement. Therefore, the staff's MELCOR independent analysis confirms the applicant's MAAP simulations of the APR1400 design's thermal hydraulics, accident progression, and source term for severe accidents.

19.2.4 Containment Performance Capability

Summary of Application

The application states that the containment is designed so that the CCFP is below 0.1 and the containment meets applicable requirements of the ASME Code (ASME Section III, Division 2, Subarticle CC-3720 FLC).

Technical Evaluation

This section provides the staff's review and evaluation of the applicant's assessment of the APR1400 containment structural performance. The staff focused its review on the ability of the structural components comprising the containment pressure boundary to meet the guidance in SECY-93 087 and RG 1.216 for deterministic containment performance. The staff reviewed the applicant's assessment of deterministic containment performance to ensure such containment will remain essentially leak-tight when subjected to severe accident pressure loads for 24 hours after the onset of core damage.

Under SECY-93 087 and RG 1.216, the applicant should demonstrate that containment is able to maintain its structural integrity under beyond-design-basis internal pressure loadings. This is achieved by demonstrating that (1) the estimated ultimate pressure capacity of the containment structure provides adequate margin, (2) the containment maintains its structural adequacy under pressure loadings associated with combustible gas generation, and (3) the containment performance goal (CPG) for severe accidents is achieved.

DCD Tier 2 Section 3.8.1 "Concrete Containment," describes the APR1400 containment design and structural characteristics. DCD Tier 2 Section 3.8.1.4.11 "Ultimate Pressure Capacity," documents the containment ultimate pressure capacity evaluations and the containment structural evaluation for demonstrating the structural integrity for the hydrogen generated pressure loads. DCD Tier 2, Sections 19.1.4 "Safety Insights from the Internal Events PRA for Operations at Power" and 19.2.4 "Containment Performance Capability," describe the containment structural performance and capacity to withstand pressure loads induced by the more likely severe accident challenges.

Description of Containment

DCD Tier 2, Sections 3.8.1 and 19.2.3.1.1 "Description of Containment," describe the general arrangement of the prestressed concrete containment vessel (PCCV) for the DCD. The PCCV is a right circular cylinder topped by a hemispherical dome. The cylindrical containment and dome are prestressed by a posttensioning system consisting of horizontal and inverted "U" vertical tendons (American Society for Testing and Materials (ASTM) A416, Grade 270). Three buttresses equally spaced 240 degrees apart anchor the horizontal tendons. The dome is prestressed by horizontal tendons up to a 45 degree vertical angle and two groups of inverted "U" vertical tendons oriented 90 degrees to each other and anchored at the tendon gallery. The specified minimum compressive strength for the containment wall and dome is 41.37 megapascals (MPa) (6,000 pounds per square inch (psi)) at 91 days. The minimum compressive strength for the basemat is 34.5 MPa (5,000 psi). In addition, steel reinforcement (ASTM A615, Grade 60) is used in the PCCV cylinder and dome.

The containment has an inner diameter of 45.72 meters (m) (150 feet (ft)) and an inside height of 76.66 m (251 ft, 6 inches (in.)). The cylinder and dome have wall thicknesses of 1.37 m (4 ft, 6 in.) and 1.22 m (4 ft), respectively. The PCCV has a 6 millimeter (mm) (0.25 in) steel liner plate (American Society of Mechanical Engineers (ASME) SA 516, Grade 60 and ASME SA 240, Type 304) anchored to the concrete cylinder with embedded vertical angles. The liner plates are welded together and attached to the concrete dome with radial and hoop stiffeners.

Major penetrations include an equipment hatch (inside diameter of 7.92 m [26 ft]) and two personnel airlocks (inside diameter of 3.05 m [10 ft]). Additional penetrations include feedwater and main steam line penetrations. The concrete walls in the vicinity of these penetrations are thickened to provide additional strength.

The PCCV is designed to withstand various combinations of dead loads, live loads, environmental loads (including earthquakes), and loads generated by a postulated LOCA. As specified by General Design Criteria (GDC) 16, "Containment Design," and GDC 50, "Containment design basis," in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," the primary function of the containment is to provide the principal barrier to control potential fission product releases to the environment. The containment houses the reactor vessel, steam generators, reactor coolant loops, and portions of the auxiliary and engineered safety features systems. The applicant stated that the APR1400 containment has a design pressure of 4.218 kg/cm2 (60 psig) and a design temperature of 416.5 K (290F). The PCCV is also designed for a negative pressure of 27.6 kilopascals gauge (kPaG) (4 psig). Section 3.8.1 of this report describes the staff evaluation of the concrete containment.

Ultimate Pressure Capacity

10 CFR Part 50, Appendix A, GDC 50 "Containment design basis" requires the prediction of containment internal pressure capacity above design-basis pressure. In addition, GDC 50 requires the reactor containment structure and its internal components to accommodate the calculated pressure and temperature conditions caused by a LOCA without exceeding the design leakage rate and with sufficient margin. In DCD Tier 2 Section 3.8.1.4.11, the applicant stated that the ultimate pressure capacity of the PCCV is 1.089 MPaG (158 psig), at which the maximum strain of the liner plate is approximately 0.4 percent. Section 3.8.1 of this report describes the staff evaluation of the applicant's analysis of ultimate containment capacity.

Combustible Gases and Hydrogen Burning

10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors," requires that an analysis be performed to demonstrate the containment structural integrity, specifically that the analysis addresses loads generated in an accident that releases hydrogen from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. The regulation also requires a structural analysis of the containment under an internal pressurization resulting from 100 percent fuel clad-fuel reaction followed by hydrogen burning. Section 3.8.1 of this report describes the staff's evaluation of the applicant's analysis of this loading condition.

19.2.4.1 Containment Performance Goal

The Commission's CPG ensures that containment will maintain its structural integrity under loads associated with severe accident phenomena. The CPG includes both a deterministic and a probabilistic goal. The staff describes these goals in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, and SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993. The deterministic goal is to maintain containment integrity for approximately 24 hours following the onset of core damage for the more likely severe accident challenges. The probabilistic goal is for the CCFP to be less than approximately 0.1 for the composite of all credible core damage sequences evaluated in the PRA.

The scope of review described in this section of the SER is to evaluate the deterministic calculation and assumptions relating to model details, descriptions of computer codes, material properties and modeling, loading and loading sequences, failure modes, and the interpretation of results. This section of the SER describes the staff's evaluation of the applicant's analysis of the pressure associated with the ASME Boiler and Pressure Vessel Code (BPVC) (ASME Code), Section III, Division 2, Subarticle CC 3720 factored load category (FLC). Section 19.1.4 of this report describes the staff's evaluation of the probabilistic CCFP.

Summary of Application

DCD Tier 2 Section 19.2.4, addresses the SECY-93-087, Section I.J, guidance regarding the deterministic assessment of containment performance under the pressure and temperature loads generated for the more likely accident scenarios. The CPG is for containment to maintain its role as a reliable leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges, and, following this period, for containment to continue to provide a barrier against the uncontrolled release of fission products.

Technical Evaluation

This section describes the staff's assessment of the APR1400 containment structural performance based on the information provided in the DCD. RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design-Basis Pressure," issued August 2010, provides the primary basis for the staff's review. The staff primarily used the review criteria from RG 1.216, Regulatory Position 3, to perform the review and evaluation of the CPG and to determine the adequacy of the applicant's assessment of the containment structural performance. The staff reviewed the DCD, "Severe Accident Analysis Report" (APR1400-E-P-14003), and audited the "Containment Building Capacity Evaluation on Severe Accident (Global and Local)" calculations (1-316-C304-006 and 1-316-C304-007, respectively).

PCCV Model Description

Consistent with RG 1.216, Regulatory Position 3, the staff reviewed the development of the global and local finite element (FE) models of the containment using the approach described under Regulatory Position 1.

Section 5 of Calculation 1-316-C304-006 describes the applicant's analysis methods used to evaluate the PCCV under selected severe accidents. The PCCV was modeled and analyzed using the three-dimensional (3-D) FE software ABAQUS. In order to ensure an adequate level of detail in the DCD, on March 8, 2016, the staff issued RAI 433-8363, Question 19-70, to request additional details from the aforementioned calculation that should be included in the DCD (ML16068A099). These details include modeling details, description of computer codes, material properties and modeling, loading and loading sequences, and failure modes.

On August 30, 2016, the applicant submitted a response to RAI 433-8363, Question 19-70, which included DCD markups to address the requested information (ML16243A525). The applicant revised DCD Section 19.2.4.2.2 to describe the FE model of the concrete containment. The description provides the computer codes used and states the material models and properties for the various elements of the concrete containment (e.g., concrete wall and dome, liner plate, steel reinforcing bars, and tendons).

Based on the information provided to the staff, the staff finds that the level of detail included in the RAI response related to the FE model of the PCCV is acceptable. Consistent with RG 1.216, Regulatory Position 1, the staff concludes that the use of a 3 D FE model is acceptable. Additionally, the use of a static analysis is appropriate, the use of a nonlinear stress-strain relationship for concrete in compression is appropriate, and all material properties are based on the accident temperatures and developed consistent with NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories - An Overview," issued July 2006. The staff confirmed that DCD Revision 2 contains the changes committed to in the RAI response; therefore, RAI 433-8363, Question 19-70, is resolved and closed.

The staff also reviewed Calculation 1 316 C304 007 to evaluate the behavior of large penetrations. As stated in RG 1.216, ASME Code, Section III, Division 1, Subsection NE, covers the metal portions of concrete containments that are not backed by concrete. Therefore, such components need to be shown to meet the Subsection NE 3220 Service Level C requirements. The staff's review of Calculation 1 316 C3 4 007 did not find sufficient information to confirm that the covers for the large penetrations were analyzed to meet the Subsection NE-3220 Service Level C requirements. To address this issue, on March 8, 2016, the staff issued RAI 433-8363, Question 19-84, requesting the applicant demonstrate that the covers for the large penetrations meet the requirements of ASME Code, Subsection NE-3220, Service Level C (ML16068A099).

The applicant's July 15, 2016, response to RAI 433-8363, Question 19-84, added Item COL 19.2(3) as follows (ML16197A479):

> The COL applicant and/or holder will demonstrate that the covers for the large penetrations, such as equipment hatch and personnel airlocks, meet the Service Level C requirements in Subsection NE-3220 of the ASME Code and explain how the consideration of containment leakage is accounted for when modeling local regions of containment.

The staff finds that the applicant's Item COL 19.2(3) addresses the acceptance criteria for metal portions of concrete containments that are not backed by concrete, including the covers of large penetrations, consistent with RG 1.216. On this basis the staff finds Item COL 19.2(3) to be acceptable.

Severe Accident Scenarios

Consistent with RG 1.216, Regulatory Position 3, the staff reviewed the technical basis for identifying the more likely severe accident challenges. The staff then reviewed the application to determine whether the containment goal is met during the first 24 hours after onset of core damage.

In the DCD Section 19.2.4.2.2 "Containment Pressurization Results," the applicant stated that the highest containment pressure at 24 hours following the onset of core damage would result from a LLOCA scenario. The corresponding pressure produces strains below the allowable strains in ASME Code, Section III, Division 2, Subarticle CC 3720 FLC. In Section 4 of Calculation 1 316 C304 006, the applicant stated that the severe accident scenarios analyzed are those with the most significant pressure-loading histories, i.e., LLOCA, SBO, and TLOFW. The methodology used to select the severe accident challenges is not clear to the staff. The staff also notes that the DCD and Appendix E to APR1400 E P 14003 P state that the LLOCA results in the highest containment pressure; however, the analysis performed in Calculation 1-316-C304-006, Revision 2, uses the SBO as the representative severe accident scenario. Therefore, on March 8, 2016, the staff issued RAI 433-8363, Question 19-71, to request the applicant clarify the selection of severe accidents to determine the bounding pressure and temperature time history (ML16068A099).

In the response to RAI 433-8363, Question 19-71 (ML16196A260), dated February 27, 2017, the applicant stated the following.

> The top ten dominant sequences contributing to the core damage frequency (CDF) are selected from the Level 1 PRA results at the time of performing the analysis. Accident initiators for these sequences include: station blackout (SBO),

large break LOCA (LLOCA), small break LOCA (SLOCA), loss of feedwater (LOFW), and steam generator tube rupture (SGTR). These ten sequences account for 87.6 [percent] of the cumulative CDF. The applicant believes this to be an acceptable approach to identifying the more likely severe accident challenges since the probabilistic sequences and the dominant sequences from the deterministic approach are included.

The applicant provided a comparison of CDF from accident initiators for the draft Level 1 study used for containment performance analysis and final Level 1 study provided in APR1400 DCD Tier 2 Section 19.1. The applicant showed that "[b]oth top 10 sequences (87.6 percent of CDF) in draft Level 1 study and top 30 sequences (93.7 percent of CDF) in final Level 1 study is mainly composed by SBO, LLOCA, SLOCA, LOFW, and SGTR." In addition to the top 10 sequences, the applicant constructed and analyzed 9 sequences initiated by dominant events such as SBO, SLOCA, LOCA, SGTR, and LOFW from a deterministic approach. The staff finds the applicant's technical basis for identifying the more likely severe accident challenges acceptable because it is consistent with RG 1.216, Regulatory Position 3(a).

Additionally, in the response to RAI 199-8223, Question 3.8.1-10, dated March 20, 2017, the applicant provided a markup of DCD Section 19.2.4.2.2 related to the severe accident selection, which is evaluated in Section 3.8.1(D)(d). The applicant also provided the basis for using SBO as the representative severe accident scenario. As described in APR1400-E-P-14003P, Revision 0, Appendix F, "Severe Accident Analysis Report for Equipment Survivability Evaluation," SBO is the initiating event for the bounding pressure of the equipment survivability assessment. Further, the pressure response to the LLOCA and TLOFW sequences is also applied as input to the FE model to determine the acceptability of containment liner strains due to pressure and temperature loadings.

Consistent with RG 1.216, Regulatory Position 3, the staff also reviewed the application to determine whether the containment performance goal is met for the period after the initial 24 hours after the onset of core damage.

The applicant stated that for accident conditions beyond 24 hours, the corresponding pressure and temperature are enveloped by conditions during the first 24 hours. At 24 hours after the onset of core damage, the applicant stated that operation of the ECSBS is sufficient to prevent containment pressure from exceeding the FLC requirements in ASME Code, Subarticle CC 3720. DCD Figure 19.2.3-21 and Calculation 1 316 C304 006, Figure 4-1 depict plots of the maximum pressure generated for the LLOCA scenario. The plot depicts the maximum pressure of 110.9 psia for the LLOCA pressure load history. This maximum pressure is reached at 24 hours before actuation of the ECSBS.

This approach is consistent with RG 1.216 acceptance criteria for the period following the initial 24 hours after the onset of core damage. Multiple criteria from Regulatory Position 3 are met as follows.

- 3.2a.(1) - The maximum pressure and temperature following the initial 24 hour period are enveloped by the maximum pressure and temperature during the initial 24 hour period.

- 3.2a.(2) - The maximum pressure and temperature following the initial 24 hour period meet the factored load acceptance criteria.

Therefore, the staff finds the applicant meets the containment performance goal for the period after the initial 24 hours after the onset of core damage.

Thermal Loads

The staff reviewed the applicant's consideration of accident temperatures on the material properties of the containment elements (i.e., concrete, rebar, steel liner, tendons). Consistent with RG 1.216, for the severe accident analysis of containment pressure, it is acceptable to consider the thermal load effects by modifying the material properties to account for the accident temperature.

In DCD Section 3.8.1.5.3, "Acceptance Criteria with Respect to Concrete Temperatures," the applicant stated, "For accident or any other short-term period, temperatures are not to exceed 177 °C (350 °F) for the interior surface. However, local areas are allowed to reach 343 °C (650 °F) from steam or water jets in the event of a pipe rupture." Calculation 1 316 C304 006 states that the material degradations in strength and modulus are considered for the analysis. The applicant followed NUREG/CR-6906 and ASME Code, Section II-D, to determine the material degradation characteristics applicable to the containment materials. The staff finds the treatment of material properties based on the severe accident temperature considered for the analysis to be consistent with RG 1.216.

Material Properties

The staff reviewed the applicant's selection of material properties used for the PCCV analysis. In addition to the consideration of temperature on material properties, the staff reviewed the median strength and standard deviation.

Section 3 of Calculation 1 316 C304 006 describes the material properties used in the PCCV analysis. The applicant stated that it used generic data for material properties. As stated in RG 1.216, the material properties used in the analysis should be based on minimum code-specified properties and on the accident temperature considered for the analysis. The staff's review of Calculation 1-316-C304-006 finds references to the use of generic data, median strengths, and logarithmic standard deviations. To verify the consistency with the guidance on material properties in RG 1.216, on March 8, 2016, the staff issued RAI 433-8363, Question 19-82, to the applicant requesting such verification (ML16068A099). Furthermore, the staff asked the applicant to provide the basis and justification for using alternative methods from those in RG 1.216 for determining the material properties for use in analysis.

In the response to RAI 433-8363, Question 19-82, dated July 15, 2016 (ML16196A345), the applicant stated that the properties used are the minimum code specified properties. Further, the applicant stated that the degradation of concrete and steel material properties corresponding to the maximum temperature during severe accident is conservatively applied at all analysis phases. The applicant updated Section 3 of Calculation 1 316 C304 006 to clarify use of minimum code specified properties, which are adjusted as necessary to account for the severe accident temperature effects. The staff notes that actual material properties will obtain higher strength than the minimum code-specified properties and, therefore, finds the use of minimum code specified material properties to be conservative. Moreover, the use of minimum code specified properties and the associated consideration of severe accident temperature are consistent with RG 1.216 and, therefore, acceptable. Based on the above, RAI 433-8363, Question 19-82, is resolved and closed.

ASME Code Subarticle CC-3720 Factored Load Criteria

Consistent with RG 1.216, the staff evaluated the results of the severe accident analysis. The staff reviewed DCD Section 19.2.4 and APR1400-E-P-NR-14003, which contains proprietary information, to evaluate the applicant's demonstration that the strain limits as a result of the severe accident analysis are below the liner plate allowable strain limits corresponding to the ASME Code, Subarticle CC-3720, FLC and meet the Commission's CPG.

In DCD Section 19.2.4, the applicant compared the pressure resulting from the more likely severe accident scenarios against the FLC for concrete containments in ASME Code, Section III, Division 2, Subarticle CC 3720. DCD Section 19.2.4.2.1 states:

> The maximum pressure load on the containment structure is evaluated to be 7.0 kg/cm2 (99.8 psia) under the AICC [adiabatic isochoric complete combustion] condition. Considering the safety margin of APR1400 containment, for the FLC, the pressure resulting from 100 percent metal water reaction of fuel cladding and resulting from uncontrolled hydrogen burning is determined as 8.7 kg/cm2 (123.7 psia).

In DCD Section 19.2.4.2.2, the applicant stated that the containment pressure response for an LLOCA does not reach the 8.7 kg/cm2 (123.7 psia) pressure defined above.

Because the meaning of the statement in the previous paragraph was unclear to the staff, on March 8, 2016, the staff issued RAI 433-8363, Question 19-72, to ask the applicant to clarify the meaning of the phrase "considering the safety margin of the APR1400 containment" and to clarify the derivation or analysis that provides the basis for the 8.7 kg/cm2 (123.7 psia) pressure (ML16068A099). In the response to RAI 433-8363, Question 19-72, dated August 30, 2016 (ML16243A525), the applicant clarified that the 8.7 kg/cm2 (123.7 psia) pressure is not the pressure at which the liner strain equals the limits established by the ASME FLC requirements (i.e. the liner strains induced by this pressure do not reach the ASME FLC allowable strain values). Furthermore, all of the tendons and rebars remain in the elastic stage for this input into the FE analysis of the containment. This pressure was conservatively selected to represent severe accident load corresponding to the ASME FLC requirements. The applicant provided DCD markups to describe the use of the selected pressure (87 kg/cm2 [123.7 psia]) as the severe accident load corresponding to the FLC.

Additionally, the staff issued RAI 433-8363, Question 19-81, on March 8, 2016, to request the applicant to explain why the accident scenarios considered are those with the most significant pressure loadings (ML16068A099). In Revision 3 of Calculations 1-316-C304-006 and 1-316-C304-007, and as stated in the response to RAI 433-8363, Question 19-81 (ML16197A479), the bounding pressure selected from the more likely severe accident sequences are employed as the input load profile. The results of this analysis are provided in Section 19.2.4.2.2 and are discussed below. The applicant also considers the 8.7 kg/cm2

(123.7 psia) pressure as input to the FE analysis of containment in Appendix B to the calculations. This is consistent with the applicant's response to RAI 433-8363, Question 19-72, which states that the "severe accident load of 8.7 kg/cm2 (123.7 psia) is employed as the input loads for finite element analysis of the containment" (ML16243A525). The consideration of the bounding pressures selected from the more likely severe accident scenarios and the 8.7 kg/cm2 (123.7 psia) in the analysis of containment is acceptable to the staff.

In the response to RAI 433-8363, Question 19-70, dated February 2, 2017 (ML16243A525), the applicant provided the maximum compressive and tensile strains resulting from three pressure loadings (LLOCA, TLOFW, and SBO) in markups to DCD Section 19.2.4.2.2 and demonstrated that these strains are below the FLC strains for membrane only and combined membrane and bending. The staff finds that the liner strains from pressure resulting from the more likely severe accident scenarios are smaller than the ASME Code, Subarticle CC 3720, FLC strain limits, and that the analysis results show that the FLC criteria are met for the accident scenarios considered in the analysis. Therefore, consistent with RG 1.216, the staff finds the applicant meets the containment performance goal for the initial 24 hours after the onset of core damage.

19.2.5 Accident Management

19.2.5.1 Severe Accident Management Framework

The reactor designer develops a framework for implementation of accident management by the COL.

Summary of Application

Accident management encompasses those actions taken during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases.

Accident management consists of the actions taken by the plant's emergency response organization (including plant operations, technical support, and management staff), to prevent core damage, terminate core damage once it begins, maintain containment integrity, and minimize offsite radiation releases. Severe accident management refers to those actions that would mitigate the consequences of accidents that result in core damage. The objectives of a severe accident management program are to arrest core melt progression by cooling the molten core material, either in-vessel if possible, or ex-vessel if the debris has entered the containment building, and to ensure that fission products are not released to the environment. The ultimate objective is to achieve a safe and stable state. To accomplish these objectives, the emergency response organization should make full use of the plant's design features, including both standard and non-standard use of plant systems and equipment.

Technical Evaluation

The nuclear power industry initiated a coordinated program on accident management in 1990. Section 5 of NEI 91-04, Revision 1, "Severe Accident Closure Guidelines," lays out the elements of the industry's severe accident management closure actions that have been accepted by the NRC. This program involves the development of (1) a structured method by which utilities may systematically evaluate and enhance their abilities to deal with potential severe accidents, (2) vendor-specific accident management guidelines for use by individual utilities in establishing plant-specific accident management procedures and guidance, and (3) guidance and material to support utility activities related to training in severe accidents. Using the guidance developed through this program, each operating plant has implemented a plant specific accident management plan as part of an industry initiative.

Based on the staff's review of these efforts, severe accident evaluations in the individual plant examinations, and industry PRAs, the staff has concluded that improvements to utility accident management capabilities could further reduce the risk associated with severe accidents. Although reactor designs such as APR1400 will have enhanced capabilities for the prevention and mitigation of severe accidents, accident management will remain an important element of defense-in-depth for these designs. However, the increased attention to accident prevention and mitigation in these designs can be expected to alter the scope and focus of accident management relative to that for operating reactors. For example, increased attention to accident prevention and the development of error-tolerant designs can be expected to decrease the need for operator intervention, while increasing the time available for such action if necessary. This will tend to make it less likely for the emergency response organization to make rapid decisions and permit a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), the need for human intervention and accident management will continue.

For both operating and advanced reactors, the overall responsibility for accident management, including development, implementation, and maintenance of the accident management plan, lies with the nuclear plant operator, because the plant operator bears ultimate responsibility for the safety of the plant and for establishing and maintaining an emergency response organization capable of effectively responding to potential accident situations. For operating plants, vendors have played key roles in providing essential severe accident management guidance and strategies for implementation. This guidance has served as the basis for severe accident management procedures and for training personnel in carrying out the procedures. Computational aids for technical support have been developed, information needed to respond to a spectrum of severe accidents has been provided, decision-making responsibilities have been delineated, and utility self-evaluation methodologies have been developed and utilized.

DCD Section 19.2.5.1.1.2, "Accident Management - During Low-Power Shutdown Operations," states, "If RCS water level decreases too far, it can reach a level that is insufficient for shutdown cooling (SC) pump suction. If this occurs, SC pumps are isolated to prevent damage to the pumps. In this situation, the charging pumps can be used to increase RCS water level and allow resumed operation of the SCS." Based on staff's review of DCD Chapter 9, each charging pump has a rated flow rate of 155 gpm. The staff requested additional information to clarify whether one or two charging pumps are needed to keep the core covered early in the outage, starting with POS 3. On February 22, 2016, this information was requested by the staff in RAI 409-8325, Question 19-24 (ML16053A015).

In the response to RAI 409-8325, Question 19-24, dated April 12, 2016 (ML16103A538), the applicant updated DCD Section 19.2.5.1.1.2 to state that during POS 5, one charging pump is needed to keep the core covered. During POS 3 and POS 4, two charging pumps are required to keep the core covered. This response clarified the issue and is therefore acceptable to the staff. The staff confirmed that DCD Revision 1 contains the changes committed to in the RAI response; therefore, RAI 409-8325, Question 19-24, is resolved and closed.

19.2.5.2 **Consideration of Potential Design Improvements**

The NRC Environmental Assessment of the APR1400 design documents the staff evaluation of DCD Section 19.2.6.

19.2.6 Combined License Information

The applicant established COL information items relevant to the APR1400 SAE and tabulated the in DCD Section 19.2.7. During the course of the review of this section the staff identified additional COL information items and requested the applicant to consider inclusion in the COL information list. The basis for the request is discussed in the previous relevant section of this report. Below is the comprehensive list of COL items relevant to Section 19.2, "Severe Accident Evaluation."

The staff concludes that the established COL information items associated with DCD Section 19.2 on SEA, as listed below, are reasonable, sufficient, and in conformance with the SRP Section 19.0 guideline. Therefore they are all acceptable.

COL 19.2(1) The COL applicant and/or holder is to perform and submit site-specific equipment survivability assessment including flooding effect in accordance with 10 CFR 50.34(f) and 10 CFR 50.44 which reflects the equipment identified and the containment atmospheric assessments of temperature, pressure and radiation described in Subsection 19.2.3.3.7.

COL 19.2(2) The COL applicant and/or holder will demonstrate that the covers for the large penetrations such as equipment hatch and personnel airlocks meet the Service Level C requirements in Subsection NE-3220 of the ASME code and explain how the consideration of containment leakage is accounted for when modeling local regions of containment.

COL 19.2(3) The COL applicant and/or holder is to develop and submit an accident management plan including the evaluation of the effect of higher water level in the cavity on steam explosion loading when using In-Vessel Retention and External Reactor Vessel Cooling for accident management.

19.2.7 Conclusion and Finding

The staff evaluated the applicant's severe accident evaluation and identified several issues that were not adequately addressed in the initial submittal. The applicant has since addressed all of these issues adequately through its responses to the staff's RAIs and the staff questions raised during the regulatory audit. Based on its review, the staff finds that the applicant has adequately addressed the Commission's objectives, described above in Section 19.0.2, regarding the prevention and mitigation of severe accidents. The staff concludes that the applicant has properly assessed the APR1400 severe accidents in compliance with 10 CFR

52.47(a)(8) and (23) and in conformance with SECY-90-016, SECY-93-087, and SRP Section 19.0.

19.2.8 References

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2. Regulatory Guide 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design Basis Pressure," U.S. Nuclear Regulatory Commission, August 2010.

3. SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," U.S. Nuclear Regulatory Commission, June 1990.

4. ASME Section III, Division 2, "Rules of Construction of Nuclear Facility Components – Code for Concrete Containments," American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

5. NUREG/CR-5132, "Severe Accident Insights Report," U.S. Nuclear Regulatory Commission, April 1988.

6. NUREG/CR-5597, "In-Vessel Zircaloy Oxidation/Hydrogen Generation Behavior during Severe Accidents," U.S. Nuclear Regulatory Commission, September 1990.

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10. NEA/CSNI/R(2000)7, "Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety," OECD/NEA, October 2000.

11. APR1400-E-P-NR-14003, "Severe Accident Analysis Technical Report," KHNP, December 2014.

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16. 10 CFR 52.47, “Contents of Applications; Technical Information,” U.S. Nuclear Regulatory Commission, November 2012.

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18. NUREG-1524, “A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues,” U.S. Nuclear Regulatory Commission, August 1996.

19. NEA/CSNI/R(97)30, “OECD/CSNI Specialist Meeting on Fuel-Coolant Interactions,” OECD/NEA, May 1997.

20. M. L. Corradini et al., “A User’s Manual for TEXAS-V: A One Dimensional Transient Fluid Model for Fuel-Coolant Interaction Analysis,” UW Nuclear Engineering and Engineering Physics, August 2000.

21. R. H. Cole, “Underwater Explosions,” Princeton University Press, 1948.

22. ACI 349-06, “Code Requirements for Nuclear Safety-Related Concrete Structures (ACI 349-97) and Commentary,” American Concrete Institute, September 2007.

23. 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, November 2012.

24. 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, November 2012.

25. NUREG/CR-5334, “Severe Accident Testing of Electrical Penetration Assemblies,” U.S. Nuclear Regulatory Commission, November 1989.

26. NUREG/CR-4146, “Simulation of an EPRI-Nevada Test Site (NTS) Hydrogen Burn Test at the Central Receiver Test Facility,” U.S. Nuclear Regulatory Commission, June 1985.

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40. ERI/NRC 16-208, Revision 2, "Assessment of Combustible Gas Control during Severe Accidents in APR1400," (ML16314E431).

41. APR1400 PRA and SA Audit Summary Report, (completion in project Phase 4).

42. KHNP Calculation Note 1-035-N389-101, Revision 3, "Hydrogen Generation and Control during Severe Accidents."

43. KHNP Calculation Note 1-035-N389-102, Revision 1, "Assessment of AICC Pressure Load Due to Hydrogen Combustion in Containment" KHNP Calculation Note 1-035-N389-103, Revision 2, "Analysis of Local DDT Potential in the APR1400 Containment."

44. APR1400 Design Control Document Tier 2 Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation, APR1400-K-X-FS-14002, Revision 0., December 2014.

45. Full Power Level 2 PRA – PDS Analysis, APR1400-K-P-NR-013601, Revision 0, July 2013.

46. Full Power Level 2 PRA – CET/DET Analysis, APR1400-K-P-NR-013602, Revision 0, July 2013.

47. Full Power Level 2 PRA – Source Term Category Analysis, APR1400-K-P-NR-013603, Revision 0, July 2013.

48. LPSD Level 2 MAAP Analyses, APR1400-K-P-NR-013761, Revision 0, December 2014.

49. "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," NUREG-1935, November 2012.

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51. "Simulation of Representative Design Basis and Beyond Design Basis Accidents in APR1400," ERI/NRC 15-202, Revision 1, June 2015.

52. "MELCOR Confirmatory Analysis for APR1400," meeting between NRC and KHNP, December 16-17, 2015 (ML16019A047).

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19.3 Beyond Design Basis External Event

19.3.1 Recommendations 4.1 and 4.2 – Station Blackout and Mitigation Strategies for Beyond Design Basis External Events

19.3.1.1 Introduction

Following the Fukushima Dai-Ichi event, the Commission established additional requirements to manage and mitigate external events that are beyond the design basis of a nuclear plant. These are found in NRC Commission paper SECY-12-0025 (February 17, 2012), "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami" (ML12039A103). DCD Tier 2 Section 19.3 addresses the APR1400 conformance with SECY-12-0025, including the requirements contained in NRC Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," (ML12054A735).

The applicant submitted shutdown mitigating strategies for staff review described in Section 19.3 of the DCD and in the APR1400 Fukushima Technical Report. The APR1400 DCD Tier 2, Revision 2, Section 19.3.2.3, "Recommendations 4.1 and 4.2 – Station Blackout and Mitigation Strategies for Beyond Design Basis External Events," outlines the methods to cope with an extended loss of ac power (ELAP) event. The APR1400 design employs a three-phase approach for mitigating beyond design basis external events (BDBEE). The initial phase (Phase (1) requires the use of installed equipment and resources to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities. The transition phase (Phase 2) requires providing sufficient, portable onsite equipment and consumables to maintain or restore these functions. The final phase (Phase 3) requires obtaining sufficient offsite resources to sustain core cooling, containment, and SFP cooling indefinitely. In the initial review of the APR1400 DC application, the staff followed the guidance for satisfying the Commission directives regarding BDBEE mitigation strategies in Japan Lesson-Learned Project Directorate JLD-ISG-2012-01, Revision 0, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," (ML12229A174), which endorsed with clarifications the methodologies described in NEI 12–06, Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (ML12242A378). The guidance in JLD-ISG-2012-01 describes one acceptable approach for satisfying the Commission directives regarding BDBEE mitigation strategies (i.e., Order EA-12-049). The applicant describes APR1400 SFP mitigation strategies as being consistent with the guidelines addressed in NRC EA-12-049; NEI 12-06, Revision 0; and JLD-ISG-2012-01, Revision 0.

The APR1400 DCD Tier 2, Revision 2, Section 19.3.1, "Introduction," states that specific details addressing the NRC Fukushima Near-Term Task Force (NTTF) recommendations in "Recommendation for Enhancing Reactor Safety in the 21st Century," are provided in Technical Report APR1400-E-P-NR-14005, "Evaluations and Design Enhancements to Incorporate Lessons Learned from Fukushima Dai-Ichi Nuclear Accident," Revision 2, dated July 2017 (ML18044B041).

NTTF Recommendation 4.1 directed the staff to initiate rulemaking on how a nuclear site is to properly address an ELAP following a BDBEE. Because it is the subject of a rulemaking, Recommendation 4.1 is not discussed in this safety evaluation report (SER). Recommendation 4.2 is discussed below.

19.3.1.2 Summary of Application

DCD Tier 1: There is no Tier 1 information regarding mitigation strategies for beyond design basis external events based on NTTF Recommendation 4.2.

DCD Tier 2: The design basis and complete description for the mitigation strategies for beyond design basis external events can be found in DCD Tier 2, Revision 2, Section 19.3.2.3, and Technical Report APR1400-E-P-NR-14005, Revision 2, "Evaluations and Design Enhancements to Incorporate Lessons Learned from the Fukushima Dai-Ichi Nuclear Accident," Revision 0. The information contained in DCD Tier 2, Revision 2, Section 19.3.2.3 describes how the mitigation strategies for beyond design basis events address all the design criteria identified in NEI 12-06.

Inspections, Tests, Analyses and Acceptance Criteria (ITAAC): There are no ITAAC items for this area of review, because the mitigation strategies in this section are for beyond design basis events.

Initial Test Program: No specific initial test has been identified for the mitigation strategies for beyond design basis external events.

Technical Specifications (TS): No Technical Specifications are provided for the mitigation strategies for beyond design basis external events.

19.3.1.3 ***Regulatory Basis***

The requirements and guidance for the staff's review of beyond-design-basis external event mitigation strategies are as follows:

- Atomic Energy Act of 1954, as amended (the Act), Section 161, authorizes the Commission to regulate the possession and utilization of special nuclear material in a manner that is protective of public health and in accordance with the common defense and security.

- SRM-SECY-12-0025, "Staff Requirements – SECY-12-0025 – Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012, which approves the issuance of orders for beyond-design-basis external events as necessary for ensuring continued adequate protection under 10 CFR) 50.109(a)(4)(ii) exception to the Backfit Rule.

- Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012. Although Order EA-12-049 does not apply to the APR1400 DC, the staff has followed the current NRC and industry guidance for mitigation strategies in evaluating the equipment used as part of the FLEX mitigation strategy for the APR1400 reactor.

- The Japan Lesson-Learned Project Directorate, Interim Staff Guidance (ISG), JLD-ISG-2012-01, Revision 0, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," which accepts the methodology described in NEI 12-06 Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," with exceptions and clarifications.

- 10 CFR 52.47, “Contents of applications; technical information,” Item (a)(25), which requires that a DC application (DCA) contain interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the DCD.

19.3.1.4 Technical Evaluation

The staff reviewed the mitigation strategies described in DCD Tier 2, Revision 2, Section 19.3.2.3.

19.3.1.4.1 Evaluation of External Hazards

Sections 4 through 9 of NEI 12-06 provide an NRC-accepted methodology for determining the applicable extreme external hazards, in order to identify potential complicating factors for the protection and deployment of equipment needed to mitigate beyond-design-basis external events leading to an ELAP and a loss of normal access to ultimate heat sink (LUHS). Installed plant equipment and onsite portable equipment used for mitigation must be protected against the above external events. According to NEI 12-06, this equipment should be designed to be robust and housed in robust buildings. Robust is defined in NEI 12-06, Appendix A, as “the design of an SSC either meets the current plant design basis for the applicable external hazards or has been shown by analysis or test to meet or exceed the current plant design basis” with respect to seismic events, floods, and high winds and associated missiles.

The DCD Tier 2, Revision 2, Section 19.3.4, identifies COL information items, COL 19.3(1), COL 19.3(2), COL 19.3(3), and COL 19.3(4) to address the site-specific external hazards including seismic risk evaluation, flood requirements for wet sites, and storage location for FLEX equipment. These COL information items are described and reviewed in Section 19.3.1.5 of this SER, and COL 19.3(4) was reworded as a result of this review. The staff will review the site-specific external hazards in COL applications.

19.3.1.4.2 Phased Approach and Acceptance Criteria

The DCD Tier 2, Revision 2, Section 19.3.2.3, describes the proposed three-phase approach used for the APR1400, in order to conform to Order EA-12-049. DCD Tier 2, Revision 2, Section 19.3.2.3, summarizes the APR1400 phase approach for each of the plant operation modes:

<u>Three-Phase Approach</u>

Phase 1: Coping with installed plant equipment.

Phase 2: Coping with installed plant equipment and onsite portable (FLEX) equipment.

Phase 3: Coping with both onsite portable (FLEX) equipment and offsite resources in addition to installed equipment.

The staff noted that addressing the three-phase approach for all modes of operations is consistent with Order EA-12-049 and NEI-12-06. The staff’s detailed review of the proposed three-phase approach for all modes of operations is in Section 19.3.1.4.3 of this SER for core cooling, Section 19.3.1.4.4 for containment function, and Section 19.3.1.4.5 for spent fuel cooling.

The DCD Tier 2, Revision 2, Section 19.3.2.3, and the APR1400-E-P-NR-14005 describe the details of the proposed mitigation strategies. The applicant clarified that APR1400-E-P-NR-14005 is provided as incorporated by reference per DCD Tier 2 Table 1.6-2.

This clarification is documented in a letter (ML17242A314), dated August 30, 2017, in response to request for additional Information (RAI) 333-8397, Question 19.03-9, Part (d). The response to RAI 333-8397, Question 19.03-9, is reviewed in Section 19.3.1.4.7 of this SER, and was tracked as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 2 of the DCD. Therefore, the staff considers RAI 333-8397, Question 19.03-9, Part (d), to be resolved and closed.

The applicant provided the information in APR1400-E-P-NR-14005, Revision 2 to show the status of conformance to the NRC review guide and industry implementation guidance. APR1400-E-P-NR-14005, Revision 2, Table 5-7, "Conformance with JLD-ISG-2012-01, Revision 0," and Table 5-8, "Conformance with NEI 12-06, Revision 0," itemized APR1400 conformance to the guides. In these tables, the applicant identified that some items are a COL applicant's responsibility, some items are not applicable for APR1400, and all the remaining items conform to the above guidance documents.

The applicant performed analyses in APR1400-E-P-NR-14005, Revision 2, to demonstrate the capability of the proposed mitigation strategies for core and SFP cooling, and containment function. The proposed acceptance criteria for APR1400 are as follows:

- Core Cooling – Appendix A, Section A.3 of APR1400-E-P-NR-14005, Revision 2, states that the acceptance criteria for core cooling are (1) core cooling being maintained, and (2) no fuel failures.
- Spent Fuel Pool Cooling – Appendix B, Section B.1 of APR1400-E-P-NR-14005, Revision 2, states that the acceptance criterion for SFP cooling is that fuel in the SFP remains water covered.
- Containment Function – DCD Tier 2, Revision 2, Section 19.3.2.3.3 states that the containment pressure and temperature are maintained below the design limits, and containment structure integrity is maintained during the course of the event.

The staff reviewed the above criteria, and requested the following additional information in RAI 407-8447, Question 19.3-25 (ML16048A200).

NEI 12-06, Section 3.2.1.1 states that, for core cooling in a pressurized water reactor (PWR), the requirement is to keep the fuel in the reactor covered. In Table 5-9, Item 3.2.1.1 of APR1400-E-P-NR-14005, Revision 0, the applicant indicates that APR1400 complies with NEI guidance. However, the staff found that the applicant's criteria for core cooling, as stated in Appendix A of APR1400-E-P-NR-14005, Revision 0, are inconsistent with the NEI guidance regarding keeping the fuel covered. In the RAI, the applicant was requested to:

a) explain the inconsistency to the guidance and justify the deviation, and

b) confirm the acceptance criterion for the containment function.

The applicant responded to RAI 407-8447, Question 19.03-25, in a letter dated April 2, 2016 (ML16093A005), and revised its response in a letter dated July 1, 2016 (ML16183A222). In the response to Part (a) of Question 19.03-25, the applicant states that in APR1400-E-P-NR-14005, Revision 0, the criterion of "no fuel failure" is used instead of the NEI 12-06 criterion of "keeping the fuel in the reactor covered," because the core cooling capability is ensured by the fuel heat-up analysis, which directly calculates the fuel temperature. The acceptance criteria for mitigation strategies are documented in Section 19.3.2.3.1 of DCD Tier 2, and Table 5-8, item 3.2.1.1 of APR1400-E-P-NR-14005 would be revised to reflect the above justification for the general criteria.

The staff reviewed the applicant's response and found that the APR1400 mitigation analysis for core cooling includes Modes 1 through 6, as defined by the Technical Specifications Table 1.1-1 in DCD Chapter 16, which is more than what NEI 12-06, Revision 0, guidance covers. As indicated in Appendix A to APR1400-E-P-NR-14005, during Modes 1 through 4, the core cooling is achieved by using feed-and-bleed in the steam generator and natural circulation in the primary side of reactor coolant system (RCS). The APR1400 analysis in Appendix A to APR1400-E-P-NR-14005 shows that it is consistent with the NEI guidance of "keeping the fuel in the reactor being covered" for Modes 1 through 4. Based on above, the staff finds the response to Part (a) of Question 19.03-25 acceptable, because the response justified the inconsistency in question regarding the acceptance criteria for core cooling at full power and documented the proposed acceptance criteria in the DCD. A more detailed review of the core cooling at full power is in Section 19.3.1.4.3, "Core Cooling," of this SER. A more detailed review of the core cooling strategies for shutdown and refueling is in Section 19.3.1.4.9, "Shutdown and Refueling Analyses," of this SER.

In the response to RAI 401-8402, Question 19.03-16, the applicant clarifies that the acceptance criteria for the containment function is the containment design limits for pressure and temperature. The acceptance criteria regarding containment integrity are documented in Section 19.3.2.3.3 of DCD Tier 2 Revision 2 and Section 5.1.2.5.2 of Technical Report APR1400-E-P-NR-14005, Revision 2. A more detailed review of containment integrity is in Section 19.3.1.4.4, "Containment Function Strategies," of this SER.

The staff found the response to RAI 407-8447, Question 19.03-25, acceptable because documenting and clarifying the acceptance criteria in the DCD and justifying the inconsistency in acceptance criteria for core cooling adequately addressed the staff's request in the RAI. Therefore, RAI 407-8447, Question 19.03-25, was resolved and the DCD markup associated with the response to Question 19.03-25 was being tracked as a confirmatory item. The staff confirmed that the DCD markups associated with the RAI response have been incorporated into Revision 1 of the DCD. Therefore, the staff considered RAI 407-8447, Question 19.03-25, to be resolved and closed.

19.3.1.4.3 Core Cooling

The NRC Order EA-12-049 states that the mitigation strategies must ensure the reactor core is adequately cooled following a simultaneous ELAP and LUHS. During an ELAP event, the normal reactor core cooling function using reactor coolant pumps (RCP) of the RCS is lost due to ac power loss while the decay heat from the reactor continues to heat the core. The applicant describes the mitigation strategies for core cooling in DCD Tier 2 Section 19.3.2.3.1. The staff's evaluation of the proposed mitigation strategies follows.

The DCD Tier 2 Section 19.3.2.3.1 states the following:

The APR1400 core cooling capability to cope with the BDBEE, ELAP concurrent with LUHS, is addressed for all of the following operation modes:

a. Full-power operation
b. Low-power operations and shutdown conditions with steam generators (SGs) available
c. Shutdown conditions with SGs not available

The applicant performed supporting analysis to demonstrate the APR1400 baseline coping capability based on both of the FLEX strategies.

The DCD Tier 2, Section 19.3.2.3, summarizes the APR1400 phased approach for each of the plant operation modes:

> The full-power operation case was selected by the applicant as a representative one for the operational strategy for Modes 1 through 5 with SGs available. Phase 1 is between 0 to 8 hours; Phase 2 is between 8 to 72 hours; Phase 3 is for an indefinite time period following Phase 2.
>
> The mid-loop operation case was selected by the applicant as a representative one for the shutdown operations with SGs not available. This analysis is described in Section 19.3.1.4.9 of this SER.

The applicant described that during full-power operation, Phase 1 (0 to 8 hours), only the installed plant equipment is used for coping. Specifically, two turbine-driven auxiliary feedwater pumps (TDAFWP) automatically start on an auxiliary feedwater actuation signal (AFAS) to provide core cooling through the SGs. Auxiliary feedwater storage tanks (AFWST) are used to supply water to the TDAFWPs, and steam generated in the SGs is released through the main steam safety valves (MSSVs). Class 1E batteries supply direct current (dc) power to essential instrumentation and control (I&C) equipment for the TDAFWPs. Therefore, the RCS is maintained in a hot standby condition by the natural circulation cooling (NCC) operation without any operator action during this phase. The RCP seal leakage is assumed to be 94.64 L/min (25 gpm) per RCP at full-power steady-state condition from the beginning of the event.

In the full-power operation case, the RCS is cooled down to and maintained at hot shutdown (about 176.67 °C (350 °F)) using installed plant equipment such as TDAFWPs, auxiliary charging pump (ACP), as well as the FLEX equipment, such as 480 V mobile GTG. The RCS is cooled down to the hot shutdown condition by feed and steaming operations through the secondary side of the SG using the TDAFWPs and main steam atmospheric dump valves (MSADVs).

The AFWSTs, using the RWT as a backup water source, continue to supply water to the SGs using the TDAFWPs, while each SG level is maintained between 25 to 40 percent wide range by on-off control of the auxiliary feedwater isolation valves. The ACP is used to provide makeup water for maintaining RCS inventory and provide RCP seal cooling. The suction source for ACP is the boric acid storage tanks (BAST) and in-containment refueling water storage tank (IRWST).

Two 480 V, 1,000 kW mobile GTGs are provided to meet N+1 guidance, as described in NEI 12-06. One of the 480 V mobile GTGs is connected to the 480 V Class 1E power system train A or B, and supplies power to the 125 Vdc battery charger, the 480 V load center, and the motor control center.

In case the installed TDAFWPs are inoperable even after connection of a 480 V mobile GTG, the RCS is cooled down to approximately 98.89 °C (210 °F) with SGs fed by the secondary side FLEX pumps instead of the plant installed TDAFWPs. RCS makeup is carried out by the primary side high-head FLEX pump instead of the ACP. A primary high-head FLEX pump is connected to a safety injection (SI) line to be used as an alternative RCS makeup pump, when the ACP is unavailable.

Two secondary FLEX pumps are also connected to the SG auxiliary feedwater (AFW) supply lines: one for each AFW line. The secondary FLEX pumps can be used to supply feedwater to the SGs when the TDAFWPs are unavailable. Two primary high-head FLEX pumps and three secondary FLEX pumps are provided to meet the N+1 guidance.

In Phase 3, offsite resources, including a 4.16 kV mobile generator, fuel, and cooling water can be assumed to be available for long-term coping with the BDBEE. The 4.16 kV mobile generators will be used to restore train A or B of the 4.16 kV Class 1E power system. The plant will be brought to cold shutdown, using the shutdown cooling system (SCS), if the ultimate heat sink (UHS) is available after 4.16 kV Class 1E power is restored. If not, the plant is maintained at the same safe shutdown state as in Phase 2. In this phase, the primary and secondary makeup water sources and fuel oil for the mobile 480V mobile GTGs and the 4.16 kV mobile generator will be refilled from offsite resources.

Initial Plant Assumptions and Conditions

The staff confirmed that the applicant's initial plant assumptions and conditions as identified in Technical Report APR1400-E-P-NR-14005, Table 5.9, are consistent with NEI 12-06, Section 3.2.1, and are as summarized below:

(1) Reactor operating at 100 percent power for at least 100 days.

(2) Reactor and supporting systems parameters are within normal operating ranges.

(3) The loss of offsite power (LOOP) is assumed to affect all units at a plant site.

(4) All installed sources of emergency on-site ac power and station blackout (SBO) Alternate ac power sources are assumed to be not available and not imminently recoverable.

(5) Cooling and makeup water inventories are assumed available if contained in systems or structures designed to withstand seismic events, floods, and high winds, and associated missiles.

(6) Normal access to the ultimate heat sink is lost, but the water inventory in the UHS remains available and seismic designed piping connecting the UHS to plant systems remains intact. The motive force for UHS flow is lost.

(7) Permanent plant equipment is available if located in structures designed to withstand seismic events, floods, and high winds, and associated missiles.

(8) No additional events or failures are assumed to occur immediately prior to or during the event, including security events.

(9) Reliance on the fire protection system ring header as a water source is acceptable only if properly protected.

Phase 1: 0 to 8 Hours, Basic Operational Strategy

The initiating event is the LOOP that results in the complete loss of forced reactor coolant flow from the simultaneous loss of alternating current (ac) electrical power to all reactor coolant pumps. While the RCS flow is coasting down, a total SBO is assumed to occur with complete loss of the emergency ac power to the Class 1E and non-Class 1E switchgear buses. However, the ac power to the buses fed by station batteries through inverters is available. Within one minute of the loss of ac power, the normal feedwater flow is lost, the turbine trips, the reactor coolant pumps continue to coast down toward RCS natural flow circulation which results in a core protection calculator (CPC) generated reactor trip, the pressurizer heaters are lost, and the UHS is lost due to the loss of heat removal capability of the essential service water system and component cooling water system.

As described in APR1400-E-P-NR-14005, the resulting reactor conditions include an increase in core average coolant temperature and system pressure, and a decrease in margin to departure from nucleate boiling. On the primary side, from APR1400-E-P-NR-14005, Figure A-2, the initial RCS hot leg temperature of approximately 590˚F increases to about 615˚F within a few minutes, and then decreases to approximately 590˚F after 15 minutes and continues to decrease to 570˚F during the Phase 1 eight hour period. From Figure A-1 in the technical report, the RCS pressure is initially 2250 psia and increases to about 2270 psia before decreasing to approximately 1350 psia over a 2-hour period. The reactor pressure continues to decrease to about 1290 psia during the next 4 hours before decreasing, at a more rapid rate, to 1210 psia in the final 2 hour period of Phase 1. The initial increase in reactor pressure and temperature is due to the loss of forced reactor core flow and the reduction in heat removal by the steam generators from the loss of SG inventory and main feedwater makeup. After approximately 15 minutes into the event, the heat removal capability between the primary and secondary side improves with the automatic start of TDAFWP upon the receipt of the auxiliary feedwater actuation system (AFAS) signal from the engineered safety features actuation system (ESFAS). Thus, the reactor core heat removal cycle on the primary side is accomplished by natural circulation of the RCS coolant through the core and then transferred to the secondary side through the SG. In the SG feedwater makeup is converted to steam and released through the MSSVs. During steam generation, the opening of the MSSVs occur when the secondary side pressure reaches the MSSV setpoint pressures of 1,174 and 1,205 psig, respectively. Therefore, in Phase 1 (0-8 hours) no operator action is require to remove heat and maintain the reactor in Hot Standby (Mode 3). During this phase, the auxiliary feedwater system (AFW) supplied approximately 132,086 gallons of AFW storage tank inventory (750,000 gallons) to the steam generator. On the primary side, the pilot-operated safety relief valves (POSRV) did not open since the maximum reactor system pressure reached during the transient was below the valve setpoint pressure of 2,470 psia.

The above results are based on the best-estimate of the initial plant, reactor, and event conditions as discussed in NEI 12-06 Section 3.2.1, "General Criteria and Baseline Assumptions." The model assumes the reactor is operating at 100 percent power with LOOP as the initiating event. All installed sources of emergency on-site ac power and SBO alternate ac power sources are unavailable and normal access to the ultimate heat sink is lost. Plant cooling and makeup water inventories are available. The DCD Tier 2 system design parameters at 100 percent power were used in the applicant's RELAP5/Mod 3.3 analysis to confirm the core cooling capability to cope with the BDBEE, ELAP, and LUHS were in agreement with the FLEX strategies. The staff finds the APR1400 Phase 1 coping strategy acceptable because it complies with NEI 12-06 guidance. Furthermore, the staff finds that the applicant's strategy to maintain or restore core cooling and RCS inventory during a full power ELAP event is consistent with NEI 12-06 guidance.

Phase 2: 8 to 72 Hours, Basic Operational Strategy

During Phase 1 the operator prepares the plant for Phase 2 entry by confirming that both the installed plant equipment, such as TDAFWP and the ACP, along with the FLEX equipment, such as the 480 V mobile GTG, are operable. When the Phase 2 preparations are completed, the operator starts the reactor cool down from hot standby to hot shutdown or cold shutdown, using the installed plant equipment and/or the onsite FLEX equipment.

During the initial start of this phase, one of the 480 V mobile GTGs is connected to the 480 V Class 1E power system train to provide power to ACP, MSADVs, and other essential equipment to allow the RCS to be cooled down to the hot shutdown condition through feed-and-bleed operation on the secondary side of SG using the TDAFWPs and MSADVs. Feedwater makeup inventory continues from the AFWSTs with backup water source from the RWT. The SG level is maintained between 25 to 40 percent by operator action of controlling the auxiliary feedwater isolation valves. In addition, the ACP is operated to provide makeup water for maintaining RCS inventory and provide RCP seal cooling. The boric acid storage tanks (BASTs) and In-containment Refueling Water Storage Tank provide the makeup inventory to the ACP. The water inventory required for RCS makeup during Phase 2 is approximately 170,000 gal. In addition to the ACP makeup, the SIT provides borated water when the RCS pressure reaches the SIT actuation setpoint. The RCS makeup from these sources is sufficient to overcome the decrease in level due to RCS pump seal leakage and RCS inventory contraction due to cooling. Therefore, the RCS level is maintained above the top of active core as shown in Figure A-7 of the technical report. Also, the cladding temperature of the active core does not exceed the fuel design limit as shown in Figure A-8 of the technical report. On the secondary side, Figure A-9 of the technical report shows that the MSADV flow is sufficient in heat removal during Phase 2. Thus, the staff finds the Phase 2 acceptable because the applicant's approach described above is consistent with the guidance of NEI 12-06.

Phase 3: Indefinite Time Period After 72 Hours

In Phase 2, operator actions to prepare Phase 3 will be finished within 72 hours following the event. The offsite resources include a mobile Generator, fuel, and cooling water for long-term coping with the BDBEE. The plant is brought to cold shutdown, using the SCS if the UHS is available after 4.16 kV Class 1E power is restored. Shutdown cooling is discussed in Chapter 5 of this report. If the SCS is not available, the plant is maintained at the same safe shutdown state as in Phase 2.

Contingency Plan: Core Cooling – Full-Power Operation

The applicant provided a full-power operation contingency plan using the FLEX equipment. In the contingency plan strategy, installed plant equipment is assumed to be inoperable. Therefore, on the secondary side of the SG, two FLEX pumps are used in place of the plant installed TDAFWPs to reduce the RCS temperature to approximately 230 °F. On the primary side, RCS makeup is provided by a high-head FLEX pump in place of an ACP.

Figure A-11 and Figure A-12 show the primary side and secondary side pressures and the RCS temperatures, respectively. These figures show the coping operation of each step of natural circulation cooling operation in maintaining the hot standby condition, cooling down the RCS to hot shutdown condition, further cooldown to the SG pressure of 14.7 psia, and maintaining the RCS at cold shutdown. The computer model demonstrated that the contingency plan operation results are consistent with the basic operational strategy.

The applicant's strategy to ensure conformance with staff guidance for conditions where the unit is shut down or being refueled is reviewed separately in Section 19.3.1.4.9 of this evaluation.

The staff finds that the applicant's approach described above is consistent with the guidance of NEI 12-06 and provides reasonable assurance that the plant can be brought to a cold shutdown condition and maintained at this condition for an indefinite period.

19.3.1.4.4 Containment Function Strategies

19.3.1.4.4.1 Containment

The industry guidance document, NEI 12-06, Table 3-2, provides some examples of acceptable approaches for demonstrating the baseline capability of the containment strategies to maintain containment functions during all phases of an ELAP event. One such approach is to perform an analysis demonstrating that containment pressure control is not challenged.

In accordance with NEI 12-06, the applicant performed a containment evaluation (APR1400-E-P-NR-14005), which was based on the boundary conditions (e.g., full power operation) described in Section 2 of NEI 12-06. The calculation which analyzed this strategy demonstrates that the containment parameters of pressure and temperature remain well below the respective DCD Tier 2 Table 6.2.1-3, "Principal Containment Design Parameters," design values of 4.22 kg/cm2 (60 psig) and 143 ºC (290 ºF) for more than 7 days following full-power operation.

From its review of the evaluation, the staff noted that the required actions to maintain containment function following full power operation have been developed, and are summarized below.

Phase 1 (0 to 8 hours)

The applicant's analysis shows there are no Phase 1 actions.

Phase 2 (8 to 72 hours)

The applicant's analysis shows there are no Phase 2 actions.

Phase 3 (beyond 72 hours)

The applicant's analysis shows that with no actions, containment parameters remain far below their design limits for more than 7 days and approach design limits after 16 days. Through the use of the emergency containment spray backup system (ECSBS), the applicant's analysis shows that the containment pressure and temperature can be maintained well below design limits. The applicant's analysis demonstrates that the containment response following a postulated ELAP event does not challenge design limits until well after the availability of offsite equipment and implementation of long term strategies to control pressure and temperature. After 72 hours, offsite resources can be assumed available for long-term coping. Therefore, the containment function is maintained following full-power events through all phases. The long-term coping strategy is the responsibility of the COL applicant and is addressed by COL items.

Staff Evaluations

The APR1400 containment is a pre-stressed concrete containment with a steel liner. During a BDBEE while at full power operation that results in an ELAP/LUHS, the applicant assumes the RCP seals fail due to loss of cooling. During normal operation, RCP seals limit the leakage of reactor coolant along the pump shaft and thereby into the containment. These seals, which form part of the reactor coolant pressure boundary, require cooling during normal operation, even while the reactor is in hot standby or hot shutdown. Without such cooling, these seals are susceptible to increased leakage once temperatures exceed design limits. During a BDBEE from full power operation, the applicant assumes RCP seal leakage rate of 94.64 L/min (25 gpm) per RCP into containment for a total of 378.5 L/min (100 gpm) for four RCPs. The applicant's evaluation indicates that the containment parameters of pressure and temperature remain well below the respective DCD Tier 2 Table 6.2.1-3, “Principal Containment Design Parameters,” design values of 4.22 kg/cm2 (60 psig) and 143 ºC (290 ºF) for greater that 7 days following full-power operation.

The NEI 12-06, Section 3.2.1.5, “Reactor Coolant Inventory Loss,” identifies that normal system leakage is a source of expected reactor coolant inventory loss. APR1400-E-P-NR-14005, Table 5-9, “Conformance with NEI 12-06, Revision 0,” indicates conformance with NEI 12-06 Section 3.2.1.5. However, this technical report does not identify if normal reactor coolant inventory loss (Technical Specifications may permit up to 10 or 11 gpm), in addition to RCP seal leakage, is considered as contributing to the mass and energy input into the containment. In RAI 401-8402, Question 19.03-14 (ML16039A017), the staff requested that the applicant assess all potential sources into the containment to include normal system leakage and evaluate the impact on containment capabilities. In response to RAI 401-8402, Question 19.03-14, dated May 20, 2016 (ML16142A056), the applicant clarified that the assumed RCP seal leakage (100 gpm total) for containment analysis was very conservative and bounding. The tested RCP seal leakage is expected to be under one-tenth gpm per pump (staff evaluation of RCP seal performance is found in Section 19.1.4 of this safety evaluation), which is much less than the assumption of 25 gpm per RCP. Because the assumed mass release from the reactor coolant system into the containment conservatively bounds the expected RCP seal leakage plus normal reactor coolant inventory loss, the staff finds the assumed RCP seal leakage used for containment analysis to be acceptable. Therefore, given the discussion above, the staff finds that the applicant’s response to RAI 401-8402, Question 19.03-14, is acceptable and is closed.

The staff conducted an audit of Calculation 1-310-N380-008, Revision 0, “Containment Integrity Analysis Following RCP Seal Failure and Loss of RHR” (see ML17037A756 for audit summary, ML15169A255 for audit plan). The basic assumptions with regard to the containment model such as nodalization, passive heat sinks, and heat transfer coefficients are the same as those used in the containment peak pressure calculation for the design basis LOCA. The calculation discusses a single heat source into containment (corresponding to a mass source) from reactor coolant leakage flow (from reactor coolant pump seals). In RAI 401-8402, Question 19.03-15 (ML16039A017), the staff requested that the applicant describe how sensible heat transfer (e.g., heat removed from a hot object to its surroundings) from the reactor coolant system into containment was evaluated. In response to RAI 401-8402, Question 19.03-15, dated June 30, 2016 (ML16182A557), the applicant stated that the containment analyses were re-performed to account for the contribution due to sensible heat. The sensible heat loss used in the analysis was based on heat loss measured from the first APR1400 plant (i.e., Shin-Kori Unit 3). Because the applicant’s re-analysis accounts for sensible heat, taking into account operating experience, the staff finds the applicant’s response acceptable. Therefore, based on the above discussion, the staff finds that the applicant’s response to RAI 401-8402, Question 19.03-15, is acceptable and is closed.

The APR1400 DCD Tier 2 Section 19.3.2.3.3 describes use of the ultimate pressure capacity (UPC) as the acceptance criterion for assessing the APR1400 containment capabilities during a beyond-design-basis external event that results in an ELAP. In RAI 401-8402, Question 19.03-16 (ML16039A017), the staff requested that the applicant provide justification for the selected containment pressure acceptance criterion (i.e., UPC). In response to RAI 401-8402, Question 19.03-16, dated December 7, 2016 (ML16342C726), the applicant changed the containment pressure acceptance criterion. Instead of UPC, the applicant now uses containment design pressure (60 psig (4.2 kg/cm2)). In addition, the applicant specified that temperature would be maintained below the design limits. The applicant reflects these changes in markups to DCD Tier 2 Section 19.3.2.3.3. Because the staff considers it appropriate to implement strategies to control containment pressure and temperature before exceeding containment design limits (acceptance criteria used by U.S. operators) and well before approaching the applicant's initial proposed UPC acceptance criterion, the staff find the response to RAI 401-8402, Question 19.03-16, acceptable. The staff confirmed that the RAI response markup has been incorporated into Revision 2 of the DCD. Therefore the staff considers the response to RAI 401-8402, Question 19.03-16, to be resolved and closed.

APR1400-E-P-NR-14005 does not contain simplified drawings to show how the FLEX strategy, using the ECSBS, is used to maintain containment capabilities. In RAI 401-8402, Question 19.03-18 (ML16039A017), the staff requested that the applicant provide a simplified drawing(s) that identifies the flow path to deliver water to containment. In response to RAI 401-8402, Question 19.03-18, dated May 20, 2016 (ML16142A056), the applicant provided a simplified drawing, as a mark-up to the Technical Report (APR1400-E-P-NR-14005). Because the simplified drawing depicts the FLEX strategy for adding water to containment using the ECSBS system, the staff finds the applicant's response is acceptable. The staff confirmed that the RAI response markup has been incorporated into Revision 1 of the Technical Report. Therefore, the staff considers RAI 401-8402, Question 19.03-18, to be resolved and closed (note, the Technical Report is incorporated by reference into the DCD, see response to RAI 333-8397, Question 19.03-9).

APR1400-E-P-NR-14005 does not describe the connection points (for FLEX equipment) necessary to maintain the containment capabilities. NEI 12-06 Section 3.2.1.3, "Initial Conditions," indicates that permanent plant equipment that is contained in structures with designs that are robust with respect to seismic events, floods, and high winds, and associated missiles, are available. The report should provide connection design information that justifies that a connection is robust and remains available to address a beyond-design-basis external event. In RAI 401-8402, Question 19.03-19 (ML16039A017), the staff requested that the applicant provide information in the Technical Report that describes the connection design and connection quality classification used to maintain the containment capabilities and provide a basis for assuming that the connections will be available. In response to RAI 401-8402, Question 19.03-19, dated June 30, 2016 (ML16182A557), the applicant stated that the connection line is designed to Seismic Category I and provided the quality classification. The staff confirmed that the quality classification for this connection was consistent with regulatory guidance contained in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." In addition, the staff confirmed that the same quality classification information is provided in DCD Tier 2 Figure 6.2.2-1, "Containment Spray System Flow Diagram." With regard to the basis for assuming the connections will be available, the applicant directed the staff to review the applicant's response to RAI 407-8447, Question 19.03-26. In the response to RAI 407-8447, Question 19.03-26, dated June 28, 2016 (ML16180A293), the applicant indicated that the protection of connections from applicable hazards is the responsibility of the COL applicant. The applicant provided mark-ups to the technical report and DCD that included a new Table and a new COL item. The Table lists all the connections for the FLEX strategies, and the COL item ensures that all the connections for FLEX strategies located outside of buildings (e.g., connection for ECSBS) are accessible and protected from all applicable hazards (e.g., seismic, floods, high winds, and associated missiles). The COL item refers to the Table for the list of connections. Because the applicant provided the quality classification information consistent with staff guidance and established a COL item to address the protection of connections, the staff finds the response acceptable. The staff confirmed that the RAI response has been incorporated into Revision 1 of the Technical Report (e.g., Table 6-5) and Revision 1 of the DCD (i.e., COL 19.3(8)). Therefore, with respect to the containment strategy, the staff considers the responses to RAI 401-8402, Question 19.03-19 and RAI 407-8447, Question 19.03-26, to be resolved and closed.

The NEI 12-06 Section 3.2.1.9, "Personal Accessibility," states that areas requiring personnel access should be evaluated to ensure the conditions will support the actions required by the plant-specific strategy for responding to the event. APR1400-E-P-NR-14005, Table 5-9, "Conformance with NEI 12-06, Rev. 0," indicates conformance to NEI 12-06, Section 3.2.1.9. The technical report does not describe plant areas requiring personnel access and the actions needed to maintain or restore the containment capabilities. In RAI 401-8402, Question 19.03-20 (ML16039A017), the staff requests that the applicant provide a listing of the areas requiring personnel access, the actions required (e.g., opening a valve, making a connection), and an evaluation of those areas to ensure the conditions will support the actions required by the plant-specific strategy for responding to the event. In a response to RAI 401-8402, Question 19.03-20, dated December 2, 2016 (ML16337A397), the applicant provided a listing of the pathways and associated areas requiring personnel access to maintain containment capabilities. One pathway (i.e., a combination of stairs, corridors, and general access areas) to access the reach rod for ECSBS operation is located within the auxiliary building, which is designed to seismic category I. The applicant predicts dose levels in the pathway consistent normal radiation zones. The applicant expects the temperature rise in the pathway to be insignificant as the pathway is located away from the containment (no direct contact with the containment). In addition, the applicant provided the specific tasks required to initiate ECSBS as a markup to the technical report. Furthermore, the applicant provided a markup to DCD Figure 6.2.2-1, "Containment Spray System Flow Diagram," to identify that ECSBS isolation valve (CS-V1013) is operated by a reach rod. Because the applicant provides an accessible pathway (e.g., auxiliary building) and required tasks to maintain or restore containment capabilities, the staff finds the response acceptable. The staff confirmed that the RAI response markup has been incorporated into Revision 1 of the technical report and Revision 1 of the DCD. Therefore the staff considers the response to RAI 401-8402, Question 19.03-20, to be resolved and closed.

The staff finds that the applicant's approach described above is consistent with the guidance found in NEI 12-06, as endorsed by JLD-ISG-2012-01, and provides reasonable assurance that the requirements of Order EA-12-049 can be met by a licensee with respect to maintaining containment.

19.3.1.4.4.2 Ventilation

The staff reviewed ventilation related strategies outlined in DCD Tier 2 Section 9.4.1 and APR1400-E-P-NR-14005. The review guidelines, outlined in NEI 12-06, Section 3.2.1.8, "Effects of Loss of Ventilation," are to verify that the effects of loss of heating, ventilation, and air conditioning (HVAC) in an extended loss of ac power event can be addressed consistent with NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," or by plant-specific thermal hydraulic calculations. NUMARC 87-00, Section 2.7, "Effects of Loss of Ventilation," discusses technical bases on equipment operability outside containment and control room habitability.

According to APR1400 DCD, Tier 2 Section 9.4.1.1, "Design Bases," during an SBO, the safety-related equipment of the control room HVAC System is powered from the alternate AC (AAC) source. The Control Room HVAC System is unavailable for 10 minutes until the AAC generator restores power after an SBO occurs. However, all safety-related electrical and I&C equipment in the main control room (MCR) are designed to keep their integrity during loss of the control room HVAC system.

According to DCD Tier 2 Section 19.3.2.3, the APR1400 FLEX strategy follows a three-phase approach as required in the Order EA-12-049. See Section 9.4.3.1.4.2 of this safety report for a detailed description of these three phases.

The APR1400-E-P-NR-14005, Section 5.1.2.3.1.2.1, states that during Phase 2 of full-power operation, 8 to 72 hours, additional cooling in the MCR, electrical and I&C equipment rooms, TDAFWP rooms, and ACP room is not required based on heat-up calculations. RAI 406-8427, Question 19.03-24 (ML16048A195), was issued to request clarification on the acceptance criteria to satisfy equipment qualification and control room operator habitability.

In response to RAI 406-8427, Question 19.03-24, dated March 15, 2016, (ML16075A418) the applicant states:

> The purposes of the room heatup calculations are to (1) determine the maximum temperatures in the main control room (MCR), technical support center (TSC), essential electrical and I&C equipment rooms housing post-Fukushima event mitigation equipment, turbine-driven auxiliary feedwater pump (TDAFWP) rooms, and auxiliary charging pump (ACP) room, and (2) demonstrate that the maximum temperature of the each room does not exceed the maximum allowable temperature during 72 hours assuming a loss of HVAC system.

The applicant indicates that the maximum allowable temperatures for the ACP Room, TDAFWP Rooms, and essential electrical and I&C equipment rooms housing post-Fukushima event mitigation equipment are 150 F, 150 F, and 120 F, respectively.

The applicant also indicates that the maximum allowable temperature for both the MCR and technical support center (TSC) is 110 ˚F, which follows the habitability requirement specified in NUMARC 87-00.

The staff reviewed KHNP heatup calculations. As shown below, in Table 19.3-1, the calculated maximum allowable temperatures of all rooms housing post-Fukushima event mitigation equipment satisfy NUMARC 87-00 temperature conditions.

Table 19.3-1

Room	Maximum Allowable Temperature	NUMAC-87-00 Condition and Temperature Limit
MCR	110 ˚F	110˚F for habitability
TSC	110 ˚F	110˚F for habitability
Essential Electrical and I/C Rooms in Auxiliary BLDG, Compound BLDG, & EDG BLDG	120 ˚F	Condition 1 – 120˚F
Auxiliary Charging Pump Room	150 ˚F	Condition 2 – 150 ˚F
Turbine Driven Auxiliary Feedwater Pump Rooms	150 ˚F	Condition 2 – 150 ˚F

NUMARC 87-00 was developed to address 4-hour station blackout at light water reactors. NUMARC 87-00 Condition 1 applies to rooms of low concern with respect to elevated temperature effects. NUMARC 87-00, Condition 2 applies to rooms generally require no forced cooling. According to KHNP's "Room Heatup Calculation for Main Control Room," 1-601-M370-001, the calculated maximum temperatures do not degrade equipment performance.

The staff noticed that "Section 2.7.2(3) Control Room Habitability, NUMARC 87-00 Rev. 1," refers to habitability conditions in ASHRAE (American Society of Heating, Refrigerating, and Air-Conditioning Engineers) Handbook: 1985 Fundamentals (ASHRAE 1985), which concludes that light work at temperature above 110 ˚F and relative humidity above 50 percent would be intolerable. The present APR1400 calculation, "Room Heatup Calculation for Main Control Room, 1-601-M370-001, Rev. 1," submitted in response to RAI 406-8427, Question 19.03-24, shows that the maximum temperature of the MCR and TSC is below 110 ˚F; however, the RAI response and calculation do not demonstrate consideration of relative humidity. Therefore, as a follow-up, RAI 516-8646, Question 19.03-40 (ML16232A635), was issued to request the applicant to justify how the relative humidity component of ASHRAE 1985 is satisfied in concert with the temperature requirement. The staff considers RAI 406-8427, Question 19.03-24, resolved and closed.

In response to RAI 516-8646, Question 19.03-40 dated September 9, 2016 (ML16253A265), the applicant states that:

> Before Beyond Design Basis External Events (BDBEE), it is assumed that the main control room (MCR) is at 77°F and 50% relative humidity, and technical support center (TSC) is at 80°F and 50% relative humidity. Although the temperature increases due to loss of HVAC, the relative humidity decreases.
>
> The humidity ratio in the MCR is 69.5 grains/lb at the initial condition, 77°F and 50% relative humidity. The humidity ratio in the TSC is 76.8 grains/lb at the initial condition, 80°F and 50% relative humidity. The maximum temperatures during 72 hours under loss of HVAC system are 109.8°F for MCR and 107.6°F for TSC.
>
> Since HVAC equipment is assumed to be failed, there will be no airflow from and out of MCR and TSC, thus there will be few factors in increasing absolute humidity, or humidity ratio. In the case of MCR, the relative humidity both at the humidity ratio of the initial condition (69.5 grains/lb) and at the maximum temperature (109.8°F) is 18.1%. In the case of TSC, the relative humidity both at the humidity ratio of the initial condition (76.8 grains/lb) and at the maximum temperature (107.6°F) is 21.3%. Thus it meets what NUMARC 87-00 requires as criteria of relative humidity, below 50%.

The applicant calculated the final relative humidity in the MCR (18.1 percent) and TSC (21.3 percent) 72 hours after loss of HVAC system assuming no water vapor added from the occupants in the MCR. The staff performed a calculation by adding water vapor from five operators in the MCR, using latent heat 145 watts per person (EPRI NP-4453-L, Revision 1). The calculation results showed that the final relative humidity would increase roughly 2 percent in the MCR and TSC. These relative humidity values are substantially below the 50 percent NUMARC 87-00 criteria. Therefore, the staff considers RAI 513-8446, Question 19.03-40, resolved and closed.

The staff finds that the applicant's approach described above is consistent with the guidance found in NEI 12-06, as endorsed by JLD-ISG-2012-01, and provides reasonable assurance that the requirements of Order EA-12-049 can be met by a licensee with respect to maintaining area ventilation capability.

19.3.1.4.5 Spent Fuel Pool Cooling

The NRC Order EA-12-049 states that the mitigation strategies must ensure the SFP being adequately cooled following a simultaneous ELAP and LUHS. Moreover, NEI 12-06, Section 3.2.1.1 states that the requirement for the SFP cooling is to keep the fuel in the spent fuel pool covered. In Appendix B, Section B.1 of APR1400-E-P-NR-14005, Revision 1, the applicant states that the APR1400 acceptance criterion for SFP cooling is that fuel in the SFP remains covered. Therefore, the staff finds that the proposed criterion for SFP mitigation strategies is consistent with the NEI guidance and NRC Order.

The DCD Tier 2, Revision 2, Section 19.3.2.3.2, and APR1400-E-P-NR-14005, Revision 2 describe the details of the mitigation strategies, design features, and analyses to demonstrate the acceptance criteria being satisfied. During an ELAP event, the spent fuel pool cooling system (SFPCS) heat exchanger is lost due to ac power loss while the decay heat from spent fuel continues to heat the pool. The following mitigation strategies to maintain SFP cooling are described in DCD Tier 2, Revision 2, Section 19.3.2.3.2, based on the most limiting plant condition, i.e., Mode 6 with full core offload.

a) The operators have approximately 29.6 hours to restore cooling and/or makeup to the SFP in order to keep the spent fuel covered. Therefore, boiling of the SFP can be credited as the Phase 1 event mitigation method.
b) To maintain at least 3.05 m (10 ft) of water inventory over the fuel assemblies, makeup water to the SFP is provided within 15.36 hours.
c) For Phase 2 and 3 of event mitigation, an SFP makeup rate of 493.2 L/m (130.3 gpm) is needed to match the boiloff rate. The boil-off rate decreases over time as the spent fuel decay heat decreases.

In Appendix B of APR1400-E-P-NR-14005, Revision 2, the applicant performed an analysis of the SFP decay heat removal to obtain the parameters in the above strategies. The results of the analysis are shown in APR1400-E-P-NR-14005, Revision 2, Tables B-1, “Time to Reach SFP Bulk Boiling and Input Parameters,” Table B-2, “Time to Reach SFP Water Level 2 and Level 3,” and Table B-3, “Required Makeup Volume and Water Source.”

The DCD Tier 2, Revision 2, Section 19.3.2.3.2, states that, during Phase 1 and prior to the onset of boiling, action is taken to establish a vent path for the steam generated in the SFP by opening the rollup door to the fuel handling area of the auxiliary building. Based on the analysis, SFP boiling is calculated no sooner than 2.0 hours after the ELAP event occurs. In Phases 2 and 3, one SFP makeup FLEX pump, one SFP spray FLEX pump and an alternate FLEX pump are used to makeup SFP water and maintain the water level at least 3.05 m (10 ft) above the fuel assemblies. The level instrument, as described in DCD Tier 2, Revision 2, Section 9.1.3.5, is safety-related. DCD Tier 2 Section 19.3.2.4 provides more details for the reliable spent fuel pool instrumentation. APR1400-E-P-NR-14005, Revision 2, Section 5.1.2.4.1.2 states that FLEX pumps are provided to meet the N+1 guidance for a single-unit site and will meet 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” GDC 2, “Design Bases for Protection Against Natural Phenomena.” The N+1 guidance is stated in NEI 12-06 guidance. The RWT can be used as the water source for SFP makeup and spray. Flexible hoses, fuel for FLEX pump(s), and other equipment required for the mitigation strategies are located away from the auxiliary building, and mobilization of the equipment for SFP makeup capability can occur within the most limiting case within 15.36 hours. The FLEX pump discharge hose is routed to one of the two permanent SFP makeup connections located outside the east wall of the auxiliary building. Figure 6-3 of APR1400-E-P-NR-14005, Revision 2 shows the connection for SFP makeup and spray line. Figure 6-4 of APR1400-E-P-NR-14005, Revision 2 shows the layout of SFP makeup and SFP spray line connections. The alternate connection is close to SFP makeup connections. The FLEX pump connections are each connected to an independent, seismically qualified standpipe that runs inside the auxiliary building from the pump staging areas. Operators are able to connect flexible hoses from FLEX pump to the standpipes. The standpipes for SFP makeup have hard-piped connections to the SFP edge to allow water makeup to the pool. The specific storage location, mobilization, and other details for the FLEX equipment are COL items COL 19.3(4) and COL 19.3(8). Establishing procedures and guidance is included in COL 19.3(10). In Phase 3, makeup to the RWT is provided from offsite sources by the COL applicant, as described in COL 19.3(3).

The staff concludes that the results for the most limiting case are shown in the strategies (a), (b) and (c) above, and these results satisfy the proposed acceptance criterion that the fuel in the SFP remains water covered. The acceptance criterion is consistent with NEI 12-06 guidance and the Order. Therefore it is acceptable.

The staff reviewed the DCD Tier 2 and APR1400-E-P-NR-14005, Revision 2 regarding SFP mitigation strategies including onsite equipment, FLEX equipment, connections to the portable equipment as discussed above, and identified the following RAIs.

Non-seismic Piping Connected to the SFP

In Appendix B of APR1400-E-P-NR-14005, Revision 0, the applicant states that there is no non-seismic piping connected to the SFP that could potentially drain water from the SFP during a seismic event. However, the response to RAIs in SER Section 9.2.1 (RAI 77-7991, Question 09.01.03-1 (ML15196A607) and RAI 473-8582, Question 09.01.03-4 (ML16123A040)) indicated that additional justification may be needed for the above statement, which resulted in a follow-up question, RAI 517-8670, Question 19.03-40 (ML16232A636).

In RAI 473-8582, Question 09.01.03-4 (ML16123A040), the staff requested the applicant to clarify the minimum safety water level credited to be retained in the SFP. The applicant's response stated that the minimum safety water level for the SFP is EL. 146 ft. under the worst postulated accident condition. In RAI 77-7991, Question 09.01.03-1 (ML15196A607), the staff requested the applicant to indicate the elevation of all pipes that interact with the SFP (pipes that penetrate the SFP wall and pipes that extend down into pool). The applicant's response stated that the SFP cleanup suction nozzle is at EL. 149'. In APR1400-E-P-NR-14005, Revision 0, Appendix B for BDBEEs, one of the key assumptions used in the SFP time to boil and makeup analysis is to assume the initial SFP water level to be at normal water level, i.e. EL.154'. However, since the SFP cleanup suction nozzle is non-seismic, it could fail during a seismic BDBEE, causing the water in the SFP to drop to EL.149' instead of EL. 154' as assumed in the time to boil calculation mentioned above. A lower water level means less water inventory available for the pool boil-off and less time available for the operator actions to mitigate the event.

The NEI 12-06, Section 3.2, "Performance Attributes," indicates that "installed equipment that is designed to be robust with respect to design basis external events is assumed to be fully available." In RAI 517-8670 Question 19.3-41 (ML16232A636), the staff requested that the applicant justify/clarify the robustness of pipes that interact with the SFP that are located below the initial SFP water level assumption used in the calculation (i.e., 154'), or to revise the assumption of the initial SFP water level and re-do the SFP analysis for BDBEEs, and revise the technical report and DCD accordingly. In response to RAI 517-8670, Question 19.3-41, dated December 27, 2016 (ML16363A027), the applicant revised the initial SFP water level to EL.149 ft instead of 154 ft as assumed in the time-to-boil calculation. A lower water level means less water inventory available for the pool boil-off and less time available for the operator action to mitigate the event. As a result of this change, there are changes in the technical report and DCD as shown in the markups of this RAI response. The staff reviewed these markups and found them acceptable because these changes reflect the revised calculation resulting from the change of the initial SFP water level and the results of the revised calculation do not change the conclusion of the above evaluation that the fuel in the SFP remains water covered. RAI 517-8670, Question 19.3-41, was therefore resolved and was being tracked as a confirmatory Item. The staff confirmed that the DCD and the technical report markups associated with the RAI 517-8670, Question 19.3-41, response have been incorporated into Revision 1 of the DCD and technical report. Therefore, the staff considered RAI 517-8670, Question 19.03-41, to be resolved and closed.

FLEX Equipment Connections

The NEI 12-06, Section 3.2.2, states that the portable fluid connections for core and SFP cooling functions are expected to have a primary and an alternate connection. Both the primary and alternate connection points do not need to be available for all applicable hazards, but the location of the connection points should provide reasonable assurance of at least one connection being available.

The staff reviewed the information in APR1400-E-P-NR-14005, Section 5.1.2.4.1.2 comparing with the NEI guidance. It is not clear whether the connections being used in the proposed mitigation strategies for SFP cooling are consistent with the guidance that the location of the connection points should provide reasonable assurance of at least one connection being available for all applicable external hazards. In RAI 407-8447, Question 19.03-26 (ML16048A200), the applicant is requested to clarify how APR1400 design for connections to the FLEX equipment is consistent with the NEI guidance.

The applicant responded to the RAI in the letters, dated April 2 (ML16093A005) and June 28, 2016 (ML16180A293). In the response, the applicant states the following:

> The APR1400 design provides two locations for SFP makeup water connections: a set of two primary connections (one for makeup and one for spray) is mounted on the outside wall of the Auxiliary Building (AB), adjacent to the south side of the Emergency Diesel Generator (EDG) building; and the alternate set of identical connections (one for makeup and one for spray) is mounted on the outside wall of AB, adjacent to the north side of the EDG building.
>
> Since the external hazards (e.g., seismic events, floods, high winds, and associated missiles) are dependent on the site-specific characteristics, the protection design for applicable hazards should be the COL applicant's responsibility. Therefore, new COL item for the protection of the connections for FLEX strategies from external hazards will be added to DCD Tier 2 Subsection 19.3.2.3.4 and Technical Report APR1400-E-P-NR-14005, Section 6.2.11, respectively, as indicated in the attachment. In addition, a list of connections for the FLEX strategies (Table 6-5) will be added to APR1400-E-P-NR-14005 as also indicated in the Attachment.

The staff found the response acceptable, because the connections are provided at two separate locations and the new COL 19.3(8) requires COL applicants the protection of the connections from the site-specific external hazards. COL 19.3(8) is included in the list of COL items in Section 19.3.1.5 of this report, COL Information Items. Therefore, RAI 407-8447, Question 19.03-26 was resolved, and the markup for the DCD and APR1400-E-P-NR-14005 associated with the response to RAI 407-8447, Question 19.03-26, was being tracked as a confirmatory item. The staff confirmed that the DCD and APR1400-E-P-NR-14005 markups associated with the RAI response have been incorporated into Revision 1 of the DCD and technical report. Therefore, the staff considered RAI 407-8447, Question 19.03-26, to be resolved and closed.

Raw Water Makeup

The DCD Tier 2 Section 19.3.2.3.2 states that in Phase 2 and Phase 3, makeup water will be taken from the RWT using a portable pump. In RAI 407-8447, Question 19.03-27 (ML16048A200), the staff asked the applicant to explain the design of RWT level instrument. Subsequently, this RAI was combined with RAI 401-8402, Question 19.03-22 (ML16039A017). The applicant was requested to provide appropriate information in the DCD on the design of the RWT water source and its associated flow path (structures, piping, components, and connections) to deliver water to support the containment mitigating strategy and assess if the RWT water source and its associated flow path to the suction of FLEX pump are robust with respect to seismic events, floods, high winds, and associated missiles.

In response to RAI 401-8402 Question 19.03-22, dated June 23, 2016, (ML16175A671) the applicant provided the following information:

> The detailed design of the raw water tank related with site-specific data is the responsibility of COL applicant. The COL applicant will confirm the specific design of raw water tank including associated instrument, capacity, location, flow path to the on-site, the valve pit connected to FLEX equipment, and so on. Also, the COL applicant will confirm that the RWT and flow path to the FLEX equipment (structures, piping, components, and connections) are designed to be robust with respect to seismic events, floods, high winds, and associated missiles. The COL items will be added as below.
>
> > "The COL applicant is to confirm, satisfy or fulfill the specific design functional requirements of raw water tank including the associated instrument, capacity, location, flow path to on-site, the valve pit connected to FLEX equipment, and any other design features as described in DCD Section 19.3 in support of BDBEE mitigation strategies."
> >
> > "The COL applicant is to confirm and ensure that the raw water tank and flow path to the FLEX equipment (structures, piping, components, and connections) are designed to be robust with respect to applicable hazards (e.g., seismic events, floods, high winds, and associated missiles)."

DCD Tier 2, Subsection 19.3.2.3.4 and APR-E-P-NR-14005, Subsection 5.1.2.6.2 will be revised to reflect the above information.

The staff found the response acceptable, because incorporating the new COL items as described above in the RAI responses ensures the design of RWT and the associated equipment to be robust and the design data for robustness are dependent on the site-specific characteristics. These COL items are included in the list of COL items in Section 19.3.1.5 of this report. Therefore, RAI 407-8447, Question 19.03-27, is resolved and closed. The staff confirmed that although the COL item numbering was adjusted in the revised DCD as compared to the RAI response, the content of the RAI response has been incorporated into Revision 1 of the DCD and Revision 1 of the technical report. The staff finds that the applicant's approach described above is consistent with the guidance found in NEI 12-06, as endorsed by JLD-ISG-2012-01, and provides reasonable assurance that the requirements of

Order EA-12-049 can be met by a licensee with respect to spent fuel pool cooling. Therefore the staff considers RAI 401-8402, Question 19.03-22, to be resolved and closed.

19.3.1.4.6 Water and Fuel Supply

Section 5.1.2.6.2 of APR1400-E-P-NR-14005, Revision 1 states that the primary source of water for the core cooling function is the auxiliary feedwater storage tank (AFWST) for the first 72 hours, and the RWT can be used thereafter for up to 11 days. Table 5-2 of APR1400-E-P-NR-14005, Revision 1, shows the water volumes in AFWST and RWT. For the SFP makeup and spray function, the RWT is the source of water.

The detailed design of the RWT related with site specific data is the responsibility of COL applicant. The COL applicant will confirm, satisfy, or fulfill the specific design functional requirements of raw water tank including the associated instrument, capacity, location, flow path to on-site, the valve pit connected to FLEX equipment, and any other design features as described in DCD Section 19.3 in support of BDBEE mitigation strategies. Also, the COL applicant will confirm and ensure that the RWT and flow path to the FLEX equipment (structures, piping, components, and connections) are designed to be robust with respect to applicable hazards (e.g., seismic events, floods, high winds, and associated missiles).

Section 5.1.2.6.3 of APR1400-E-P-NR-14005, Revision 1, addresses the fuel oil supply. EDG fuel oil day tank and the underground 7-day fuel oil storage tanks are used for running the diesel driven FLEX pumps. During Phase 3, fuel oil is provided from an offsite source. For the limiting case, the existing fuel tanks have a capacity for 32 days. In Phase 3, the fuel oil can be refilled from offsite resources.

Table B-3 of APR1400-E-P-NR-14005, Revision 1, provides information on the required makeup volume and available water source. In Table B-3, Column 2, is for Mode 1 to 6 (with no full core offload) and Column 3 is for Mode 6 (with no full core offload). The NRC staff reviewed the table and found both columns are for Mode 6 (with no full core offload). The applicant was requested in RAI 407-8447, Question 19.03-28 (ML16048A200), to explain the differences between these two columns that both refer to Mode 6 with no full core offload.

In response to RAI 407-8447, Question 19.03-28, dated April 2, 2016 (ML16093A005), the applicant clarifies an editorial error by changing the title of the last column from "with no full core offload" to "with full core offload." The staff found the response acceptable. Therefore, RAI 407-8447, Question 19.03-28, is resolved. The staff confirmed that the DCD markups associated with the RAI response have been incorporated into Revision 1 of the technical report. The staff finds that the applicant's approach described above is consistent with the guidance found in NEI 12-06, as endorsed by JLD-ISG-2012-01, and provides reasonable assurance that the requirements of Order EA-12-049 can be met by a licensee with respect to water and fuel supply. Therefore, the staff considered RAI 407-8447, Question 19.03-28, to be resolved and closed.

19.3.1.4.7 FLEX Equipment and Offsite Resources

The following FLEX equipment is identified from DCD Tier 2, Revision 2, Section 19.3.2.3 and APR1400-E-P-NR-14005, Revision 2, Table 5-6 and Table 6-3:

- Three secondary FLEX pumps for core cooling
- Two primary high-head FLEX pump for RCS makeup
- One SFP makeup FLEX pump and one SFP spray pump for SFP cooling

- Two primary low-head FLEX pump for RCS cooling
- Two ECSBS pump for containment spray
- Two 480 V 1000 kW GTG for power supply
- On-site fuel for FLEX equipment
- Flexible hoses

Offsite resources are:

- One 4.16 kV mobile Generator
- Offsite makeup water for refill
- Offsite fuel oil for refill

The staff reviewed the information in the DCD and APR1400-E-P-NR-14005, and requested additional information in RAI 333-8397, Question 19.03-9 (ML15348A114) to:

a) Identify the FLEX equipment that the COL applicant is responsible for,
b) Specify the functional capabilities and interface design parameters for the FLEX equipment,
c) Address the requirement of reasonable protection and accessibility for all the on-site FLEX equipment and the connections of the equipment, and
d) Clarify whether APR1400-E-P-NR-14005 is incorporated by reference in the DCD.

The applicant responded in letter (ML16188A404), dated July 6, 2016 and revised in a letter, dated August 30, 2017 (ML17242A314). In the response to Part (a) and (b), the applicant added Table 6-3 (List of on-site FLEX equipment) and Table 6-4 (List of offsite FLEX equipment) in APR1400-E-P-NR-14005, Revision 1, to summarize the equipment descriptions, quantity, interface parameters, and the functional requirements for the corresponding FLEX equipment. In the response to Part (c), the applicant revised DCD Tier 2, Sections 19.3.2.3.4 and 19.3.4, and APR1400-E-P-NR-14005, Section 6.2.9 to provide detailed guidance for storage of FLEX equipment for COL applicant. The COL item COL 19.3(4) includes that the COL applicant is to address reasonable protection and accessibility for all the on-site equipment. In the response to Part (d), the applicant clarified that APR1400-E-P-NR-14005, Revision 1, is provided as incorporated by reference (IBR) per DCD Tier 2 Table 1.6-2.

The staff reviewed the RAI response and found it acceptable, because the response to Part (a) (b), and (d) adequately addressed the information being requested by the staff and the response to Part (c) is consistent with NEI 12-06, Section 11.3, “Equipment Storage.” Based on the above, RAI 333-8397, Question 19.03-9, was resolved. The DCD and Technical Report markups were being tracked as a confirmatory item. The staff confirmed that the DCD markups associated with the RAI responses have been incorporated into Revision 1 of the DCD and the Technical Report with one exception. The exception has to do with a revised response to RAI 333-8397, Question 19.03-9 (ML17242A314). In this revised response, the applicant changed the number of ECSBS pump for containment spray from one to two. This change has not been reflected in DCD Revision 1, therefore, RAI 333-8397, Question 19.03-9, remained a confirmatory item. The staff confirmed that the revised RAI response markup has been incorporated into Revision 2 of APR1400-E-P-NR-14005, Table 6.3. Therefore, the staff considers RAI 333-8397, Question 19.03-09, and Confirmatory Item 19.03-9, to be resolved and closed.

The applicant states in DCD Tier 2 Section 19.3.2.3.4 that the design approach meets the NEI 12-06 guidance of the N+1 approach for the FLEX equipment. However, it is not clear to the staff how many primary low-head FLEX pumps, which are used for core cooling in low modes Phase 2 operation (see APR1400-E-P-NR-14005, Table 5-6), are in the design to satisfy the N+1 guidance. In RAI 407-8447, Question 19.03-29 (ML16048A200), the applicant is requested to clarify the number of primary low-head pumps in the design.

In response to RAI 407-8447, Question 19.03-29 dated April 2, 2016 (ML16093A005) the applicant states the following:

> There are two primary low-head FLEX pumps designed for APR1400. One primary low-head FLEX pump is required for core cooling in phase 2 of "FLEX strategy for shutdown operation with SGs not available." An additional pump is kept in reserve to satisfy the N+1 guidance.
>
> Section 19.3.2.3.1.2 of DCD Tier 2 and Section 5.1.2.3.3.2 of Technical Report APR1400-E-P-NR14005, Rev. 0 will be revised to clarify the number of FLEX pumps in accordance with the N+1 guidance by adding the following statement:
>
> "Two primary low-head FLEX pumps are provided to meet the N+1 requirement."

The staff found the response acceptable, because it is consistent with NEI 12-06 N+1 guidance. Therefore, RAI 407-8447, Question 19.03-29, was resolved, and the DCD and Technical Report markups were being tracked as a confirmatory item. The staff confirmed that the DCD markups associated with the RAI responses have been incorporated into Revision 1 of the DCD and the Technical Report. Therefore, RAI 407-8447, Question 19.03-29, is resolved and closed.

The applicant in DCD Tier 2, Section 19.3.2.3.4, COL 19.3(9), states that the COL applicant is to address site-specific strategies to mitigate BDBEE as specified in the NRC Order EA-12-049. The staff found that COL 19.3(9) and COL 19.3(10), which will be discussed below in Section 19.3.1.4.9 of the SER, together will require COL applicants to adequately address the training of personnel and programmatic controls and procedures according to NEI 12-06 guidance. The staff finds that the applicant's approach described above is consistent with the guidance found in NEI 12-06, as endorsed by JLD-ISG-2012-01, and provides reasonable assurance that the requirements of Order EA-12-049 can be met by a licensee with respect to FLEX equipment and offsite resources.

19.3.1.4.8 Power Supply

This section of the SE provides the staff's review of the electric power systems available for BDBEE to support key safety functions, which are core cooling, SFP cooling, and containment cooling. In the review of the electric power systems for Phases 1, 2, and 3, of mitigation strategies for BDBEE, the staff used ISG-2012-01, Revision 1. ISG-2012-01, Revision 1, endorsed with comments the methodology describes in NEI 12-06, Revision 0.

Technical Report APR1400-E-P-NR-14005, Section 5.1.2, "Recommendations 4.1 and 4.2 – SBO and FLEX," summarizes the APR1400 diverse and flexible coping (FLEX) strategies for BDBEE, extended loss of all alternating current (ac) power (ELAP) concurrent with loss of normal access to ultimate heat sink (LUHS). The purpose of establishing the FLEX strategies is to maintain core cooling, SFP cooling, and containment heat removal functions. This section of the safety evaluation addresses electric power for equipment needed to maintain core cooling, SFP cooling, and containment heat removal during the following plant conditions: (1) Full-power operation, (2) Low-power operations and shutdown conditions with SGs available, and (3) Shutdown conditions with SGs not available.

According to DCD Tier 2 Section 19.3.2.3, and APR1400-E-P-NR-14005, Revision 1, the APR1400 FLEX strategy follows a three-phase approach.

The three phases are:

- Phase 1 – Initial response using installed equipment (Class 1E DC batteries).
- Phase 2 – Transition phase using portable equipment (1,000 kW, 480 Vac mobile GTGs) and consumables.
- Phase 3 – Indefinite sustainment of these functions using offsite resources (5,000 kW, 4.16 kVac mobile generator).

In the Technical Report, Table 4-1, "Post-Fukushima NRC Recommendations and Requirements," the applicant provides COL 19.3(3), 19.3(4), and 19.3(9) for the COL applicant to implement and satisfy the guidance in ISG-2012-01, Revision 1. The above COL items include the implementation of procedures, guidance, training, and acquisition, staging or installation of equipment needed for the mitigation strategies. According to DCD Tier 2 Section 19.3.2.3, the COL applicant is to address site-specific strategies to mitigate BDBEEs and includes the establishment of procedures and guidance on mitigation of BDBEEs.

<u>Phase 1-Initial Response Using Installed Equipment</u>

During BDBEE ELAP, the nuclear plant unit(s) lose all alternating current (ac) power including offsite ac and onsite ac power sources, and these power sources are assumed to be unavailable and not imminently recoverable. The applicant's mitigating strategy for a BDBEE assumes that the only available electric power sources during Phase 1 are the Class 1E (safety-related) station batteries. During Phase 1 ELAP, the APR1400 design includes 125 direct current (dc) volts (Vdc) Class 1E batteries which provide power for the operation of 4.16 kV switchgear, 480 Vac motor operated valves (MOVs), air operated valves (AOVs), instrumentation and controls (I&C) panels, 125 Vdc loads, shutdown system instrumentation, and 120 Vac loads.

DCD Tier 2 Figure 8.3.2-1, "Class 1E DC Power System," depicts the Class 1E DC power system. According to DCD Tier 2 Section 8.3.2.1.2.1, "Class 1E 125 Vdc Power System," the onsite Class 1E 125 Vdc power system is composed of four independent subsystems (trains A, B, C, and D) and supplies power to the plant safety system dc loads and essential I&C system loads. Each dc power subsystem consists of a battery, two battery chargers (normal and standby), a dc control center, and distribution panels. The Class 1E 125 Vdc power systems are located in a seismic Category I structure. The Class IE 125 Vdc power system is designed to remain functional in the event of a safe shutdown earthquake, operating basis earthquake, tornadoes, hurricanes, floods, and other design basis events including missile impact and internal accidents.

The APR1400 mobile GTGs supplying ac power are not available during the Phase 1 period. The Technical Report, Section 5.1.2.6.1.3, "Emergency Lighting," explains that the APR1400 emergency lighting design provides lighting in areas such as the MCR and TSC/operational support center (OSC) through Class 1E batteries during Phase 1.

DCD Tier 2 Section 19.3.2.3.4, Technical Report, Section 5.1.2.3.1.1.2 explains that the Phase 1 coping time can be extended to 16 hours. Further, the Technical Report states in Phase 1 that additional cooling is not required in the MCR, rooms containing electrical, turbine driven auxiliary feed-water pump (TDAFWP), and instrumentation and control (I&C) equipment; and that the TDAFWPs are powered from the Class 1E batteries in trains C and D for 16 hours without load shedding. Technical Report, Section 5.1.2.6.1.2, "DC Power" states in part that trains C and D batteries have a capacity of 8,800 ampere hour (Ah) and can supply dc power up to 16 hours without load shedding.

In Part 1 of RAI 420-8482, Question 19.03-30 (ML16060A444), the staff asked the applicant to clarify whether the I&Cs associated with the TDAFWPs are powered by the trains C and D Class 1E batteries. In response to RAI 420-8482, Question 19.03-30, Part 1, dated June 3, 2016 (ML16155A342), the applicant stated all instrumentation, controls and valves that are essential to the operation of the steam turbine driven pump (trains C and D) are powered from the Class 1E batteries of trains C and D for 16 hours without load shedding.

The staff finds that the response to Part 1 of RAI 420-8482, Question 19.03-30, is acceptable because the applicant provided clarification that the Class 1E batteries of trains C and D power the instrumentation, controls and valves that are essential to the operation of the TDAFWPs. In addition, the applicant stated that during Phase 1, the trains C and D Class 1E batteries are capable of providing power to equipment beyond the 8 hour Phase 1 duration. Therefore, Part 1 of RAI 420-8482, Question 19.03-30, is resolved and closed.

DCD Tier 2 Section 8.3.2.1.2.6, “System Capacity and Capability,” states in part that the Class 1E batteries are sized based on the duty cycle of the respective subsystems. Each battery is capable of supplying power to the worst-case operating loads for a period of the battery duty cycle. The sizing of the battery is performed in accordance with the IEEE Std. 485-2010, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications”. Class 1E battery loads and duty cycles are shown in the DCD Tier 2 Table 8.3.2-1 and the battery rating is shown in DCD Tier 2 Table 8.3.2-4. The Class 1E batteries are qualified in accordance with IEEE Std. 535-2006, “IEEE Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations.” Section 8.3.2 of this report addresses the applicant’s use of IEEE Std. 535.

In Part 2 of RAI 420-8482, Question 19.03-30 (ML16060A444), the staff requested that the applicant provide load analysis and methodology used in order to demonstrate that the batteries have the capacity to last for 16 hours without load shedding. In response to RAI 420-8482, Question 19.03-30, dated June 3, 2016 (ML16155A342), the applicant stated the following:

- Each battery is capable of supplying power to the worst-case operating loads for a period of the battery duty cycle. The sizing of the battery is performed in accordance with IEEE Std. 485-2010. The capacity of trains C and D Class 1E 125 Vdc batteries are sized to last for 16 hours without load shedding.

- The list of battery loads for battery sizing is provided as Attachment 1 (pages 5 thru 8) of response to RAI 441-8549, Question 08.03.02-3, based on load analyses for trains C and D Class 1E 125 Vdc. The duty cycle diagrams of trains C and D Class 1E 125 Vdc batteries for battery sizing are provided as Attachment 2 (Figure 3 and 4) of response to RAI 441-8549, Question 08.03.02-3. The cell sizing worksheets of trains C and D Class 1E 125 Vdc batteries as per guidance provided in IEEE Std. 485-2010 for battery sizing are provided as Attachment 3 (pages 3 thru 38) of response to RAI 441-8549, Question 08.03.02-3.

The staff finds that the response to RAI 420-8482, Question 19.03-30, Part 2, is acceptable because the applicant provided the load analysis and methodology used and provided the results in the RAI 441-8549, Question 08.03.02-3 response. The response to RAI 441-8549, Question 08.03.02-3 stated that IEEE Std. 535-2013 methodology is used to qualify the batteries for duty cycles greater than eight hours, as is the case for the extended duty cycle of 16 hours for trains C and D batteries. Section 8.3.2 of this report addresses the applicant’s use of IEEE Std. 535. The applicant also revised DCD Tier 2 Table 1.9-1, Sections 3.11.8, 8.1.3.3, and 8.3.4 to address the qualification for the extended duty cycles. Staff found that the results demonstrated that the batteries have the capacity to provide power for 16 hours without load shedding. The staff finds the response acceptable because the battery sizing was done in accordance with IEEE Std. 485 as endorsed by RG 1.212, "Sizing of Large Lead-Acid Storage Batteries." The staff confirmed that the RAI response markup has been incorporated into Revision 1 of the DCD. Therefore, the staff considers Part 2 of RAI 420-8482, Question 19.03-30, to be resolved and closed.

In Part 3 of RAI 420-8482, Question 19.03-30 (ML16060A444), the staff asked the applicant to provide the battery duty cycle diagram for train C and D that depicts the battery load profile and the battery division(s) providing power to the corresponding loads along the timeline for the mitigating strategies to maintain core cooling, containment, and spent fuel pool cooling during all modes of operation. In response to RAI 420-8482, Question 19.03-30, Part 3, dated June 3, 2016 (ML16155A342), the applicant stated that the dc battery load requirements for BDBEE mitigation strategies have been incorporated in the list of trains C and D Class 1E 125 Vdc battery loads and the battery duty cycle diagrams for trains C and D Class 1E 125 Vdc battery loads which were provided as Attachment 1 (pages 5 thru 8) and Attachment 2 (Figure 3 and 4) of response to RAI 441-8549, Question 08.03.02-3.

The staff finds that the response to RAI 420-8482, Question 19.03-30, Part 3, is acceptable because as part of the response to RAI 441-8549, Question 08.03.02-3, the applicant provided the battery duty cycle diagram for trains C and D that depicts the load profile. The battery duty cycle diagrams depict the timelines for the loads needed for mitigating strategies. The staff evaluated the battery sizing in Section 8.3.2 of this report. The staff finds that the battery duty cycle is used for sizing per IEEE Std. 485 and as such, the sizing methodology is adequate to determine the capacity of the battery. The staff confirmed that the RAI response markup has been incorporated into Revision 1 of the DCD. Therefore, the staff considers Part 3 of RAI 420-8482, Question 19.03-30, to be resolved and closed.

In Part 4 of RAI 420-8482, Question 19.03-30 (ML16060A444), the staff asked the applicant to provide the basis for the assumed minimum battery voltage that is required to ensure proper operation of all electrical equipment as included in the load profile. In response to RAI 420-8482, Question 19.03-30, Part 4, dated June 3, 2016 (ML16155A342), the applicant stated that considering voltage drop across the cable between the dc bus and equipment terminals, Class 1E 125 Vdc system of the APR1400 design was determined as 105 V at the dc bus terminal for proper component operation. The staff finds that the response to RAI 420-8482, Question 19.03-30, Part 4, is acceptable because the applicant ensures proper operation of all electrical equipment. Therefore, Part 4 of RAI 420-8482, Question 19.03-30, is resolved and closed.

DCD Tier 2 Section 8.3.2.1.2.1, "Class 1E 125 Vdc Power System," states, in part, that the onsite Class 1E 125 Vdc power system is composed of four independent subsystems (trains A, B, C, and D) and supplies power to the plant safety system dc loads and essential I&C system loads. DCD Tier 2 Figure 8.3.2-1, "Class 1E DC Power System" shows Class 1E dc power systems for trains A, B, C, and D. However, for BDBEE Phase 1 coping, the Class 1E batteries in train C and D are relied on to power loads for up to 16 hours. Technical Report, Section 5.1.2.6.1.2, "DC Power," states in part that both train A and B batteries have a capacity of 2,800 Ah and can supply dc power up to 2 hours without load shedding and an additional 6 hours with load shedding. Train C and D batteries each have a capacity of 8,800 Ah and can supply dc power up to 16 hours without load shedding.

In RAI 420-8482, Question 19.03-31 (ML16060A444), the staff requested that the applicant clarify whether train A and B Class 1E dc power subsystems are used for mitigating strategies during BDBEE and specify for which phase they are credited. In response to RAI 420-8482, Question 19.03-31, Part 1, dated June 10, 2016 (ML16162A150), the applicant stated that during BDBEE Phase 1, trains C and D Class 1E dc power systems are credited to support the operation of TDAFWP, as stated in TR, Section 5.1.2.6.1.1. The APR1400 design does not take credit for trains A and B dc power systems since the decay heat removal function performed by the TDAFWP can be maintained even if trains A and B Class 1E dc power is not available. During Phase 2, a 480 V mobile GTG will be connected to either train A or train B of the Class 1E load center to supply power and recharge respective battery to full charged condition. The applicant also explained that during Phase 3, a 4.16 kV mobile Generator will be used to restore train A or B of the 4.16 kV Class 1E power system. Train A or B Class 1E 125 Vdc power system is credited to cope with BDBEE during Phase 2 and 3. The staff finds that the response to RAI 420-8482, Question 19.03-31, Part 1 is acceptable because the applicant provided clarification that trains C and D Class 1E dc power systems are used to cope with BDBEE during Phase 1, while the train A or B Class 1E 125Vdc power system are used to cope with BDBEE during Phases 2 and 3. Therefore, Part 1 of RAI 420-8482, Question 19.03-31, is resolved and closed.

DCD Tier 2 Section 8.3.2.1.2.1 states that the Class 1E 125 Vdc power systems, located in a seismic Category I structure, are designed to remain functional in the event of a safe shutdown earthquake, operating basis earthquake, tornadoes, hurricanes, floods, and other design basis events including missile impact and internal accidents. Technical Report, Section 5.1.2.3.1.1.2, explains that during the Phase 1, additional cooling in MCR, electrical and I&C equipment rooms, and the TDAFWP rooms is found not to be required based on heat-up calculations.

In Part 1 of RAI 420-8482, Question 19.03-33 (ML16060A444), the staff requested that the applicant explain the environmental conditions, including temperature, existing in the room housing the Class 1E DC batteries, and whether there is any impact to the functioning of the batteries during Phase 1 and beyond. In response to RAI 420-8482, Question 19.03-33 Part 1, dated May 23, 2016 (ML16144A683), the applicant stated that during normal operation, the environmental conditions of Class 1E battery rooms are as follows: Temperature (°F): 65 ~ 85; Relative Humidity (%): 7 ~ 90; Pressure (psig): 0. During Phase 1, the applicant stated that the temperatures of trains A, B, C, and D Class 1E battery rooms are 95.4 °F, 97.0 °F, 97.1 °F and 97.1 °F, respectively. The applicant also explained that during Phase 2 the train A and B Class 1E battery rooms maximum temperatures are maintained at the outside summer temperature around 100 °F and the minimum temperature of 65 °F. Also, the train C and D Class 1E battery room temperatures will be 102.8 °F, respectively, based on the heat-up calculations. The applicant also explained that the operating temperature limit of the battery, to prevent mechanical and/or performance degradation (or failure) is 32 °F (minimum) and 113 °F (maximum) for 72 hours. The above mentioned battery operating temperature limits are in accordance with the manufacturer’s information. In addition, the functioning of trains A, B, C, and D Class 1E battery will not be affected during Phase 1 and 2.

The staff finds that the response to RAI 420-8482, Question 19.03-33, Part 1, is acceptable because the batteries are designed to withstand the temperature range from 32 °F (minimum) to 113 °F (maximum) and the applicant explained that the room temperature is not going to exceed 97.1 °F during Phase 1 for the trains A, B, C, and D Class 1E battery rooms. The staff also finds that the response is acceptable because the trains A, B, C, and D Class 1E battery room temperatures (minimum, maximum) are within the allowed design range. In addition, the applicant provided battery cell sizing details, which accounts for temperature factors as part of the response to RAI 441-8549, Question 08.03.02-3. Therefore, Part 1 of RAI 420-8482, Question 19.03-33, is resolved and closed.

In Part 2 of RAI 420-8482, Question 19.03-33 (ML16060A444) the staff asked the applicant to explain what would happen in the case of higher than normal temperatures in the battery rooms and to discuss if and how higher than normal temperatures are factored into the analysis to support both functioning and duration of the battery life. In response to RAI 420-8482, Question 19.03-33 Part 2, dated May 23, 2016 (ML16144A683), the applicant explained that in the case of higher than normal temperatures in the battery rooms, the temperature excursion of the battery room will be maintained within the allowable maximum temperature (113 °F) of the battery based on the manufacturer's information. The functioning of trains A, B, C, and D Class 1E battery will not be significantly affected by the abnormal temperature during Phase 1 and 2. The staff finds that the response to RAI 420-8482, Question 19.03-33, Part 2, is acceptable because as discussed above the trains A, B, C, and D Class 1E battery room temperatures (minimum, maximum) are within the allowed design range and are designed to account for temperature excursions. Therefore, Part 2 of RAI 420-8482, Question 19.03-33, is resolved and closed.

DCD Tier 2 Section 9.5.3.1, "Design Bases," states in part that the emergency lighting system is composed of emergency ac and emergency dc lighting systems. Emergency ac lighting is supplied from Class 1E buses. Emergency dc lighting system is composed of the lighting powered from the non-Class 1E 125 Vdc station battery and the lighting powered by an individual 8hours rated self-contained battery pack units in accordance with RG 1.189, "Fire Protection for Nuclear Power Plants." Technical Report, Section 5.1.2.6.1.3, "Emergency Lighting," states in part that emergency lighting in areas such as the MCR and TSC/operational support center (OSC) is provided from the Class 1E batteries during Phase 1. In RAI 420-8482, Question 19.03-38 (ML16060A444), the staff asked the applicant to clarify the power source (emergency ac, emergency dc, or combination) for emergency lighting system during Phase 1 of the BDBEE for areas such as MCR, TSC/OSC, and other areas requiring lighting for operator actions.

In response to RAI 420-8482, Question 19.03-38, dated April 29, 2016, (ML16120A173), the applicant explained that emergency dc lighting is powered from Class 1E batteries from train C or D during Phase 1 of the BDBEE and provide lighting to the MCR and TSC/OSC. For other areas requiring lighting for operator actions, an individual 8-hour rated self-contained battery pack lighting fixtures which are addressed in DCD Tier 2 Subsection 9.5.3.2, provide the lighting during Phase 1 of the BDBEE. DCD Tier 2 Table 8.3.2-1 and Technical Report, Section 5.1.2.6.1.3 address lighting. Technical Report, Section 5.1.2.6.1.3 explained that emergency ac lighting is provided from the 480 V mobile GTG during Phase 2 and from the 4.16 kV mobile generator during Phase 3.

The staff finds that the response to RAI 420-8482, Question 19.03-38, is acceptable because the applicant provided clarification that the emergency dc lighting is being powered from the trains C and D batteries and staff confirmed the lighting loads are contained in DCD Tier 2 revised Table 8.3.2-1, "Class 1E 125 Vdc Power System Loads." The staff confirmed that the RAI response markup has been incorporated into Revision 1 of the DCD. Therefore, the staff considers RAI 420-8482, Question 19.03-38, to be resolved and closed.

The APR1400 electrical design supports Phase 1 of the mitigation strategies for BDBEE with Class 1E batteries. The trains C and D Class 1E batteries supply dc power to essential I&C equipment, emergency dc lighting, and for the I&C power of the TDAFWPs. The trains A and B batteries are not credited for use during Phase 1. The dc battery power is also converted to ac power via inverters and transformers, and together they provide electric power for equipment needed to maintain core cooling, spent fuel pool cooling, and containment heat removal. The design relies on trains C and D batteries which are sized to support the required loads for 16 hours without load shedding, which is beyond the 8 hours needed for Phase 1. The batteries will also be qualified per the IEEE Std. 535-2013 to ensure that they are capable for duty cycles up to 16 hours as described in DCD Tier 2 Section 19.3, and RAI responses discussed above.

The NEI 12-06, Revision 0 guidance explains Phase 1 coping would rely on installed plant equipment. The APR1400 Class 1E dc electrical distribution system remains available following a BDBEE, and is consistent with the NEI 12-06, Revision 0 guidance which states that installed electrical distribution systems remain available provided they are protected. As discussed above, the Class 1E dc electrical distribution system has adequate capacity and capability to maintain core cooling, containment and SFP cooling capabilities in Phase 1 of a BDBEE.

Therefore, the staff finds that the APR1400 electrical design can support Phase 1 of mitigation strategies for BDBEE, and is consistent with the NEI 12-06, Revision 0 guidance which relies on installed plant equipment for Phase 1.

<u>Phase 2 - Transition Phase Using Portable Equipment and Consumables</u>

In Technical Report, Section 6.2.6, “Electric Power Supply System,” the applicant explained that the APR1400 design includes two 480 Vac, 1,000 kW mobile onsite GTGs and they provide power for the Class 1E 480 V load centers during Phase 2. The mobile GTGs are connected to the 480 V load center train A (or B). The 480 V mobile GTGs power the 480 V load center and MCC, 480 Vac / 125 Vdc battery charger, 125 Vdc battery, 125 Vdc / 120 Vac inverter, and 120 Vac distribution panel in train A (or B). Technical Report, Section 5.1.2.6.1.2 states, in part, that, during Phase 2, a 480 V mobile GTG is connected to either train A or train B of the Class 1E load center to supply power and recharge respective batteries to the fully charged condition. Fuel for the 480 V mobile GTGs is provided from the EDG fuel oil storage and day tank as shown in Technical Report, Table 5-5, “Summary of Fuel Oil Demand.” Technical Report, Section 5.1.2.6.1.3 states, in part, that emergency lighting in areas such as the MCR and TSC/ OSC is provided from the mobile GTG during Phase 2.

The Technical Report, Table 5-4, “480 V and 4.16 kV Mobile Generator Electrical Load Summary List (in kW),” and Technical Report, Table C-1, “480 V Mobile GTG Electrical Loadings,” provide load information and total load with a 10 percent margin for the 480 V mobile GTGs used during Phase 2. Based on the total load, the staff found that the mobile GTG has adequate capacity to meet the load requirements. The staff also confirmed that the GTGs were adequately sized during an audit on May 25, 2016, and documented its findings in a report dated December 7, 2016 (ML16300A205).

Technical Report, Section 5.1.2.6.1.2, “DC Power,” states, in part, that during Phase 2, a 480 V mobile GTG is connected to either train A or train B of the Class 1E load center to supply power and recharge respective batteries to fully charged condition. During battery charging in Phase 2, forced ventilation of battery rooms may be required to prevent an unacceptable buildup of hydrogen released during the charging process. Ventilation of battery rooms may be needed to maintain an acceptable temperature for long-term battery operation. The RG 1.128, “Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants,” states that each ventilation system of the Class 1E battery rooms limits hydrogen accumulation to less than 1 percent of the total volume of the battery area. DCD Tier 2 Section 8.3.2.2.2, “Conformance with NRC Regulatory Guides,” states, in part, that each ventilation system of the Class 1E battery rooms limits hydrogen accumulation to less than 1 percent of the total volume of the battery area. Technical Report does not identify considerations related to the need for battery room ventilation.

In RAI 420-8482, Question 19.03-32 (ML16060A444) the staff asked the applicant how mitigation strategies will address ventilation requirements in support of battery charging and operation, and to provide a discussion of the hydrogen gas exhaust. Specifically, the staff requested a discussion on how hydrogen concentration in the battery rooms will be maintained below the limits established by national standards and codes (i.e., less than 1 percent according to the National Fire Code and RG 1.128, "Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants," which endorses IEEE Std. 484, "IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications," with exceptions) when the batteries are being recharged during Phase 2. In addition, staff requested the applicant to specifically address considerations related to maintaining an acceptable temperature range for long-term battery operation.

In response to RAI 420-8482, Question 19.03-32, dated April 18, 2016, (ML16109A170), the applicant stated that the battery room supply fans, battery room exhaust fans, and battery room electrical duct heater (EDH) in train A or B are powered from the 480 V mobile GTG during Phase 2. In addition, the applicant explained that there is no hydrogen generation from the train C or D Class 1E battery, because the train C or D Class 1E battery is not recharged during the coping phases (Phases 1, 2, and 3). The battery room supply fan supplies filtered outside air to the battery room and the air in the battery room is exhausted by the battery room exhaust fan to maintain hydrogen gas concentration at less than one (1) percent by volume in the battery room.

The staff finds that the response to RAI 420-8482, Question 19.03-32, related to hydrogen generation, concentration, and exhaust to be acceptable because the applicant explained that the ventilation system will be operable to maintain the appropriate hydrogen level in Phase 2 BDBEE during battery charging, and that the battery functionality is not affected. Therefore, Part 1 of RAI 420-8482, Question 19.03-32, is resolved and closed.

In response to RAI 420-8482, Question 19.03-32, related to maintaining acceptable temperature range for long term battery operation, the applicant explained that during normal operation, the battery room is maintained in the temperature range of 65 °F to 85 °F by operating the safety-related battery room cooler. The temperature of the battery room is maintained around the temperature of the outside air supplied by the battery room supply fan during summer season (100 °F), because heat generation from the battery is less than 0.2 kW. The battery can provide power to vital loads when the battery room is at 113 °F for 72 hours. Also, the minimum temperature of the battery room, 65 °F, is maintained by operating the electric duct heater to preserve battery capacity.

The staff finds that the response to RAI 420-8482, Question 19.03-32, Part 2, is acceptable because the batteries can power the vital loads because the battery rooms can be maintained within the manufacturer's recommended temperature range (Min. 32°F and Max. 113 °F). Therefore, Part 2 of RAI 420-8482, Question 19.03-32, is resolved and closed.

In Part 3 of RAI 420-8482, Question 19.03-32 (ML16060A444), the staff asked the applicant to explain whether only train A or train B Class 1E batteries are recharged or whether there is charging done to the train C or train D Class 1E batteries during Phase 2. In response to RAI 420-8482, Question 19.03-32, Part 3 dated April 18, 2016 (ML16109A170), the applicant stated that a 480 V mobile GTG is connected to the 480 V Class 1E power system train A or B during Phase 2, either train A or B Class 1E batteries are recharged during Phase 2. The train C or D Class 1E batteries are not recharged during Phases 1, 2, and 3.

The staff finds that the response to RAI 420-8482, Question 19.03-32, Part 3, is acceptable because the applicant clarified that the train C or D Class 1E batteries are not recharged during the coping phases (Phases 1, 2, and 3). The staff also verified that only the trains A and B battery charger loads are accounted for in the Technical Report, Table C-1 480 V loads. Therefore, Part 3 of RAI 420-8482, Question 19.03-32, is resolved and closed.

Technical Report, Section 5.1.2.3.1.2, "Phase 2: Coping with Installed Plant Equipment and Onsite Portable Resources (8 to 72 hours)," Subsection 5.1.2.3.1.2.1 states that two 480 V, 1,000 kW, mobile GTGs are provided to meet N+1 guidance. NEI 12-06, Revision 0 (ML12242A378) explains that provision of at least N+1 sets of portable on-site equipment stored in diverse locations or in structures designed to reasonably protect from applicable BDBEEs is essential to provide reasonable assurance that N sets of FLEX equipment will remain deployable to assure success of the FLEX strategies. Therefore, provision of a spare capability to support the safety functional requirements beyond the minimum necessary to support the "N" units on-site. One of the 480 V mobile GTGs is connected to its respective 480 V Class 1E power system train, and supplies power to the 125 Vdc battery charger, the 480 V load center, and the motor control center (MCC). During this phase, additional cooling in MCR, electrical and I&C equipment rooms, TDAFWP rooms, and auxiliary control panel (ACP) room is not required.

NEI 12-06, Revision 0, Section 11.3, "Equipment Storage," states that FLEX equipment should be stored and maintained in a manner that is consistent with assuring that it does not degrade over long periods of storage and that it is accessible for periodic maintenance and testing. In Part 1 of RAI 420-8482, Question 19.03-34 (ML16060A444), the staff asked the applicant to explain the environmental conditions, including temperature, in the room housing the mobile GTGs, and whether there are any impacts to the functioning of the GTGs during Phase 2 and beyond.

In response to RAI 420-8482, Question 19.03-34, Part 1, dated April 29, 2016 (ML16120A173) and revised response dated November 17, 2017 (ML17322A041), the applicant explained that as stated in COL 19.3(4), the details of the storage location for FLEX equipment, including mobile GTGs, are to be addressed by the COL applicant. The environmental conditions (e.g., temperature, humidity, etc.) of the specific storage room will also be addressed by COL applicant. The functionality (performance) of the proposed 480 V, 1000 kW mobile gas turbine generator(s) is not expected to be affected by temperature and humidity during BDBEE, since the GTGs procured from the manufacturer will meet or exceed the specification and will consider impact on the GTG rating at elevated temperature. Additionally, periodic testing and proper maintenance of the mobile GTGs will be conducted in accordance with manufacturer's recommendations to demonstrate GTG readiness when called upon to operate during FLEX Phases 2 and beyond. The applicant also explained that COL items COL 19.3(4) and 19.3(9), as discussed in DCD Tier 2, Sections 19.3.2.3.4, will be fully addressed by the COL applicant to conform to the guidance of NEI 12-06, Revision 0 and JLD-ISG-2012-01, Revision 0.

DCD Tier 2 Section 19.3.2.3.4 discusses the requirements for the COL 19.3(4) in which the COL applicant is to address details of the storage location for FLEX equipment and COL 19.3(9), which includes (1) evaluation of site-specific external hazards, (2) determination and protection of portable equipment, (3) providing means for acquisition, staging, and installation of equipment, and (4) establishing means for maintaining and testing of portable equipment. NEI 12-06, Revision 0, Section 2.2, "Determine Applicable Extreme External Hazards," explains that each plant will evaluate the applicability of external hazards and, where applicable, address the implementation considerations associated with each for the protection of FLEX equipment. The staff finds that COL 19.3(4) satisfies the guidance for protection of FLEX equipment, because the COL applicant will provide suitable storage location and reasonable protection of FLEX electrical equipment. NEI 12-06, Revision 0, Section 1.1, “Background,” explains that programmatic controls would establish standards for quality, maintenance, testing of FLEX equipment, configuration management and periodic training of personnel. The staff finds that COL 19.3(9) satisfies the NEI 12-06 guidance for ensuring that FLEX equipment will be available following a BDBEE, because the COL applicant will implement procedures for training, acquisition, staging, installation, maintenance and testing of portable electrical equipment used for FLEX. Therefore, the staff finds that response to RAI 420-8482, Question 19.03-34, Part 1, is acceptable because the applicant explained that the functionality of the GTGs will not be impacted during BDBEE and that the COL applicant will be responsible for ensuring that the environmental conditions will not impact the GTG performance, per COL 19.3(4) and 19.3(9). The staff finds that COL 19.3(4) and 19.3(9) conforms to the guidance in NEI 12-06, for each plant will evaluate the applicability of external hazards and, where applicable, address the implementation considerations associated with each for the protection of FLEX equipment. Therefore, Part 1 of RAI 420-8482, Question 19.03-34, is resolved and closed.

In Part 2 of RAI 420-8482, Question 19.03-34 the staff asked the applicant to discuss how isolation between Class 1E and non-Class 1E equipment (mobile generators) is maintained, in accordance with NEI 12-06, Section 3.2.2, “Minimum Baseline Capabilities,” Guideline (13) “Use of portable equipment, e.g., portable power supplies, portable pumps, etc., can extend plant coping capability. The procedures/guidance for implementation of these portable systems should address the transitions from installed sources to portable sources.”

In response to RAI 420-8482, Question 19.03-34, Part 2, dated April 29, 2016, (ML16120A173), the applicant explained that the Class 1E safety buses are designed with the physical and electrical independence from non-Class 1E equipment. The interface arrangement between the Class 1E safety buses and the non-Class 1E equipment is maintained by Class 1E circuit breakers, which serve as isolation devices in accordance with IEEE Std. 384 as endorsed by RG 1.75, “Physical Independence of Electric Systems.” The applicant also explained that the specific design, location, and connection configuration has been addressed in the response to RAI 61-7984, Question 08.03.01-5 (ML15251A244). In addition, the applicant explained that there will be plant procedures prepared by the COL applicant (COL 19.3(9)) in accordance with NEI 12-06, Section 3.2.2, Guideline (13). The staff’s review of electrical separation between Class 1E and non-Class 1E equipment for conformance to RG 1.75 is provided in Section 8.3.1 of this SE.

The staff finds that response to RAI 420-8482, Question 19.03-34, Part 2, is acceptable because (1) the applicant explained that there will be a physical independence and electrical separation between the Class 1E and non-Class 1E equipment provided with Class 1E circuit breakers, (2) provided a diagrammatic representation of the design including the isolation device in response to RAI 61-7984, Question 08.03.01-5 (ML15251A244), which depicts the isolation of the Class 1E and non-Class 1E systems, and, (3) provided COL 19.3(9) for the COL applicant to establish procedures to ensure that isolation is maintained. RG 1.75, Revision 3, 2005 (ML043630448) explains that Sections 7.1.2.1, 7.1.2.4, and 7.2.2.3 of IEEE Std. 384-1992 should be supplemented to allow circuit breakers or fuses that are automatically opened by fault current to be used as an isolation device. Thus, the use of Class 1E circuit breakers as isolation devices between Class 1E and non-Class 1E equipment is in accordance with RG 1.75. The staff confirmed that the RAI response markup has been incorporated into Revision 1 of the DCD. Therefore, the staff considers RAI 420-8482, Question 19.03-34, to be resolved and closed.

In RAI 420-8482, Question 19.03-35 (ML16060A444), the staff stated that Technical Report, Section 5.1.2.3.1.2, "Phase 2: Coping with Installed Plant Equipment and Onsite Portable Resources (8 to 72 hours)," Subsection 5.1.2.3.1.2.3 states that the specific storage location, mobilization, and other details for the FLEX pumps and mobile GTGs are COL items. Technical Report, Section 5.1.2.3.1.2.3, "Common Strategy to Both the Basic Strategy and Contingency Plan" states that the specific storage location, mobilization, and other details for the FLEX pumps and mobile generators are COL items. DCD Tier 2 Section 19.3.4, "Combined License Information," COL 19.3(4) states that the COL applicant is to address the details of storage location for FLEX equipment. Therefore, the staff requested that the applicant specify and provide details associated in the COL item to ensure that the COL applicant is able to provide mobilization and storage to ensure adequate protection of FLEX equipment for the Phase 2 coping strategy.

In response to RAI 420-8482, Question 19.03-35, dated June 30, 2016 (ML16182A539), the applicant stated that COL 19.3(4) in DCD Tier 2 Section 19.3.4 and Technical Report, Section 6.2.9, "Storage of FLEX Equipment," are revised to provide guidance for storage and deployment of FLEX equipment to ensure that the COL applicant is able to provide mobilization and storage to ensure adequate protection of FLEX equipment for the Phase 2 coping strategy in the response to RAI 333-8397, Question 19.03-9. In the response to RAI 333-8397, Question 19.03-9 (ML16188A404), the applicant revised COL 19.3(4) in DCD Tier 2, Subsection 19.3.2.3.4 and 19.3.4, and Technical Report, Section 6.2.9. The revised COL 19.3(4) in Table 1.8-2, "Combined License Information Items," and in Tier 2 Section 19.3.2.3.4, which states that the COL applicant is to address the details of selecting suitable storage locations for FLEX equipment as provided in NEI 12-06, Sections 5 through 9, and the details of the guidance for storage of FLEX equipment provided in the Technical Report, Section 6.2.9. In Technical Report, Section 6.2.9, the applicant provided a list of items and stated the following in the heading of the list: "The COL applicant is responsible for addressing the details of the following guidance for the storage and deployment of the FLEX equipment." For example, the FLEX equipment is required to be stored and maintained in a manner that is consistent with assuring that it does not degrade over long periods of storage and that it is accessible for periodic maintenance and testing, and is stored in a dedicated buildings/structures that will withstand the BDBEEs.

The staff finds that the response to RAI 420-8482, Question 19.03-35, is acceptable because the applicant provided the details of COL 19.3(4) to ensure that the COL applicant provides mobilization, storage, and adequate protection of FLEX equipment for the Phase 2 coping strategy. The staff confirmed that the RAI response markup has been incorporated into Revision 1 of the DCD and the Technical Report, Revision 1. Therefore, the staff considers RAI 420-8482, Question 19.03-35, to be resolved and closed.

The APR1400 electrical design supports Phase 2 of the mitigation strategies for BDBEE with two - 1,000 kW, 480 Vac GTGs and trains A and B Class 1E batteries. During Phase 2, the 480 V mobile GTGs provide power to the trains A and B Class 1E 480 V load centers, which feed the 480 V MCCs, auxiliary charging pump (ACP), main steam atmospheric dump valves (MSADVs), essential I&C equipment, battery chargers, inverters, transformers, and 120 Vac distribution panels. The load requirements are described in Technical Report, Appendix C, Table C-1, "480 V Mobile GTG Electrical Loadings," and includes loads for the trains A and B Class 1E battery chargers. The 480 V trains A and B systems provide electric power for equipment needed to maintain core cooling, spent fuel pool cooling, and containment heat removal in Phase 2. The trains A and B Class 1E batteries are recharged during Phase 2 and are sized to support the required loads for 2 hours without load shedding, and an additional 6 hours with load shedding. The 480V mobile GTGs and 480V systems are designed to meet the

Phase 2 load requirements to maintain the reactor in a safe condition and is described in the Technical Report, DCD Tier 2 Section 19.3, and RAI responses discussed above.

The NEI 12-06, Revision 0, guidance explains that FLEX will increase defense-in-depth for beyond-design-basis scenarios to address an ELAP and LUHS occurring simultaneously at all units on a site. The applicant provided COL 19.3(4) and 19.3(5), which are evaluated above, to ensure that FLEX equipment will be available to support Phase 2 of a BDBEE. Therefore, the staff finds that the APR1400 electrical design can support Phase 2 of mitigation strategies for BDBEE and is consistent with NEI 12-06, Revision 0 guidance which relies on onsite FLEX equipment for Phase 2.

<u>Phase 3 – Indefinite Sustainment of these Functions Using Offsite Resources</u>

The APR1400 design includes one 4.16 kV, 5,000 kW mobile Generator to provide power for the Class 1E 4.16 kV switchgear during Phase 3 and can be connected to the 4.16 kV switchgear train A (or B). The 4.16 kV mobile generator powers the 4.16 kV switchgear, 480 V load center and MCC, 480 Vac / 125 Vdc battery charger, 125 Vdc battery, 125 Vdc / 120 Vac inverter, and 120 Vac distribution panel in train A (or B). TR, Table 5-5, “480 V and 4.16 kV Mobile Generator Electrical Load Summary List,” and Technical Report, Appendix C, Table C-2, “4.16 kV Mobile Generator Electrical Loads,” provide load information and total load with a 10 percent margin for the 4.16 kV mobile Generator used during Phase 3. Based on the total loads, the mobile Generator has adequate capacity to meet the load requirements. The staff also determined that the mobile generator is adequately sized during an audit on May 25, 2016, and documented its findings in a report dated December 7, 2016 (ML16300A205). Generator fuel is supplied from offsite for long-term coping and COL 19.3(3) is provided for the COL applicant to develop the details for offsite resources including fuel oil for the Generator. DCD, Tier 2, Sections 19.3.2.3.1.1, “Full-Power Operation,” and 19.3.2.3.1.2, “Low-Power Operation,” also states that fuel oil for the 4.16 kV mobile generator and 480 V mobile GTGs will be refilled from offsite resources.

Technical Report, Table 5-9, “Conformance with NEI 12-06, Rev. 0,” states that the appropriate standard electrical connections need to be specified and the COL applicants are responsible to establish a means to ensure the necessary resources are available from offsite. Table 5-9 (page 8 of 20) of the Technical Report also states that connections for primary and secondary FLEX pumps, and mobile Generators, are provided on the outside of the exterior wall of the auxiliary building, thereby providing reasonable assurance of the accessibility of personnel and equipment.

Section 5.1.2.6.1.1 of the Technical Report stated that there are provisions to connect the GTGs that is incorporated into the APR1400 design. In RAI 420-8482, Question 19.03-36 (ML16060A444) the staff asked the applicant to provide discussion about GTG connections and COL item the COL applicant to ensure that the appropriate connection points for the electrical equipment, including voltages and classification. In response to RAI 420-8482, Question 19.03-36, dated April 29, 2016 (ML16120A173) and supplemental responses dated November 17, 2017 (ML17322A046), June 4, 2018 (ML18155A378), and July 9, 2018 (ML18190A160), the applicant included COL 19.3(7) for the COL applicant to ensure the appropriate connection points are provided the electrical equipment including locations, voltages, and classification. As it pertains to the generator connections, the applicant explained that those connections for the mobile generator, are provided on the outside of the exterior wall of the auxiliary building, thereby providing reasonable assurance of the accessibility of personnel and equipment. In addition, the applicant explained that the electrical connections, power cables between the connection boxes and incoming circuit breakers of the Class 1E buses are designed as permanent installations and the temporary cables will be connected between the mobile Generator and connection boxes according to the APR1400 FLEX strategies, as discussed in the Technical Report. The revised DCD Tier 2, Figure 8.1-1, "Electric Power System Single Line Diagram," in the response to RAI 61-7984, Question 08.03.01-5 (ML15251A244 and ML18155A406), also provided locations for the generator connections, and thus, the staff finds that provisions to connect the generator are incorporated into the APR1400 design.

The staff finds that the response to RAI 420-8482, Question 19.03-36 is acceptable because the applicant included COL 19.3(7) for the COL applicant to ensure the appropriate connection points are provided for the electrical equipment including locations, voltages, and classification. Based on the review of the DCD, the staff has confirmed incorporation of the changes described above; therefore, RAI 420-8482, Question 19.03-36, Part 1, is resolved and closed.

The APR1400 electrical design supports Phase 3 of the mitigation strategies for BDBEE with one - 5,000 kW, 4.16 kVac mobile generator and trains A and B Class 1E batteries. During Phase 3, the 4.16 kV mobile Generator provide power to the trains A or B Class 1E 4.16 kV switchgear, which feed the 480V MCCs, shutdown cooling pump, essential I&C equipment, battery chargers, inverters, transformers, and 120 Vac distribution panels. The load requirements are described in Technical Report, Appendix C, Table C-2, "4.16 kV Mobile Generator Electrical Loadings," and includes loads for the trains A and B Class 1E battery chargers.

The 4.16 kV trains A and B systems provide electric power for equipment needed to maintain core cooling, spent fuel pool cooling, and containment heat removal in Phase 3. The trains A and B Class 1E batteries are recharged during Phase 3 and are sized to support the required loads for 2 hours without load shedding, and an additional 6 hours with load shedding. As discussed in the Technical Report, in Phase 3, offsite resources including a 4.16 kV mobile Generator, fuel, and cooling water can be assumed to be available for long-term coping with the BDBEE. The 4.16 kV mobile Generator is used to restore train A or B of 4.16 kV Class 1E power system. As discussed in the Technical Report, the plant is brought to cold shutdown, using the shutdown cooling system (SCS) if the ultimate heat sink (UHS) is available after 4.16 kV Class 1E power is restored. If not, the plant is maintained at the same safe shutdown state (i.e. hot shutdown or cold shutdown) as in Phase 2. The 4.16 kV mobile Generator and 4.16 kV systems are designed to meet the Phase 3 load requirements to maintain the reactor in a safe condition and is described in the Technical Report, DCD Tier 2 Section 19.3, and RAI responses discussed above. The applicant has COL 19.3(7), which states that the COL applicant is to provide site-specific details of the electrical connection points (including locations, voltage level, and electrical classification) for the 480 V mobile GTGs and the 4.16 kV mobile generators.

NEI 12-06, Revision 0 guidance explains Phase 3 coping would have provisions for obtaining additional capability and redundancy from offsite equipment until power, water, and coolant injection systems are restored. The applicant provided COL 19.3(3) and 19.3(7), which are evaluated above, to ensure that offsite resources will be available to support Phase 3 of a BDBEE. Therefore, the staff finds that the APR1400 electrical design can support Phase 3 of mitigation strategies for BDBEE and is consistent with NEI 12-06, Revision 0 guidance which relies on offsite equipment and fuel for Phase 3.

The staff finds that the APR1400 power supply design is consistent with the guidance in JLD-ISG-2012-01, Revision 1, for Phases 1, 2 and 3 of mitigation strategies for BDBEE.

19.3.1.4.9 Shutdown and Refueling Analyses

In the initial review of the APR1400 DC application, the NRC staff followed the guidance for satisfying the Commission directives regarding BDBEE mitigation strategies in Japan Lesson-Learned Project Directorate, JLD-ISG-2012-01, Revision 0, which endorsed with clarifications the methodologies described in NEI 12–06, Revision 0. The guidance in JLD-ISG-2012-01 describes one acceptable approach for satisfying the Commission directives regarding BDBEE mitigation strategies (i.e., Order EA-12-049). In NEI 12-06, Revision 0, Table 3-2, PWR FLEX Baseline Capability Summary, for Core Cooling, the Safety Function states, "RCS Inventory Control and Core Heat Removal (shutdown modes with steam generators (SGs) not available)" the baseline capability includes:

- Diverse makeup connections to the RCS for long-term RCS makeup and shutdown mode heat removal.
- Source of borated water
- Letdown path if required.

In APR1400 Fukushima Technical Report, Section 5.1.2.3.2.2, FLEX Strategy for Mode 4 and Mode 5 with SGs Available, the strategy includes RCS heat up and pressurization to hot standby conditions so that the full power core cooling strategy can be employed. Specifically, after the RCS temperature increases to the Low Temperature Overpressure Protection (LTOP) disable temperature (136.11 °C [277 °F]), the operator must manually isolate the RCS from the SCS by manually closing the SCS isolation valves. The operator must complete this action before the RCS temperature exceeds the SCS entry temperature 176.67 °C (350 °F). After that, a postulated RCS over-pressurization can be protected by pilot-operated safety relief valves (POSRVs).

On February 2, 2016, the staff issued RAI 393-8432, Question 19.03-13 (16033A321). In Question 19.03-13, the staff asked additional information to ensure that the applicant's strategy is feasible. Specifically, the staff asked what alarms indicate that the LTOP disable temperature has been reached to alert the operators to isolate the SCS manually. The staff also requested: (1) the time required for the operators to manually close the SCS isolation valves, (2) the number of operators necessary to perform the task, and (3) any necessary equipment to perform the task. The staff requested that this information be documented in Section 19.3 of the DCD.

On March 9, 2017, the applicant provided their final response to RAI 393-8432, Question 19.03-13 (ML17068A119) to describe the core cooling strategy for Mode 4 and Mode 5 with the SGs Available. The applicant updated DCD, Subsection 19.3.2.3.1.2 and the Technical Report, to describe the core cooling strategy for Mode 4 and Mode 5 with the SGs Available. Should an ELAP concurrent with LUHS occur during these Modes, the RCS is anticipated to heat and pressurize due to the loss of the SCS. If the RCS temperature is initially below the maximum RCS temperature requiring LTOP, i.e., 136.11 °C (277 °F), the RCS pressure can be maintained below the LTOP protection limiting pressure of 43.94 kg/cm2A (625 psia) by LTOP valve operation. After the RCS temperature increases to the LTOP disable temperature, the operator can isolate the RCS from the SCS by manually closing the SCS isolation valves. There are three SCS isolation valves at each train, and these valves are arranged in series. One of the SCS isolation valves is backed up by a battery which can be manually closed in the MCR. This operator action can be completed before the RCS temperature exceeds the SCS entry temperature 176.67 °C (350 °F) (approximately 4 hours after event initiation). There is no additional equipment needed to perform this task. After closing of the SCS isolation valves and the plants returns to hot standby conditions, the SG side feed-and-bleed operation can start cooling down the RCS, as described in the baseline cooling capability for ELAP and LUHS at full-power operation. This update to the DCD resolves Question 1 in RAI 393-8432, Question 19.03-13. Based on the review of the DCD to verify that it incorporated the text as shown in the RAI response, the staff has confirmed incorporation of the changes described above; therefore, to RAI 393-8432, Question 19.03-13, is resolved and closed.

In Fukushima Technical Report, Section 5.1.2.3.3 FLEX Strategy for Shutdown Operation with SGs Not Available, in Phase 1, the safety injection tanks (SITs) are used as a water source for gravity feed to the RCS. On February 2, 2016, the staff issued RAI 393-8432,

Question 19.03-13 (ML16033A321). In Question 19.03-13, the staff asked additional information to ensure that this core cooling strategy is feasible. The staff requested additional information regarding how the safety injection tanks (SITs) can keep the core covered assuming the RCS is vented via the pressurizer given possible pressurizer surge line flooding. Surge line flooding following an extended loss of DHR may negate the elevation head necessary for SIT flow. Based on the APR1400 Shutdown Evaluation Report, the staff understands "With the earliest nozzle dam installation occurring at 4 days after shutdown, the decay heat present would require approximately 481 L/min (127 gpm)." In addition, the staff requested additional information regarding: (1) the number of operators, (2) the time required, and (3) the equipment required for the operators to manually open the SIT isolation valves. The staff also requested additional information concerning what alarms and instrumentation will be used to verify core coverage. The staff also asked for additional information on the impact of boiling through the pressurizer manway on the accuracy of the level indication, including the midloop ultrasonic indication. The staff asked that the requested information be documented in Section 19.3 of the DCD.

On March 9, 2017, the applicant provided their final response to RAI 393-8432, Question 19.03-13 (ML17068A119), to describe the shutdown core cooling strategy with the SGs not available. The applicant updated the DCD, Subsection 19.3.2.3.1.3, Shutdown Conditions with the SGs Not Available. During Phase 1, decay heat is removed by the latent heat resulting from water boil off in the core. In Phase 2, the plant can be maintained at cold shutdown by the RCS feed-and-bleed operation using the one of two primary side low head FLEX pumps which can be connected to a SIS injection line. This connection point is outside of containment. The primary side low-head FLEX pump has a rated flow of 2,839.06 L/min (750 gpm), which is sufficient capacity for removing decay heat and flushing the RCS. Prior to core uncovery, the primary side low-head FLEX pump must be aligned to take suction from the acceptable coolant source and deliver the coolant to the vessel. A mobile GTG is used to connect to train A or train B 480 V Class 1E ac power system to supply power to Class 1E battery. Two primary low-head FLEX pumps are provided to meet the N+1 requirement. The applicant added a COL requirement for the COL applicant to address the details of storage location for FLEX equipment (COL 19.3(4)). The applicant also added a COL requirement for the COL applicant to develop shutdown risk processes and procedures, and verify the ability to deploy FLEX equipment to provide core cooling and containment cooling in shutdown operation with SGs not available per Section 3.2.3 of NEI 12-06, Revision 2, consistent with operating plants (COL 19.3(17)). The addition of the two COL requirements into the DCD and the updates to the DCD and the Technical Report, resolves Question 2 in RAI 393-8432, Question 19.03-13. Based on the review of the DCD to verify that it incorporated the text as shown in the RAI response, the staff has confirmed incorporation of the changes described above; therefore, to RAI 393-8432, Question 19.03-13, is resolved and closed.

In Technical Report, Section 5.1.2.3.3 FLEX Strategy for Shutdown Operation with SGs Not Available, in Phase 2, the plant is expected to be maintained at cold shutdown by RCS feed-and-bleed operation using the FLEX pump. Decay heat is removed by boil off from the core, while the steam generated from the core is released through the pressurizer manway. In this feed-and-bleed operation, the RCS is expected to be maintained at the initial boron concentration because the rate of unborated water injection is expected to be balanced with the rate of steam discharge. The rate of injection flow is expected to be controlled to maintain the RCS water level between the core top and the hot leg center line.

On February 2, 2016, the staff issued RAI 393-8432, Question 19.03-13 (16033A321). In Question 19.03-13, the staff asked additional information to ensure that this phase 2 core cooling strategy is feasible. Specifically, the staff requested the following additional information: (1) confirm whether additional alarms and instrumentation will be used to maintain RCS level between the top of the core and the hot leg centerline beyond what is needed in Phase 1, (2) document the impact of boiling through the pressurizer manway on the accuracy of the additional indication or instrumentation, and (3) document in Section 19.3 of the DCD the plant impact if the operators raise RCS level above midloop conditions.

Based on staff review of NEI 12-06 Table 3-2, PWR FLEX Baseline Capability Summary, for Core Cooling, the Method states, "All Plants Provide Means to Provide Borated RCS Makeup." In RAI 393-8432, Question 19.03-13 (ML16033A321), the staff asked the applicant to justify in Section 19.3 of the DCD why their phase 2 approach is acceptable, since the applicant plans to inject with unborated water.

On March 9, 2017, the applicant provided their final response to RAI 393-8432, Question 19.03-13 (ML17068A119), to describe the shutdown core cooling strategy COL. In response to this RAI, as discussed above, the applicant added a COL requirement for the COL applicant to develop shutdown risk processes and procedures, and verify the ability to deploy FLEX equipment to provide core cooling and containment cooling in shutdown operation with SGs not available per Section 3.2.3 of NEI 12-06, Revision 2, consistent with operating plants (COL 19.3(17)).

The applicant also updated the DCD Subsection 19.3.2.3.1.3 to include indications of Core Coverage. The operator can monitor the RCS level in MCR via flat panel display (FPD) from the qualified indication and alarm system (QIAS-P). Coolant level during reduced inventory operations is measured by the permanent refueling water level indication system (PRWLIS), the local refueling water level indication system (LRWLIS), and the ultrasonic level measurement system (ULMS). The PRWLIS consists of the wide range (WR) level instrument and the narrow range (NR) level instrument. Each of two trains provides the means of monitoring water level of each RCS loop to the MCR during reduced RCS inventory operations. The PRWLIS (WR) indicates coolant level between 10 percent level of the pressurizer and the bottom of the hot leg. It provides level indication to the MCR without alarm. The PRWLIS (NR) indicates coolant level between the top of the hot leg and 2 in above the bottom of the hot leg. It provides level indication and Low, Low-Low and High alarms to the MCR. Core exit temperature (CET) also can be monitored on FPD and the operator can verify core coverage by monitoring changes of this value. QIAS-P has a high CET alarm.

The addition of the COL requirement (COL 19.3(17)) into the DCD and the proposed update to the DCD on indications core coverage, resolves Question 3 in RAI 393-8432, Question 19.03-13. Based on the review of the DCD to verify that it incorporated the text as shown in the RAI response, the staff has confirmed incorporation of the changes described above; therefore, to RAI 393-8432, Question 19.03-13, is resolved and closed.

19.3.1.4.10 Protection of Equipment

The NRC Order EA-12-049 requires equipment credited for FLEX mitigation strategies be protected from applicable external events. NRC Commission Paper SECY-12-0025 states that the NRC staff expects new reactor DC applicants to address the Commission-approved Fukushima actions in their applications to the fullest extent practicable. NEI 12-06, Section 3, provides guidelines to establish plant's baseline coping capability based on an assumption that installed equipment designed to be robust with respect to design-basis external events is fully available during a simultaneous ELAP and LUHS event. Specifically, NEI 12-06, Section 3.2.1.3, addresses a set of initial conditions that should be taken into account in establishing plant-specific baseline coping capability. The initial conditions include those pertaining to the storage and protection of installed plant equipment.

The APR1400 DCD Tier 2 Section 19.3.2.3 describes that APR1400 FLEX strategies ensure reasonable protection of the equipment used in FLEX strategies from applicable external events. APR1400-E-P-NR-14005, Section 5.1.2.2 states that guidance for developing, implementing, and maintaining mitigation strategies from JLD-ISG-2012-01 and methodology to establish baseline coping capability from NEI 12-06 are considered in developing the APR1400 FLEX strategies and in evaluating the resultant baseline coping capability for a BDBEE. However, the staff found that neither APR1400 DCD, Revision 0, Tier 2 Section 19.3, APR1400-E-P-NR-14005, Revision 0, provided sufficient information concerning the robustness and structural capacity of the structures and tanks that are credited to provide protection and storage for mitigation equipment and cooling inventories. In order to complete an evaluation of APR1400 FLEX strategies with respect to the credited structures and tanks, the staff issued an RAI to the applicant.

In RAI 354-8416, Question 19.03-10 (ML15357A220), the NRC staff requested the applicant to provide a description of the structures credited for protection of permanent plant equipment identified for APR1400 mitigation strategies. The description should include classifications, locations, and structural functions credited for protection. If a structure that houses such equipment is not classified as safety-related, the applicant was requested to demonstrate that the design of the structure is robust enough to withstand the effect of an applicable external event including seismic events, floods, and high winds and associated missiles so that the permanent plant equipment credited for APR1400 mitigation strategies remains protected from the event. The staff also requested the applicant to provide a description of the tanks credited for protection and storage of cooling and makeup water inventories or fuel for equipment identified for APR1400 mitigation strategies. The description should include classifications, locations, and structural functions credited for protection and storage. If such tanks are not safety-related, the applicant was requested to demonstrate that the design of these tanks and the structure that houses tanks are robust enough to withstand the effect of an applicable external event so that the tanks remain functional for the protection of the cooling and makeup water inventories or fuel for equipment credited for APR1400 mitigation strategies.

The applicant submitted a response to RAI 354-8416, Question 19.03-10, on April 16, 2016 (ML16107A052). The staff reviewed the information provided in the RAI response and found that the applicant did not fully address the issues raised by the staff including information pertaining to the BAST and RWT. Incorporating staff's feedback, the applicant submitted a revised response to RAI 354-8416, Question 19.03-10, on June 28, 2016 (ML16180A286). In its response, the applicant indicated that all installed equipment used in APR1400 mitigation strategies are housed inside the reactor containment building, auxiliary building, essential service water building, emergency diesel generator building, or component cooling water heat exchanger building, all of which are safety-related structures and are designed for applicable external hazards including seismic events, floods, high winds and associated missiles. The applicant also indicated that the specific location, function and classification of these structures are described in DCD Tier 2 Table 3.2-1. The applicant stated that APR1400 does not have installed equipment housed in non-safety-related structures. The applicant also indicated that tanks utilized in the mitigation strategies include AFWST, IRWST, SIT, EDG Fuel Oil Storage Tanks, EDG Fuel Oil Day Tanks, and BAST and that these tanks are all safety-related and their location, function, and safety classification are provided in DCD Tier 2 Table 3.2-1.

Additionally, the applicant indicated that the RWT is utilized for APR1400 mitigation strategies and is designed to seismic Category I. The applicant stated that the COL applicant will confirm and ensure that the RWT and its flow path to the FLEX equipment (structures, piping, components, and connections) are designed to be robust with respect to applicable hazards including seismic events, floods, high winds, and associated missiles. To this effect, the applicant proposed a new COL Information Item, COL 19.3(6), and provided the following markup that will be incorporated into the next revision of APR1400 DCD Tier 2:

> COL 19.3(6) The COL applicant is to confirm and ensure that the raw water tank and flow path to the FLEX equipment (structures, piping, components, and connections) are designed to be robust with respect to applicable hazards (e.g., seismic events, floods, high winds, and associated missiles).

The applicant indicated that Table 1.8-2 and Section 19.3.4 of APR1400 DCD Tier 2 will be revised to include this new COL Information Item, COL 19.3(6). The applicant also provided the following markup that will be incorporated into Section 19.3.2.3.4 of APR1400 DCD Tier 2:

> All permanent installed, safety related equipment (pumps, valves, etc.) that is utilized in the mitigation strategies for BDBEE are housed inside the reactor containment building, auxiliary building, essential service water/component cooling water heat exchanger building, emergency diesel generator building. All of these structures are safety related and are designed for seismic, flood, high wind and missile. The specific location, function, and classification of these structures are described in DCD Tier 2 Table 3.2-1.
>
> The specific tanks (AFWST, IRWST, SIT, EDG fuel oil storage tank, EDG fuel oil day tank, BAST) are utilized in the mitigation strategies. All of these tanks are safety related, seismic Category I, and Quality Group C. DCD Tier 2 Table 3.2-1 provides the location, function, and safety classifications for these tanks. Additionally, RWT is utilized for mitigating strategies and is designed to seismic Category I and Quality Group D. It will remain functional for the mitigating strategies.
>
> The detailed design of the raw water tank related with site specific data is the responsibility of the COL applicant. The COL applicant is to confirm, satisfy, or fulfill the specific design functional requirements of raw water tank including the associated instrument, capacity, location, flow path to on-site, the valve pit connected to FLEX equipment, and any other design features as described in DCD Section 19.3 in support of BDBEE mitigation strategies (COL 19.3(5)). The COL applicant is to confirm and ensure that the raw water tank and flow path to the FLEX equipment (structures, piping, components, and connections) are designed to be robust with respect to applicable hazards (e.g., seismic events, floods, high winds, and associated missiles) (COL 19.3(6)).

The applicant further indicated that a new section (Section 6.2.10) will be added to KHNP report ARP1400-E-P-NR-14005 that includes information similar to that provided in the above markup for APR1400 DCD Tier 2 Section 19.3.2.3.4.

Based on the above review, the staff finds the applicant's response to RAI 354-8416, Question 19.03-10, acceptable because the structures and tanks credited to provide protection and storage for installed mitigation equipment and cooling inventories are designed to be robust in accordance with the guidelines in NEI 12-06. The staff confirmed that the RAI response has been incorporated into Revision 1 of the DCD and Revision 1 of the Technical Report, ARP1400-E-P-NR-14005. Therefore, the staff considers RAI 354-8416, Question 19.03-10, to be resolved and closed.

In selecting or designing structures that will house plant equipment credited for APR1400 mitigation strategies, potential seismic interaction effects between these structures with adjacent structures that are not seismically robust should be assessed to determine whether the failure of a seismically non-robust structure will impair the integrity of a structure that houses plant equipment credited for APR1400 mitigation strategies. However, the staff found that neither APR1400 DCD, Revision 0, Tier 2 Section 19.3, nor APR1400-E-P-NR-14005, Revision 0 provided information that addresses such potential seismic interaction effects.

In RAI 354-8416, Question 19.03-11 (ML15357A220), the NRC staff requested the applicant to provide an assessment of potential seismic interaction effects between the structures that house permanent plant equipment credited for APR1400 mitigation strategies and any adjacent structures that are not seismically robust. The staff also requested the applicant to provide an assessment of potential seismic interaction effects between non-seismically robust structures with standalone tanks that contain cooling and makeup water inventories or fuel for equipment credited for APR1400 mitigation strategies. The applicant was requested to provide information which demonstrates that the failure of a non-seismically robust structure will not affect the integrity of the structures that house permanent plant equipment credited for APR1400 mitigation strategies or the function of the standalone tanks that contain cooling and makeup water inventories or fuel for equipment credited for APR1400 mitigation strategies.

In its response to RAI 354-8416, Question 19.03-11, dated April 7, 2016 (ML16098A369), the applicant indicated that the structures and tanks outside the auxiliary building (AB) that APR1400 credited for mitigation strategies include (1) the FLEX equipment storage building(s) that stores FLEX pumps and associated connection components; (2) the emergency diesel generator building that stores fuel oil storage tanks and fuel oil day tanks which contain fuel oil for FLEX strategies; (3) the boric acid storage tank that stores borated water for RCS makeup; and (4) the raw water tank that provides water source for the FLEX strategies. The applicant stated that these features are designed as seismic Category I and protected reasonably from applicable external events. The applicant also stated that, in terms of the interaction effect between these structures and tanks with adjacent structures that are not designed to be seismically robust, the COL applicant is to confirm that any site-specific non-seismic Category I SSCs are designed not to degrade the function of a seismic Category I SSC to an unacceptable safety level due to their structural failure or interaction in accordance with COL Information Item, COL 3.7(4) addressed in DCD Tier 2 Subsection 3.7.2.8.

Based on the above review, the staff finds the applicant's response to RAI 354-8416, Question 19.03-11, acceptable because the COL applicant will ensure that no detrimental seismic interaction effects between seismically non-robust structures and structures or tanks credited for APR1400 mitigation strategies will occur, by fulfilling COL 3.7(4). Therefore, the staff considers RAI 354-8416, Question 19.03-11, to be resolved and closed.

APR1400 DCD Tier 2 Section 19.3.4 specifies COL information items that pertains to the mitigation strategies for BDBEEs. The staff noted that COL 19.3(4), specifies that the COL applicant is to address the details of storage location for FLEX equipment. NEI 12-06 provides guidance for selecting suitable storage locations that provide reasonable protection for FLEX equipment during a specific external event. Because reasonable protection for FLEX equipment during an applicable external event is a key consideration in selecting storage locations, in RAI 354-8416, Question 19.03-12 (ML15357A220), the NRC staff requested that the applicant expand the COL 19.3(4) in such a way that it highlights "reasonable protection" for FLEX equipment in selecting its storage locations.

In its response to RAI 354-8416, Question 19.03-12, dated April 16, 2016 (ML16107A052), the applicant indicated that COL 19.3(4) will be expanded to address the reasonable protection of FLEX equipment in selecting storage locations. The revised COL 19.3(4) is included in the list of COL Information Items in Section 19.3.1.5 of this report.

The applicant also indicated that DCD Tier 2 Table 1.8-2, Section 19.3.2.3.4, and Section 19.3.4 will be revised correspondingly in order to reflect the changes made to COL 19.3(4) and provided corresponding markups, which were reviewed and accepted by the staff.

The staff finds the applicant's response to RAI 354-8416, Question 19.03-12 acceptable because the relevant COL Information Item will be revised to adequately address the reasonable protection of FLEX equipment in selecting storage locations. The staff confirmed that the RAI response has been incorporated into Revision 1 of the DCD and, therefore, considers RAI 354-8416, Question 19.03-12, to be resolved and closed.

19.3.1.4.11 Mechanical Equipment Capability

The NRC regulations in 10 CFR Part 50, Appendix A, GDC 1, "Quality standards and records," require that the "[s]tructures, systems, and components important to safety … be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." In Order EA-12-049, the Commission specified that licensees and COL holders establish a three-phase approach for mitigating beyond-design-basis external events.

Section 6.2, "Equipment Quality," of JLD-ISG-2012-01 accepts NEI 12-06 with additional provisions for the quality of equipment used to respond to beyond-design-basis external events.

In Subsection 3.2.1.12, "Qualification of Installed Equipment," NEI 12-06 states that equipment relied upon to support the FLEX implementation does not need to be qualified to all extreme environments that may be posed, but some basis should be provided for the capability of the equipment to continue to function. In Section 11.2, "Equipment Design," NEI 12-06 states in Item 1 that design requirements and supporting analyses should be developed for portable equipment that directly performs a FLEX mitigation strategy for the core cooling, containment and SFP cooling capabilities. This information provides the inputs, assumptions, and documented analyses to show that the mitigation strategy and support equipment will perform as intended. Footnote 3 in NEI 12-06 states that the FLEX documentation should be auditable, but the 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," provisions are not required, and that manufacturer information may be used to establish the basis for the equipment use. In Section 11.2 of NEI 12-06, Item 3 notes that the functionality of the equipment may be outside of the manufacturer's specifications if justified in a documented engineering evaluation. In Section 11.5, "Maintenance and Testing," NEI 12-06 indicates in Item (1) that FLEX mitigation equipment should be initially tested; or other reasonable means used to verify that equipment performance conforms to the limiting functions on which the FLEX strategies and guidelines rely. Item (1) notes that validation of source manufacturer quality is not required. Section 11.5 in NEI 12-06 also states that portable equipment that directly performs a FLEX mitigation strategy for the core cooling, containment, or

SFP cooling capabilities should be subject to maintenance and testing (including surveillances and inspections) to verify proper function.

To evaluate mechanical equipment capability, the NRC staff reviewed APR1400 DCD Tier 2 Section 19.3, which describes the mitigation strategies to manage and mitigate external events that are beyond the design basis of the APR1400 nuclear power plant. Section 19.3 states that it addresses the conformance of the APR1400 design with SECY-12-0025 and Commission Order EA-12-049, and other related documents. In addition, the staff reviewed APR1400-E-P-NR-14005, Revision 2, referenced in APR1400 DCD Tier 2 Section 19.3.

In RAI 297-8309, Question 19.03-1 (ML15314A022), the NRC staff requested that the APR1400 DC applicant describe the performance requirements as part of the mitigation strategies (including initial full-power operation and mid-loop operation) to ensure core cooling, containment, and SFP cooling capabilities during a BDBEE at an APR1400 nuclear power plant as follows:

1. All safety-related installed pumps, valves, and dynamic restraints that will be used at the APR1400 plant as part of the mitigation strategies for an ELAP event;
2. All nonsafety-related installed pumps, valves, and dynamic restraints that will be used at the APR1400 plant as part of the mitigation strategies for an ELAP event; and
3. All portable or FLEX flow systems (including pumps, valves, and dynamic restraints) that will be used at the APR1400 plant as part of the mitigation strategies for an ELAP event.

In response to RAI 297-8309, Question 19.03-1, dated November 9, 2015 (ML15314A022), the applicant provided modifications to APR1400-E-P-NR-14005 and APR1400 DCD Tier 2 to address the performance requirements for components used as part of the mitigation strategies. For example, the applicant provided a table of installed safety-related pumps and valves used during BDBEEs that references the specific DCD sections for performance requirements. The applicant noted that the mitigating strategies to address BDBEEs do not include any dynamic restraints in the piping design. The applicant referred to APR1400-E-P-NR-14005 for installed nonsafety-related valves used for various functions, including the SFP external makeup and spray water, and the Emergency Containment Spray Backup System (ECSBS). The applicant reported that there are no installed nonsafety-related pumps used to mitigate BDBEEs in the APR1400 design. The applicant provided references to portable or FLEX equipment used to mitigate BDBEEs in APR1400-E-P-NR-14005. In response to RAI 297-8309, Question 19.03-1, dated November 9, 2015 (ML15314A022), as modified by its revised response to RAI 393-8432, Question 19.03-13, dated December 2, 2016 (ML16033A321), the applicant provided modifications to APR1400-E-P-NR-14005 to clarify the performance requirements for the FLEX equipment. The applicant noted that the COL applicant is responsible for determining the final FLEX pump design head for site conditions. In its proposed DCD markup, the applicant clarified the information to be developed by the COL applicant to address this COL Information Item regarding the FLEX pumps. In its response to RAI 333-8397, Question 19.03-9, dated July 6, 2016 (ML15348A114), the applicant described modifications to APR1400-E-P-NR-14005 to describe the functional requirements for on-site and offsite FLEX equipment, including the Primary side high-head and low-head FLEX pumps, Secondary side FLEX pump, SFP Makeup FLEX pump, SFP Spray FLEX pump, and ECSBS FLEX pump. In its response to RAI 393-8432, Question 19.03-13, dated March 9, 2017 (ML17068A121), the applicant described updated modifications to the DCD and APR1400-E-P-NR-14005 regarding the FLEX equipment. The NRC staff found acceptable the applicant's described modifications to APR1400-E-P-NR-14005 and the modified DCD sections submitted in response to RAIs 19.03-1, 9, and 13 to specify the performance requirements necessary to provide reasonable assurance of the capability of pumps and valves used as part of the mitigation strategies for BDBEEs in the APR1400 design. The staff tracked RAI 297-8309, Question 19.03-1, as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD and APR1400-E-P-NR-14005. Revision 3 to the DCD renumbers the COL items with this item identified as COL 19.3(9). Therefore, the staff considers RAI 297-8309, Question 19.03-1, to be resolved and closed.

In RAI 297-8309, Question 19.03-2 (ML15314A022), the NRC staff requested that the APR1400 DC applicant describe the provisions for design, manufacture, testing, installation, and surveillance to provide assurance of the seismic, environmental, and functional capability of all safety-related installed pumps, valves, and dynamic restraints to perform their intended functions as part of the mitigation strategies (including initial full-power operation and mid-loop operation) to ensure core cooling, containment, and SFP cooling capabilities during an ELAP event at an APR1400 nuclear power plant. As part of this RAI, the staff requested that the applicant indicate whether any safety-related pumps, valves, and dynamic restraints used as part of the mitigation strategies for an ELAP event will have performance requirements that exceed their original safety-related design and performance specification (such as pumps used with reduced net positive suction head available, and safety or relief valves used to support feed and bleed conditions). In addition, the staff requested that the applicant indicate where the APR1400 DCD Tier 2 specifies the provisions for the design, manufacture, testing, installation, and surveillance for the safety-related pumps, valves, and dynamic restraints that perform functions as part of the mitigation strategies, or provide proposed modifications to the APR1400 DCD Tier 2 to incorporate these provisions.

In response to RAI 297-8309, Question 19.03-2, dated May 30, 2016 (ML16151A007), the applicant described the provisions to ensure the capability of safety-related components used as part of mitigation strategies for BDBEEs in the APR1400 design. For example, the applicant indicated that the safety-related installed pumps and valves used as part of the mitigation strategies for BDBEEs are listed in Table 6-2, which will be included in APR1400-E-P-NR-14005 in response to RAI 297-8309, Question 19.03-1. The applicant stated that the performance requirements for the safety-related pumps and valves for the BDBEE mitigation strategies are bounded by the performance requirements of the original APR1400 design basis and performance specifications for these pumps and valves. The applicant noted that Table 6-2 of APR1400-E-P-NR-14005 will provide cross references to the applicable DCD sections for the provisions of design, manufacture, testing, installation, and surveillance of safety-related pumps and valves. The NRC staff found that the applicant’s response to RAI 297-8309, Question 19.03-2, together with the applicant’s described modifications to APR1400-E-P-NR-14005 discussed in the response to RAI 297-8309, Question 19.03-1, as modified by the responses to RAI 333-8397, Question 19.03-9, and RAI 393-8432, Question 19.03-13 would specify in an acceptable manner the provisions for the capability of safety-related pumps and valves to perform their functions as part of mitigation strategies for BDBEEs in the APR1400 design. The staff tracked the resolution of RAI 297-8309, Question 19.03-2, as a confirmatory item. With respect to mechanical equipment capability discussed in this subsection, the staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD and APR1400-E-P-NR-14005. Therefore, the staff considers RAI 297-8309, Question 19.03-2, and to be resolved and closed.

In RAI 297-8309, Question 19.03-3, the NRC staff requested that the APR1400 DC applicant describe the provisions for design, manufacture, testing, installation, and surveillance to provide assurance of the seismic, environmental, and functional capability of all nonsafety-related installed pumps, valves, and dynamic restraints to perform their intended functions as part of the mitigation strategies (including initial full-power operation and mid-loop operation) to ensure core cooling, containment, and SFP cooling capabilities during an ELAP event at an APR1400 nuclear power plant. In addition, the staff requested that the applicant indicate where the APR1400 DCD Tier 2 specifies the provisions for the design, manufacture, testing, installation, and surveillance for the nonsafety-related installed pumps, valves, and dynamic restraints that perform functions as part of the mitigation strategies, or provide proposed modifications to the APR1400 DCD Tier 2 to incorporate these provisions.

In response to RAI 297-8309, Question 19.03-3, initially on March 4, 2016 (ML16064A359), and revised on September 6, 2016 (ML16250A149), the applicant described the provisions to ensure the capability of nonsafety-related components used as part of mitigation strategies for BDBEEs in the APR1400 design. In particular, the applicant referenced its responses to RAIs 297-8309, Questions 19.03-1 and 4. The applicant referred to its response to RAI 297-8309, Question 19.03-4, with a revision dated September 6, 2016 (ML16250A173), for information on the design, manufacture, testing, and installation of installed nonsafety-related equipment used as part of the mitigation strategies for BDBEEs. For example, the applicant stated in its revised response to RAI 19.03-4 that it planned to revise DCD Tier 2 Section 19.03, to specify that the SFP Spray Pump, SFP Makeup Pump, ECSBS Pump, Primary & Secondary side Pumps will be designed, manufactured, tested and installed in accordance with applicable commercial codes and standards such as the hydraulic institute pump standard, and with the design, storage, maintenance, and testing as outlined in NEI 12-06. In its revised response to RAI 19.03-4, the applicant also stated that the DCD would be revised to specify that the installed nonsafety-related in-line valves for the SFP external makeup water and ECSBS are designed to Quality Group D and Seismic Category I requirements. The applicant noted that these in-line valves have no specific regulating performance requirements. The applicant planned to specify that the applicable codes and standards for valves are American Society of Mechanical Engineers (ASME) B31.1, “Power Piping,” and ASME B16.34, “Valves – Flanged, Threaded, and Welding End.” The applicant also clarified that there are no installed nonsafety-related pumps and/or snubbers in the scope of the APR1400 design to mitigate BDBEEs. In its revised response to RAI 19.03-4, the applicant provided modifications to APR1400-E-P-NR-14005 to specify that the SFP spray pump and SFP markup pump will be designed, manufactured, tested and installed in accordance with applicable commercial codes and standards such as the hydraulic institute pump standard, and with the design, storage, maintenance, and testing as outlined in NEI 12-06. The applicant also provided modifications to APR1400-E-P-NR-14005 to specify that the SFP spray pump and SFP makeup pump will not be designed as Seismic Category I, but will be stored in structures which are designed to satisfy 10 CFR Part 50, Appendix A, GDC 2, to ensure meeting functional requirements for external environments such as seismic, flooding, wind, etc. for the specific site. The applicant planned to revise APR1400-E-P-NR-14005 to specify that installed nonsafety-related in-line valves for the SFP external makeup water system are designed as Quality Group D and Seismic Category I requirements with the applicable codes and standards for valves being ASME B31.1 and B16.34.

The NRC staff found that the applicant’s revised responses to RAI 297-8309, Questions 19.03-3 and 19.03-4 together with the applicant’s described modifications to the DCD and APR1400-E-P-NR-14005 specify in an acceptable manner the provisions for the capability of installed nonsafety-related pumps and valves to perform their functions as part of mitigation strategies for BDBEEs in the APR1400 design. The staff tracked RAI 297-8309, Question 19.03-3, as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD and APR1400-E-P-NR-14005. Therefore, the staff considers RAI 297-8309, Question 19.03-3, to be resolved and closed.

In RAI 297-8309, Question 19.03-4, the NRC staff requested that the APR1400 DC applicant describe the provisions for design, manufacture, testing, installation, and surveillance to provide assurance of the seismic, environmental, and functional capability of all portable or FLEX flow systems (including pumps, valves, and dynamic restraints) that are part of the mitigation strategies (including initial full-power operation and mid-loop operation) to ensure core cooling, containment, and SFP cooling capabilities during an ELAP event at an APR1400 nuclear power plant.

In its response to RAI 297-8309, Question 19.03-4, as revised on September 6, 2016 (ML16250A173), the applicant described the provisions to ensure the capability of portable or FLEX equipment used as part of the mitigation strategies for BDBEEs in the APR1400 design. In particular, the applicant stated that NEI 12-06 (Revision 0) in Section 11.0 stipulates the quality attributes and equipment design requirements for FLEX or portable equipment utilized for the mitigating strategies for BDBEEs. The applicant also indicated that installed nonsafety-related equipment would be specified to comply with 10 CFR Part 50, Appendix A, GDC 2, as applicable. The applicant specified that the FLEX or portable equipment will be procured as commercial grade with the design, storage, maintenance, and testing as outlined in NEI 12-06. The applicant stated that the FLEX equipment is not designed as Seismic Category I, but is stored in structures which are designed to satisfy GDC 2 to ensure meeting functional requirements for external environments such as seismic, flooding, wind, etc. for the specific site. The applicant noted that as stated in a COL item, the COL applicant will address the details of selecting suitable storage locations for FLEX equipment that provide reasonable protection during specific external events as provided in NEI 12-06. In its revised response to RAI 297-8309, Question 19.03-4, the applicant provided modifications to APR1400-E-P-NR-14005 to specify that the Primary side and Secondary side FLEX pumps will be designed, manufactured, tested and installed in accordance with applicable commercial codes and standards such as the hydraulic institute pump standard, and with the design, storage, maintenance, and testing as outlined in NEI 12-06. The applicant also planned to revise APR1400-E-P-NR-14005 to specify that the Primary side and Secondary side FLEX pumps will not be designed as Seismic Category I, but will be stored in structures which are designed to satisfy GDC 2 to ensure meeting functional requirements for external environments such as seismic, flooding, wind, etc. for the specific site. The applicant planned to revise APR1400-E-P-NR-14005 to specify that installed nonsafety-related in-line valves for the safety injection system are designed as Quality Group D and Seismic Category I requirements with the applicable codes and standards for valves being ASME B31.1 and B16.34.

The NRC staff found that the applicant's revised response to RAI 297-8309, Question 19.03-4, together with the applicant's described modifications to the DCD and APR1400-E-P-NR-14005, specify in an acceptable manner the provisions for the capability of portable or FLEX pumps and valves to perform their functions as part of mitigation strategies for BDBEEs in the APR1400 design. The staff tracked RAI 297-8309, Question 19.03-4, as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD and APR1400-E-P-NR-14005. Revision 3 to the DCD renumbers the COL items with this item identified as COL 19.3(4). Therefore, the staff considers RAI 297-8309, Question 19.03-4, to be resolved and closed.

In RAI 297-8309, Question 19.03-5, the NRC staff requested that the APR1400 DC applicant provide a description of the operational programs that will provide assurance of the functional capability of the pumps, valves, and dynamic restraints used in the mitigation strategies for ensuring core cooling, containment function, and SFP cooling capabilities during an ELAP event at an APR1400 nuclear power plant. In its response to RAI 297-8309, Question 19.03-5 (ML16113A426) the applicant provided modifications to the APR1400 DCD to include a new COL item specifying that the COL applicant is to provide a description of the operational programs that will provide assurance of the functional capability of the pumps, valves, and dynamic restraints used in the mitigation strategies for ensuring core cooling, containment function, and spent fuel pool cooling capabilities during an extended loss of ac power event at an APR1400 nuclear power plant.

The NRC staff found that the planned COL item would indicate the responsibility of the COL applicant to provide a description of the appropriate operational programs for pumps, valves, and dynamic restraints used in the mitigation strategies during an extended loss of ac power event at an APR1400 nuclear power plant. In a supplemental response to RAI 297-8309, Question 19.03-5, dated June 16, 2016 (ML16168A464), the applicant provided a corrected number for this item. The staff tracked the resolution of RAI 297-8309, Question 19.03-5, as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD. Revision 3 to the DCD renumbers the COL items with this item identified as COL 19.3(11). Therefore, the staff considers RAI 297-8309, Question 19.03-5, to be resolved and closed.

Section 19.3.4, "Combined License Information," lists several COL items that will be addressed by the COL applicant based on site-specific requirements. For example, a COL item states that the COL applicant is to address site-specific strategies to mitigate BDBEEs as specified in NRC Order EA-12-049. In RAI 297-8309, Question 19.03-6, the NRC staff requested that the APR1400 DC applicant establish a COL item for a COL applicant to propose a License Condition to verify the development and implementation of the guidance, strategies, and programs for the mitigation strategies for ensuring core cooling, containment, and SFP cooling capabilities during an ELAP event at an APR1400 nuclear power plant. The staff also requested that the APR1400 DC applicant provide a model license condition in the APR1400 DCD with key elements for the implementation of mitigation strategies for ELAP events, such as found in other DC and COL applications with the applicable NRC safety evaluations. In its response to RAI 297-8309, Question 19.03-6, dated March 18, 2016 (ML16078A315), the applicant provided modifications to the APR1400 DCD Tier 2 to include a COL item for the COL applicant to propose a License Condition to verify the development and implementation of the guidance, strategies, and programs for the mitigation strategies for ensuring core cooling, containment function, and SFP cooling capabilities during an extended loss of ac power event at an APR1400 nuclear power plant. In a supplemental response to RAI 297-8309, Question 19.03-6, dated June 16, 2016 (ML16168A464), the applicant provided a corrected number for this item. The staff tracked RAI297- 8309, Question 19.03-6, as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD. Revision 3 to the DCD renumbers the COL items with this item identified as COL 19.3(10). Therefore, the staff considers RAI 297-8309, Question 19.03-6, to be resolved and closed.

Section 19.3.5, "References," in APR1400 DCD Tier 2 includes APR1400-E-P-NR-14005. APR1400 DCD Tier 2 Section 19.3.1 states that APR1400-E-P-NR-14005 provides specific details for addressing the NRC's Tier 1, 2, and 3 Near-Term Task Force items in Response to the Fukushima accident. Other subsections in Section 19.3 reference APR1400-E-P-NR-14005 in supporting the mitigation strategies. In RAI 297-8309, Question 19.03-7, the NRC staff requested that the APR1400 DC applicant specify whether the APR1400 DCD "incorporates by reference" APR1400-E-P-NR-14005 for implementation of its provisions as requirements under the APR1400 DC, or references APR1400-E-P-NR-14005 as general guidance for use by COL applicants on a voluntary basis. The staff also requested that APR1400 DC applicant revise the DCD as necessary to clarify the intent regarding the reference to APR1400-E-P-NR-14005. In its response to RAI 297-8309, Question 19.03-7, dated May 31, 2016 (ML16152B022), the applicant stated that the APR1400 DCD "incorporates by reference" APR1400-E-P-NR-14005 as indicated in DCD Tier 2 Table 1.6-2 for implementation of its provisions as required under the

APR1400 DC. The NRC staff found the modifications to incorporate by reference APR1400-E-P-NR-14005 as part of the APR1400 DCD to be acceptable to provide specific provisions for the capability of pumps and valves to perform their functions as part of the mitigation strategies for BDBEEs in the APR1400 design. The staff tracked RAI 297-8309, Question 19.03-7, as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD. Therefore, the staff considers RAI 297-8309, Question 19.03-7, to be resolved and closed.

The APR1400 DCD Tier 2 Section 19.3.2.3.1.1, on page 19.3-5 refers to "boric acid storage tanks" when describing the basic operational strategy for core cooling during an ELAP event following full-power operation. Table 5A-3, "Low-Pressure Equipment of the CVCS," in Chapter 5, "Reactor Coolant System and Connecting Systems," in APR1400 DCD Tier 2 lists only one boric acid storage tank (BAST). In RAI 297-8309, Question 19.03-8, the NRC staff requested that the APR1400 DC applicant clarify the DCD and APR1400-E-P-NR-14005 regarding the number of Boric Acid Storage Tanks in the APR1400 design. In its response to RAI 297-8309, Question 19.03-8, dated December 23, 2015 (ML15357A331), the APR1400 DC applicant clarified that there is only one BAST in the APR1400 design. The applicant indicated that DCD Tier 2 Section 19.3.2.3.1.1 and APR1400-E-P-NR-14005, Section 5.1.2.3.1.2.2, would be revised to clarify the APR1400 design, and provided modifications to the DCD and APR1400-E-P-NR-14005. The staff found that the RAI response, including the modifications to the DCD and APR1400-E-P-NR-14005, are acceptable in clarifying that the APR1400 includes one BAST. The staff tracked RAI 297-8309, Question 19.03-8, as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD and APR1400-E-P-NR-14005. Therefore, the staff considers RAI 297-8309, Question 19.03-8, to be resolved and closed.

19.3.1.4.12 Programmatic Controls and Procedures

NEI 12-06, Chapter 11, "Programmatic Controls," summarizes the programmatic controls to be considered for the implementation of the plant-specific FLEX strategies. Chapter 11 discusses quality attributes, equipment design, equipment storage, procedure guidance, maintenance and testing, training, staffing, and configuration controls.

In RAI 297-8309, Question 19.03-6, the NRC staff requested the applicant to propose a model License Condition with key elements to verify the development and implementation of the guidance, strategies, and programs for the mitigation strategies. The applicant responded in the letters dated March 18 and June 16, 2016 (ML16078A315, and ML16168A464). In the responses, the applicant added a COL item in the DCD for a COL applicant to propose a license condition to verify the development of the guidance, strategies, and programs for the mitigation strategies for ensuring core cooling, containment function, and spent fuel pool cooling capabilities.

The staff found the responses to RAI 297-8309, Question 19.03-6, acceptable because COL applicants will address the programmatic control and procedures based on applicable COL items in accordance with NEI 12-06 guidance. The DCD markup associated with RAI 19.03-6 was tracked as a confirmatory item. The staff confirmed that the RAI response markup has been incorporated into Revision 3 of the DCD. Revision 3 to the DCD renumbers the COL items with this item identified as COL 19.3(10). Therefore, the staff considers RAI 297-8309, Question 19.03-6, to be resolved and closed.

19.3.1.4.13 Emergency Plan

DCD Tier 2 Chapter 19.3.2.6 “Recommendation 9.3 – Emergency Plan” describes the design features incorporated into the APR1400 design to enhance emergency preparedness for a Beyond Design Basis External Event. This section is evaluated in Section 13.3.4.5, “NTTF Recommendation 9.3” of this SER.

19.3.1.5 COL Information Items

Revision 3 to the DCD Tier 2 Section 19.3.4 and Table 1.8-2 list the following COL information items for COL applicants to address the mitigation strategies:

Table 19.3-3 Combined License Items Identified in the DCD

Item No.	Description	Section
COL 19.3(1)	The COL applicant is to perform site-specific seismic hazard evaluation and seismic risk evaluation as applicable in accordance with NTTF Recommendation 2.1 as outlined in the NRC RFI.	19.3
COL 19.3(2)	The COL applicant is to address the flood requirements for wet sites.	19.3
COL 19.3(3)	The COL applicant is to develop the details for offsite resources.	19.3
COL 19.3(4)	The COL applicant is to address the details of selecting suitable storage locations for FLEX equipment that provide reasonable protection during specific external events as provided in NEI 12-06 guidance Sections 5 through 9, and the details of the guidance for storage of FLEX equipment provided in the Technical Report (Chapter 19.3, Reference 5) Section 6.2.9.	19.3
COL 19.3(5)	The COL applicant is to confirm, satisfy, or fulfill the specific design functional requirements of raw water tank including the associated instrument, capacity, location, flow path to on-site, the valve pit connected to FLEX equipment, and any other design features as described in DCD Section 19.3 in support of BDBEE mitigation strategies.	19.3
COL 19.3(6)	The COL applicant is to confirm and ensure that the raw water tank and flow path to the FLEX equipment (structures, piping, components, and connections) are designed to be robust with respect to applicable hazards (e.g., seismic events, floods, high winds, and associated missiles).	19.3

Item No.	Description	Section
COL 19.3(7)	The COL applicant is to provide site-specific details of the electrical connection points (including locations, accessibility voltage level, and electrical classification) for the 480V mobile GTGs and the 4.16kV mobile generator.	19.3
COL 19.3(8)	The COL applicant is to ensure that all the connections (refer to Table 6-5 in Chapter 19.3, Reference 5) for FLEX strategies located outside the buildings are accessible and protected from all applicable external hazards (e.g., seismic events, floods, high winds, and associated missiles).	19.3
COL 19.3(9)	The COL applicant is to address site-specific strategies to mitigate BDBEEs as specified in the NRC Order EA-12-049.	19.3
COL 19.3(10)	The COL applicant is to propose a License Condition to verify the development and implementation of the guidance, strategies, and programs for the mitigation strategies for ensuring core cooling, containment function and spent fuel pool cooling capabilities during an extended loss of ac power event at an APR1400 nuclear power plant.	19.3
COL 19.3(11)	The COL applicant is to provide a description of the operational programs that will provide assurance of the functional capability of the pumps, valves, and dynamic restraints used in the mitigation strategies for ensuring core cooling, containment function, and spent fuel pool cooling capabilities during an extended loss of ac power event at an APR1400 nuclear power plant.	19.3
COL 19.3(12)	The COL applicant is to address SFP level instrumentation maintenance procedure development and perform training as specified in NRC Order EA-12-051.	19.3
COL 19.3(13)	The COL applicant is to address development of EOPs, SAMGs, and EDMGs that incorporate lessons learned from TEPCO's Fukushima Dai-Ichi nuclear power plant accident as addressed in SECY-12-0025.	19.3
COL 19.3(14)	The COL applicant is to address enhancement of the offsite communication system as specified in the NRC Request for Information pertaining to NTTF Recommendation 9.3.	19.3

Item No.	Description	Section
COL 19.3(15)	The COL applicant is to address staffing for large-scale natural events as specified in the NRC RFI pertaining to NTTF Recommendation 9.3.	19.3

19.3.1.6 Conclusion

The staff reviewed the Beyond Design Basis External Event information provided in the APR1400 DCD, and the RAI responses, against the guidelines of SECY-12-0025, NRC Orders EA-12-049 and related Request for Information, and the Tier 1, 2, and 3 Near Term Task Force items addressed in Technical Report APR1400-E-P-NR-14005. The staff determined that the APR1400 DCD Tier 2 Section 19.3.2.3 mitigation strategies are acceptable, because they can provide adequate capability to achieve and maintain safe shutdown in the event of a Beyond Design Basis External Event.

19.3.2 Reliable Spent Fuel Pool Instrumentation (Based on Recommendation 7.1)

19.3.2.1 Introduction

As a result of the Fukushima Dai-Ichi event, additional requirements have been established to manage and mitigate external events that are beyond the design basis of the plant. DCD Tier 2 Section 19.3 addresses the APR1400 conformance with SECY-12-0025, including the requirements contained in NRC Orders EA-12-049 and EA-12-051 and the related Request for Information (RFI). DCD Tier 2 Section 19.3.2.4 "Recommendation 7.1 – Reliable Spent Fuel Pool Instrumentation," specifically addresses EA-12-051. The APR1400 SFP water level instrumentation is described as being consistent with the guidelines addressed in

NRC EA-12-051, NEI 12-02, and JLD-ISG-2012-03. Additionally, the applicant submitted Technical Report APR1400-E-P-NR-14005, "Evaluations and Design Enhancements to Incorporate Lessons Learned from the Fukushima Dai-Ichi Nuclear Accident," Revision 0, KHNP, December 2014, which provides additional design detail information to demonstrate compliance with applicable guidance and commission order.

19.3.2.2 Summary of Application

DCD Tier 1: There is no Tier 1 information regarding reliable spent fuel pool instrumentation based on NTTF Recommendation 7.1.

DCD Tier 2: The design basis and complete description of the spent fuel instrumentation can be found in DCD Tier 2 Section 19.3.2.4, and APR1400-E-P-NR-14005. The information contained in DCD Tier 2 Section 19.3.2.4, describes how the SFP level instruments addresses all the design criteria identified in NEI 12-02.

Inspections, Tests, Analyses and Acceptance Criteria: There are no ITAAC items for this area of review, because the mitigation strategies in this section are for beyond design basis events.

Initial Test Program: No specific initial test has been identified for the spent fuel pool level instrumentation, however, part of test 14.2.12.1.77, "Spent Fuel Pool Cooling and Cleanup System Test," requires the calibration and test of the SFP level instrumentation.

Technical Specifications: No Technical Specifications are provided for the spent fuel pool level instrumentation.

19.3.2.3 Regulatory Basis

The requirements and guidance for reliable SFP instrumentation are established or described in the following:

- SRM-SECY-12-0025, "Staff Requirements – SECY-12-0025 – Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012, approves issuance of orders for reliable spent fuel pool instrumentation under an administrative exemption to the Backfit Rule.

- JLD-ISG-2012-03, Revision 0, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," issued August 29, 2012, endorses NEI 12-02, Revision 1, "Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," with exceptions and clarifications.

19.3.2.4 Technical Evaluation

NEI 12-02 provides an acceptable approach for satisfying the applicable requirements. In DCD Tier 2 Section 19.3.2.4, the applicant discusses how the APR1400 design addresses the requirements of Commission Order EA-12-051. APR1400-E-P-NR-14005 provides additional design information in order to demonstrate conformance with applicable guidance.

In Attachment 2 to Commission Order EA-12-051, the Commission states that trained personnel shall be able to identify the following pool water level conditions:

Level 1 - level that is adequate to support operation of the normal fuel pool cooling system,

Level 2 - level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and

Level 3 - level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

NEI 12-02 describes Level 1 as a level that is adequate to support operation of the normal fuel pool cooling system. Level 1 represents the HIGHER of the following two points:

- The level at which reliable suction loss occurs due to uncovering of the coolant inlet pipe, weir or vacuum breaker (depending on the design), or
- The level at which the water height, assuming saturated conditions, above the centerline of the cooling pump suction provides the required net positive suction head specified by the pump manufacturer or engineering analysis.

This level will vary from plant to plant and the instrument designer will need to consult plant-specific design information to determine the actual point that supports adequate cooling system performance.

APR1400-E-P-NR-14005 indicates that Level 1 is at elevation 144 ft, 0 inch, which ensures that the SFP cooling suction lines are covered. Level 2 is identified as greater than 3.05 m (+/-0.305

m) (10 ft [+/-1 ft]) above the top of the fuel storage racks. Level 3 is identified as greater than 0.305 m (1 ft) above the top of the fuel storage racks.

The staff evaluated the setpoints defined in DCD Tier 2 Section 19.3.2.4 and determined that these levels are in accordance with the guidance provided in NEI 12-02, and JLD-ISG-2012-03, and therefore, in conformance with Commission Order EA-12-051.

Instruments:

Commission Order EA-12-051, Attachment 2, Section 1.1 states that the instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor spent fuel pool water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or heat and humidity from a boiling pool.

In APR1400-E-P-NR-14005, Revision 2, Section 5.1.3.2.2, the applicant states that both instrument channels are fixed and provide a continuous level indication. The staff evaluated the applicant's instruments description and determined that crediting two permanently installed instruments as primary and backup channels follows the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Arrangement:

Commission Order EA-12-051, Attachment 2, Section 1.2 states that the spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the safety-related instruments to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

In DCD Tier 2 Section 9.1.3.5.4, the applicant states that SFP level instruments are designed as safety-related instruments. In APR1400-E-P-NR-14005, Revision 2, Section 5.1.3.2.5 the applicant states that the channels/probes of level instruments are separated to reduce the potential for falling debris or missiles affecting both channels of instrumentation. In addition, the applicant describes the use of separate routing paths for cables and rigid conduit to provide reasonable protection against falling debris and structural damage.

The staff evaluated the applicant's description of the SFP level instruments and determined that the equipment description and the safety classification of the components would ensure that the SFP level instruments are arranged in a manner that provides reasonable protection against missiles; therefore, the staff concludes that these features follow the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Mounting:

Commission Order EA-12-051, Attachment 2, Section 1.3 states that the installed instrument channel equipment within the spent fuel pool shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

In DCD Tier 2 Section 9.1.3, the applicant identifies the SFP level instruments as safety-related, which requires the instruments to be design to Seismic Category I standards. Also, APR1400-E-P-NR-14005, Revision 2, Section 5.1.3.2.6 states that the primary and backup systems are installed as seismic Category I to meet the NRC JLD-ISG-2012-03 and NEI 12-02 guidance requirements. Other hardware stored in the SFP is evaluated to provide reasonable assurance that it does not adversely interact with the SFP instrument probes during a seismic event.

The staff evaluated the applicant's description of the SFP level instruments and determined that the equipment description and the safety classification of the components would ensure that the SFP level instruments are mounted in a manner that provides reasonable protection against seismic events; therefore, the staff concludes that these features follow the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Qualification:

Commission Order EA-12-051, Attachment 2, Section 1.4 states that the level instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period.

In Technical Report APR1400-E-P-NR-14005, Revision 2 the applicant states that the probe is designed to operate in borated and non-borated water over the entire expected range of pool conditions from normal water temperatures to boiling temperatures. Components in the area of the SFP are designed for the temperature, humidity, and radiation levels expected during normal, event, and post-event conditions for no fewer than 7 days post-event or until offsite resources can be deployed by the mitigating strategies. Equipment located in the SFP is qualified to withstand a total accumulated dose for the expected lifetime at normal conditions plus accident dose received at post-event conditions with SFP water level within 0.305 m (1 ft) of the top of the fuel rack seated in the spent fuel pool (Level 3).

The staff evaluated the applicant's description of the SFP level instruments provided in APR1400-E-P-NR-14005, Revision 2 and determined that the qualification requirements presented are consistent with the criteria discussed in the guidance. Therefore, the staff concludes that these features follow the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Independence:

Commission Order EA-12-051, Attachment 2, Section 1.5, states that the primary instrument channel shall be independent of the backup instrument channel.

In APR1400-E-P-NR-14005, Revision 2, Sections 5.1.3.2.4, 5.1.3.2.5, and 5.1.3.2.8, the applicant states that the primary instrument channel is independent of the backup instrument channel. Independence is obtained by physical separation of components between channels and the separate use of Class 1E power sources.

The staff verified that the physical separation of the primary and backup channels will be sufficient to establish physical independence, and that the channels are not electronically connected. Accordingly, the staff concludes that this feature follows the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Power Sources:

Commission Order EA-12-051, Attachment 2, Section 1.6 states that the instrumentation channels shall provide for power connections from sources independent of the plant alternating current (ac) and direct current (dc) power distribution systems, such as portable generators or replaceable batteries. Power supply designs should provide for quick and accessible connection of sources independent of the plant ac and dc power distribution systems. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

In APR1400-E-P-NR-14005, Revision 2, Section 5.1.3.2.9, the applicant states that the SFP level instruments are powered from independent Class 1E trains. Following a loss of normal power, the instruments are automatically aligned to dedicated batteries which support a minimum of 72 hours of operation. After this, power can be supplied by mobile (portable) gas turbine generators, as part of the mitigation strategies.

The staff evaluated the applicant description of the instrument power connections. The staff noted that the design description includes the capability of powering the instruments from sources independent of the plant alternating current (ac) and direct current (dc) power distribution systems. In order to make the alternate power connections, the applicant is crediting beyond-design-event (BDE) mitigation strategies, which the staff evaluated in Section 19.3.1 of this report. The staff concludes that the design features discussed above follows the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Accuracy:

Commission Order EA-12-051, Attachment 2, Section 1.7 states that the instrument shall maintain its designed accuracy following a power interruption or change in power source without recalibration.

In Technical Report, APR1400-E-P-NR-14005, Revision 2, Sections 5.1.3.2.9 and 5.1.3.2.10, the applicant states that the SFP level instrument accuracy and performance are not affected by power interruptions, switching power sources, or re-starting of the processor. The instrument is designed with an accuracy better than ±7.62 cm (± 3 in).

The staff evaluated the applicant's description of the SFP level instruments provided in Technical Report APR1400-E-P-NR-14005, Revision 2, and determined that the level instrumentation will retain its calibration following a power interruption; therefore, the staff concludes that this feature follows the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Testing:

Commission Order EA-12-051, Attachment 2, Section 1.8 states that the instrument channel design shall provide for routine testing and calibration.

APR1400-E-P-NR-14005, Revision 2, states that the display/processor performs automatic in-situ calibration and automatically monitors for cable, connector, and probe faults using TDR technology. Channel degradation due to age or corrosion is not expected but can be identified by monitoring trends.

The staff reviewed the applicant's system description and identified that the permanently installed instrument channels are normally used to monitor the SFP level and will be subject to routine testing and calibration in accordance with plant procedures. Accordingly, the staff concludes that these design features follow the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Display:

Commission Order EA-12-051, Attachment 2, Section 1.9 states that trained personnel shall be able to monitor the SFP water level from the control room, the alternate shutdown panel, or another appropriate and accessible location. The display shall provide on-demand or continuous indication of SFP water level.

APR1400-E-P-NR-14005, Revision 0, states that the primary and backup level instruments are continuously displayed in the main control room and remote shutdown room.

The staff evaluated the applicant's description of the SFP level instruments provided and determined that it will provide continuous indication of the SFP water level in the main control room and the remote shutdown room; therefore, the staff concludes that this feature follows the guidance provided by JLD-ISG-2012-03 and are in conformance with Commission Order EA-12-051.

Combined License Information Item

Commission Order EA-12-051, Attachment 2, Section 2 states that the spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of a training program. Personnel shall be trained in the use and the provision of alternate power to the safety-related level instrument channels. Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent fuel pool instrument channels, and processes shall be established and maintained for scheduling and implementing necessary testing and calibration of the primary and backup spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

In Revision 0 of Tier 2 Section 19.3.2.4, the applicant identified COL 19.3(6), which states:

> The COL applicant is to address SFP level instrumentation maintenance procedure development and perform training as specified in NRC Order EA-12-051.

The staff evaluated the COL item and determined that developing procedures and training is the responsibility of the COL applicant and this COL item will ensure that the COL applicant will address the training and procedure requirements presented in NRC Order EA-12-051. In a revision to the DCD the applicant has renumbered the COL item to COL 19.3(12). The staff finds the proposed COL item acceptable.

19.3.2.5 Conclusion

The staff evaluated the applicant's DCD, and the Technical Report APR1400-E-P-NR-14005, Revision 2, related to the SFP water level instrumentation, and determined that the proposed level instruments are designed in accordance with the guidance provided in JLD-ISG-2012-03. Therefore, these instruments are considered to be reliable, able to withstand beyond-design-basis natural phenomena, and monitor key spent fuel pool level parameters, as described in Commission Order EA-12-051.

19.4 Loss of Large Areas of the Plant Due to Explosions or Fire

19.4.1 Summary of Application

DCD Tier 1: There is no Tier 1 information for this area of review.

DCD Tier 2: This application describes the design enhancements for the APR1400 that could assist a COL applicant in implementing the strategies needed for a loss of large area (LOLA) event. The applicant stated that the APR1400 design utilized the guidance in NEI 06-12, Revision 3, "NEI 06-12, B.5.b Phase 2 & 3 Submittal Guideline," to address LOLA events for this new reactor design. The applicant approached the LOLA evaluation in a phased approach similar to the approach used by COL applicants and existing plants. Phase 1 focuses on firefighting enhancements and the operational aspects of responding to explosions or fire. Phase 2 focuses on issues associated with mitigating an event that involves the spent fuel pool (SFP). Phase 3 focuses on methods to provide sources of alternative cooling water to critical systems, as well as mitigating the impact of radiological release through containment.

The applicant also stated that the operational and programmatic aspects of responding to LOLA events are to be addressed in facility procedures prior to fuel load. No site-specific design is included in the submittal.

Inspections, Tests, Analyses and Acceptance Criteria (ITAAC): There are no ITAAC items for this area of review because LOLA is a beyond design basis event.

Technical Specifications (TS): There are no TS for this area of review because LOLA is a beyond design basis event.

19.4.2 Regulatory Basis

As required by 10 CFR 50.54(hh)(2), each licensee is to develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with LOLAs of the plant due to explosions or fire, to include strategies in the following areas:

- Firefighting;
- Operations to mitigate fuel damage; and
- Actions to minimize radiological release.

19.4.3 Technical Evaluation

The staff reviewed the application by using NEI 06-12, Revision 3, which was endorsed by DC/COL-Interim Staff Guidance-16 (DC/COL-ISG-016), "Interim Staff Guidance Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d) Loss of Large Areas of the Plant due to Explosions or Fires from a Beyond-Design Basis Event." The staff considers conformance with the guidance in NEI 06-12, Revision 3, as an acceptable method in satisfying the Commission's requirements in 10 CFR 50.54(hh)(2).

19.4.3.1 Fire Protection System Features to Facilitate Mitigative Strategies

In its application, the applicant described the APR1400 fire protection system (FPS), which includes the fire pumps, fire water storage tanks, and the underground yard main loop. The applicant stated that the FPS is designed to facilitate Phase 1, 2, and 3 mitigative strategies. As described, the FPS is sized such that it contains sufficient water for two hours of operation of the largest sprinkler system plus a 500 gpm (113.6 m^3/h) manual hose stream allowance. Design features include three 50-percent capacity fire pumps (one electric and two diesel-driven), redundant fire water storage tanks, and an eight-hour refill capability for one fire water storage tank. The underground yard main loop will have post-indicator valves that can provide sectionalized control and isolation to portions of the loop. The FPS water supply will also have isolation valves from each branch header from the yard main loop. These post-indicating valves are located outside of buildings and can be used to isolate the FPS water supply as it enters a building. The FPS water supply pumps and water storage tanks are physically located a minimum of 91.44 m (100 yards) from key plant target areas.

The applicant stated that in implementing LOLA strategies, the FPS may be used as a water source if site procedures provide guidance on sharing/balancing the use of this resource between firefighting and supporting the LOLA strategy.

<u>Pump Redundancy</u>

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19.4.3.2 Identification of Key Safety Functions

The applicant stated that the generic pressurized water reactor (PWR) key safety functions identified in NEI 06-12, Subsection 4.2.3.1, “Identification of Key Safety Functions,” are all applicable to the APR1400. The PWR key safety functions are as follows:

- Reactor cooling system (RCS) inventory control;
- RCS heat removal;
- Containment isolation;
- Containment integrity; and
- Release mitigation.

The applicant did not identify any new key safety functions for the APR1400.

NEI 06-12 states that for each key safety function, the applicant should first identify the minimal set of equipment for both a primary and alternate means of satisfying the key safety function.

Second, the applicant should determine if the primary and alternate means are adequately separated per the guidance provided in NEI 06-12. According to NEI 06-12, a key to survival of redundant equipment exposed to large fires and explosions is the degree of spatial separation and/or barriers that exist between the equipment. Lastly, if the primary and alternate means are not adequately separated, the applicant should provide a mitigative strategy that will satisfy the key safety function.

Evaluation of Key Safety Function - RCS Inventory Control

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Evaluation of Key Safety Function – RCS Heat Removal

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__Evaluation of Key Safety Function – Containment Isolation__

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__Evaluation of Key Safety Function – Containment Integrity__

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<u>Evaluation of Key Safety Function – Release Mitigation</u>

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19.4.3.3 Phase 1 Firefighting and Operational Enhancements

NEI 06-12 guidance lists 31 Phase 1 firefighting and operational strategies that should be considered by an applicant when developing its mitigative strategies to meet the requirements of 10 CFR 50.54(hh)(2). The applicant provided a design enhancement for supplying the fire protection ring header, which is evaluated below.

***Evaluation – Supplying the Fire Protection Ring* Header**

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The staff finds this enhancement acceptable because it follows the guidance in NEI 06-12 by providing an alternate means of supplying water to the fire protection yard main.

19.4.3.4 Phase 2 Spent Fuel Pool Strategies

The NEI 06-12 guidance lists four Phase 2 SFP strategies that should be considered by an applicant when developing its mitigative strategies to meet the requirements of 10 CFR 50.54(hh)(2). These strategies are to assist in maintaining SFP cooling capabilities. The applicant has provided design enhancements for three of the strategies, which are evaluated, below.

Evaluation - Diverse SFP Makeup Source (Internal Strategy)

As described in NEI 06-12, “B.5.b Phase 2 & 3 Submittal Guideline,” Section 2.2, “Diverse SFP Makeup Source (Internal Strategy),” the objective of this mitigative strategy is to establish a diverse means of SFP makeup with at least a concurrent makeup capability of 500 gpm (113.6 m^3/h) beyond the normal SFP makeup capability. The term “diverse” means that the makeup source does not rely upon any of the same components or piping as the normal makeup source.

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The staff finds this acceptable because the strategy described by the applicant follows the guidance in Section 2.2 NEI 06-12, which is one of the acceptable strategies in meeting the conditions of a diverse SFP makeup source.

<u>Evaluation - SFP Makeup Capability (External Strategy)</u>

As described in NEI 06-12, Section 2.3.1, "SFP Makeup Capability," the objective of this mitigative strategy is to establish an additional means to provide SFP makeup of a least 500 gpm (113.6 m^3/h) using a portable, power independent pumping capability.

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The staff finds this acceptable because the applicant's strategy follows the guidance in Section 2.3.1 of NEI 06-12 for developing an additional means of providing makeup water to the SFP.

<u>Evaluation - SFP Spray Capability (External Strategy)</u>

As described in NEI 06-12, Section 2.3.2, "SFP Spray Capability," the objective of this mitigative strategy is to establish a flexible means of providing at least 200 gpm (45.4 m^3/h) of spray to the SFP using a portable, power-independent pumping capability. This capability serves to provide spray to the SFP in the event the SFP water level cannot be maintained.

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The staff finds this acceptable because the applicant's strategy follows the guidance in Section 2.3.2 of NEI 06-12 for establishing a spray to the SFP.

19.4.3.5 Phase 3 Mitigative Strategies

The NEI 06-12 guidance lists eight Phase 3 strategies that should be considered by an applicant when developing its mitigative strategies to meet the requirements of 10 CFR 50.54(hh)(2). Phase 3 mitigative strategies are intended to restore or maintain core cooling to mitigate potential damage to fuel in the reactor system and to mitigate potential radiological releases through the containment. The applicant has provided design enhancements for seven of the strategies, which are evaluated, below.

Evaluation - Makeup to IRWST in Order to Supply ECCS Long-Term

As described in NEI 06-12, Section 3.3.1, "Makeup to RWST," the objective of this mitigative strategy is to provide a large volume makeup source to the IRWST to supply the emergency core cooling system long term. The strategy should provide at least 300 gpm (68.1 m^3/h) to the IRWST for a period of 12 hours.

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Evaluation - Manually Depressurize Steam Generators to Reduce Inventory Loss

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Evaluation - Manual Operation of Turbine-Driven Pumps

As described in NEI 06-12, Section 3.3.3, "Manual Operation of Turbine-Driven (or Diesel-Driven) AFW Pump," the objective of this mitigative strategy is to provide a power-independent means to provide core cooling and prevent or delay core damage.

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Evaluation - Manually Depressurize SGs and use Portable Pump

As described in NEI 06-12, Section 3.3.4, "Manually Depressurize SGs and Use Portable Pump," the objective of this strategy is to provide a low-pressure makeup source to provide SG makeup and core cooling.

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The staff finds this acceptable because the applicant's strategy follows the guidance in Section 3.3.4 of NEI 06-12.

<u>Evaluation - Makeup to CST/AFWST</u>

As described in NEI 06-12, Section 3.3.5, "Makeup to CST/AFWST," the objective of this strategy is to provide a high-volume makeup source to the condensate storage tank (CST)/auxiliary feed water storage tank (AFWST) to supply AFW long term.

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The staff finds this acceptable because the applicant's strategy follows the guidance in Section 3.3.5 of NEI 06-12.

<u>Evaluation - Containment Flooding with Portable Pump</u>

As described in NEI 06-12, Section 3.3.6, "Containment Flooding with Portable Pump," the objective of this strategy is to provide a power independent means to inject water into the containment to flood the containment floor and cover core debris.

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The staff finds this acceptable because the applicant's strategy follows the guidance in Section 3.3.6 of NEI 06-12.

Evaluation - Portable Sprays

As described in NEI 06-12, Section 3.3.7, "Portable Sprays," the objective of this strategy is to provide a means to reduce the magnitude of any fission product releases by spraying.

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The staff finds this acceptable because the applicant's strategy follows the guidance in Section 3.3.7 of NEI 06-12.

19.4.3.6 Initial Test Program

There are no initial test programs because LOLA is a beyond design-basis event.

19.4.4 Combined License Information Items

The following is a list of applicable COL information items as described in DCD Tier 2 Section 19.4.4, "Combined License Information."

Table 19.4-1: Combined License Items Identified in the DCD

Item No.	Description	Section
19.4(1)	The COL applicant is to provide provision and procedure/guidance for implementing SFP strategies, including the minimum pump discharge pressure required to support the minimum flow requirements. All portable equipment is to be stored at least 91.44 m (100 yards) from the auxiliary building. The COL applicant is to specify the external water source(s) to be used.	19.4.4
19.4(2)	The COL applicant is to provide firefighting response strategy.	19.4.4
19.4(3)	The COL applicant is to provide procedure/guidance for implementing the IRWST makeup strategy.	19.4.4
19.4(4)	The COL applicant is to determine the portable DC power source and connection device to SG pressure and level indication panels.	19.4.4
19.4(5)	The COL applicant is to provide procedure/guidance for implementing the manually depressurize SGs to reduce inventory loss.	19.4.4
19.4(6)	The COL applicant is to provide procedure/guidance for implementing the manual operation of TDAFW pump.	19.4.4
19.4(7)	The COL applicant is to provide procedure/guidance for implementing the manually depressurize SGs and use portable pump.	19.4.4
19.4(8)	The COL applicant is to provide procedure/guidance for implementing the makeup auxiliary feedwater tank.	19.4.4
19.4(9)	The COL applicant is to provide procedure/guidance for implementing the flooding containment with portable pump.	19.4.4
19.4(10)	The COL applicant is to provide portable equipment including portable pump, hose and connections for phase 3 strategies.	19.4.4

The staff finds the COL information items to be acceptable. The staff finds that no additional COL information items need to be included in DCD Tier 2, Table 1.8-2 for this system.

19.4.5 Conclusion

Based on the staff's review of the information provided by the applicant, the staff concludes that the applicant has adequately followed the guidance of NEI 06-12, as endorsed by DC/COL-Interim Staff Guidance-16 (DC/COL-ISG-016), for a design certification applicant. The staff also finds that the applicant provided sufficient information regarding Phases 1, 2, and 3 design enhancements and COL items to support a COL applicant's development of mitigative strategies to meet the requirements of 10 CFR 50.54(hh)(2).

19.5 Aircraft Impact Assessment

19.5.1 Introduction

This section describes the NRC staff's evaluation of design features and functional capabilities credited by the applicant to show that the facility can withstand the effects of a large, commercial aircraft impact. This safety evaluation reviews the design features, functional capabilities, and assessment described in the APR1400 DCD Tier 2 Section 19.5, "Aircraft Impact Assessment," dated December 1, 2015 (ML15335A017), DCD Tier 2 Section 19.5, Revision 1, dated March 10, 2017 (ML17096A325), KHNP letter dated September 29, 2017, Reply to Notice of Violation Report No. 05200046/2016-202 (ML17272A599), and KHNP letter dated December 1, 2017, Supplemental Reply to a Notice of Violation Report No. 05200046/2016-202 (ML17335A072) which provided markups to DCD Tier 2 information.

The impact of a large, commercial aircraft is a BDBE. Under 10 CFR Part 52.47(a)(28) and 10 CFR 50.150, "Aircraft impact assessment," applicants for new nuclear power reactors[3] are required to perform a design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Applicants are required, in part, to submit a description of the design features and functional capabilities identified as a result of the assessment (key design features) in their DCD, along with a description of how the identified design features and functional capabilities show that the acceptance criteria in 10 CFR 50.150(a)(1) are met.

The Statement of Considerations for the aircraft impact assessment (AIA) rule regarding consideration of aircraft impacts for new nuclear power reactors states that: "The NRC decision on an application subject to 10 CFR 50.150 will be separate from any NRC determination that may be made with respect to the adequacy of the impact assessment which the rule does not require be submitted to the NRC." Since the AIA is not submitted to the NRC for its review, the staff conducts its review in this Section to determine whether or not descriptions of the design features and functional capabilities are complete enough such that, assuming the design features and functional capabilities perform their intended functions, there is reasonable assurance that the acceptance criteria in 10 CFR 50.150(a)(1) can be met.

Applicants subject to 10 CFR 50.150 must make the complete aircraft impact assessment available for an NRC inspection at the applicants' offices or their contractors' offices upon the staff's request, in accordance with 10 CFR 50.70, "Inspections," 10 CFR 50.71, "Maintenance of records, making of reports," and Section 161, "General Provisions," item c of the Atomic Energy Act of 1954, as amended. The outcome of the NRC inspection is not part of this report.

19.5.2 Summary of Application

In DCD Tier 2, Section 19.5, Revision 1, dated March 10, 2017 (ML17096A325), the applicant states that an AIA was performed in accordance with the requirements in 10 CFR 50.150(a) using the methodology described in NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Revision 8, as endorsed by the NRC in RG 1.217, "Assessment of Beyond-Design-Basis Aircraft Impacts," and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 19.5, "Adequacy of Design Features and Functional Capabilities Identified and

3 "Applicants for new nuclear power reactors" is defined in the Statement of Considerations for the Aircraft Impact Rule [74 (*Federal Register*) FR 28112, June 12, 2009].

Described for Withstanding Aircraft Impacts." Based on the results of the assessment, the applicant has identified a set of key design features to show that the acceptance criteria in 10 CFR 50.150(a)(1) are satisfied. These key design features are reported in the DCD Tier 2 Section 19.5, December 1, 2015 submittal (ML15335A017), DCD Tier 2 Section 19.5, Revision 1, dated March 10, 2017 (ML17096A325), September 29, 2017 letter (ML17272A599), and December 1, 2017 letter (ML17335A072), along with references to other sections of DCD, Revision 0, dated December 23, 2014 (ML15006A059), that provide additional details.

19.5.2.1 Description of Key Design Features

As described in the APR1400 DCD Tier 2 Section 19.5, December 1, 2015 submittal (ML15335A017), DCD Tier 2 Section 19.5, Revision 1, dated March 10, 2017 (ML17096A325), September 29, 2017 letter (ML17272A599), and December 1, 2017 letter (ML17335A072), the credited design features, functions, and references to sections containing the detailed descriptions are summarized below:

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- The RCB, as described in DCD Tier 2, Sections 3.8.1, "Concrete Containment," 3.8.2, "Steel Containment," and Figures 3.8-1 and 3.8-2, "Typical Section of Containment Structures," provides protection of safety systems located inside containment from an impact by a large, commercial aircraft.

- The nominal compressive strength of the RCB concrete reaching 6,900 psi.

- The polar crane bracket and rail girder shown in Figure 19.5-11 of the DCD Tier 2, markups, protect the safety system located inside containment from the drop of polar crane due to the impact of a large, commercial aircraft.

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- The nominal compressive strength of the AB concrete reaching 5,900 psi at 28 days.

- The configuration of the AB along with strengthened measures for some exterior and interior walls in the AB, and the floor slab of the SFP, shown in Table 19.5-2, "Required Rebar Area of Wall & Slab for AIA in Auxiliary Building and EDG Building," and Figures 19.5-12 through 19.5-17 of the DCD Tier 2, markups, protects key equipment in the AB.

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- Properties of concrete and reinforcement bars, as described in DCD Tier 2 Appendix 3.8A, “Structural Design Summary,” protect key equipment in the AB and EDGB from the impact by a large, commercial aircraft.

- The location and design of spent fuel pool (SFP) and its supporting structures, as described in DCD Tier 2, Sections 3.8A.2, “Auxiliary Building,” and 9.1.2, “New and Spent Fuel Storage,” protect the integrity of the SFP from an impact by a large, commercial aircraft and assure no leakage from the SFP below the required minimum water level.

- The location of the alternate alternating current GTG (AAC GTG) relative to the EDGs, as shown in DCD, Tier 2, Figure 1.2-1, limits the loss of electrical power to key safety-related systems from the impact of a large, commercial aircraft. The AAC GTG building will be located at least 100 yards (91.4 m) from the AB.

- The design and location of 3-hour fire pressure barriers within the AB and EDGB, as identified in DCD, Tier 2, Figures 9.5A-1, “Fire Barrier DBD – RCB/AB El 55’-0”,” through 9.5A-11, “Fire Barrier DBD – EDGB El 121’-6” & 135’-0”,” protect core cooling equipment in those buildings from fire damage from impact of a large, commercial aircraft.

- The design and location of 5 pounds per square inch – differential (psid) (34.47 kPa) pressure barriers within the AB and EDGB, as identified in DCD, Tier 2, Figures 19.5-1, “5-psid barrier – AB El. 55’-0”,” through 19.5-10, “5-psid barrier – EDGB El. 100’-0” & 121’-6”,” protect core cooling equipment in those buildings from pressurized fire damage from impact of a large, commercial aircraft.

- The design and location/separation of various heat removal systems and their support systems as highlighted in DCD Tier 2 Section 19.5.4.4, “Core Cooling Features,” December 1, 2015 submittal (ML15335A017).

19.5.2.2 Description of How Regulatory Acceptance Criteria are Met

The acceptance criteria in 10 CFR 50.150(a)(1) require: (1) that the reactor core will remain cooled or the containment will remain intact; and (2) that SFP cooling or SFP integrity is maintained. The applicant indicates that it meets the 10 CFR 50.150(a)(1) acceptance criteria by including features that, following the impact of a large, commercial aircraft, show the APR1400 design can:

- Maintain core cooling and
- Maintain spent fuel pool integrity

As indicated in its DCD Tier 2 Section 19.5, December 1, 2015 submittal (ML15335A017), the applicant proposes to maintain core cooling using the safety-related systems, which have been designed specifically to ensure that the reactor can be shut down and that decay heat can be adequately removed from the reactor core. Some of this equipment is located inside the RCB and some is located inside the AB and EDGB. The applicant states that locations inside the RCB are protected from structural and fire damage by the design of the RCB structure as well as the AB structure, which limits the penetration of a large, commercial aircraft such that the RCB is not perforated. Equipment inside the AB and EDGB are protected by the structural design features of the interior and exterior walls of the AB and EDGB. In addition, the applicant states that fire barriers have been designed and located in the AB and EDGB to contain the spread of fire inside the buildings such that at least one train of credited equipment for core cooling is protected and remains functioning following an impact of a large, commercial aircraft. The applicant states that key components associated with core cooling, including instrumentation and control equipment, located inside the RCB, AB, and EDGB, are unaffected by shock-induced vibrations resulting from the impact of a large, commercial aircraft.

Section 19.5 of the DCD Tier 2, December 1, 2015 submittal (ML15335A017), states that the design meets the spent fuel pool integrity acceptance criterion in 10 CFR 50.150(a)(1) due to the design and location of the spent fuel pool and its support structures. The applicant states that spent fuel pool liner will not be perforated and no leakage will result from the spent fuel pool below the required minimum water level due to impact by a large, commercial aircraft.

19.5.3 Regulatory Basis

The staff used the relevant regulations and guidance detailed in Sections 19.5.3.1 and 19.5.3.2, below, to perform this review.

19.5.3.1 Applicable Regulations

10 CFR 52.47(a)(28) requires applicants for standard design certifications, who are subject to 10 CFR 50.150(a), to include information required by 10 CFR 50.150(b).

10 CFR 50.150(a)(1) requires that applicants perform a design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions: (1) the reactor core remains cooled, or the containment remains intact; and (2) spent fuel cooling or SFP integrity is maintained.

10 CFR 50.150(a)(3)(iii)(A), states the requirements of paragraphs (a)(1) and (a)(2) of 10 CFR 50.150 shall apply to applicants for standard DCs under 10 CFR Part 52 issued after July 13, 2009.

10 CFR 50.150(b) requires that the final safety analysis report (FSAR) include a description of: (1) the design features and functional capabilities which the applicant has identified for inclusion in the design to show that the facility can withstand the effects of a large, commercial aircraft impact in accordance with 10 CFR 50.150(a)(1); and (2) how those design features and functional capabilities meet the assessment requirements of 10 CFR 50.150(a)(1).

19.5.3.2 Review Guidance

RG 1.217, “Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts,” issued August 2011, provides guidance for applicants to demonstrate compliance with NRC regulations with regard to AIA. In particular, this RG endorses the methodologies described in NEI 07-13, “Methodology for Performing Aircraft Impact Assessments for New Plant Designs,” Revision 8, dated April 2011.

SRP Chapter 19.5, “Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts,” issued April 2013, provides guidance for meeting the requirements in 10 CFR 50.150(a)(1) and (b).

19.5.4 Technical Evaluation

The staff reviewed the AIA information in Section 19.5 of the DCD Tier 2, December 1, 2015 submittal (ML15335A017), September 29, 2017 letter (ML17272A599), and December 1, 2017 letter (ML17335A072) as well as the referenced DCD sections. The staff’s evaluation of how the applicant’s assessment was formulated is in Section 19.5.4.1 of this SER below and the evaluation of the applicant’s key design feature descriptions is in Sections 19.5.4.2 through 19.5.4.4, of this SER below.

19.5.4.1 Reasonably Formulated Assessment

The applicant states in the DCD Tier 2 Section 19.5, December 1, 2015 submittal (ML15335A017), that its AIA is based on the guidance of NEI 07-13, Revision 8, with no exceptions. Based on the applicant’s use of the NRC endorsed guidance document NEI 07-13, Revision 8, the staff finds that the applicant has performed a reasonably formulated assessment.

The staff reviewed the AIA application in the DCD Tier 2 Section 19.5, December 1, 2015 submittal (ML15335A017), and determined it was not clear with respect to whether the AIA was performed by qualified analysts. Therefore, the staff issued RAI 423-8526, Question 19.05-01 (ML16061A076), to address this issue.

In response to RAI 423-8526, Question 19.05-01, dated April 7, 2016 (ML16098A383), the applicant stated that the AIA was performed by qualified personnel. Further, the applicant stated that the contractors hired to perform the AIA are well-experienced and have previously performed an AIA for many other design centers and thus meet the qualifications depicted in SRP 19.5. The staff finds that the applicant adequately addressed this question since the two contractors are experts in the area of AIA, as verified by the NRC, and were involved in the development of NEI 07-13, Revision 8. Thus, the applicant has provided a well-supported basis for the staff to find that the contractors being used to perform the AIA are qualified, consistent with the guidance of SRP 19.5, Section III, item 2. DCD Tier 2 Revision 1, Subsection 19.5.3 was revised, as shown in the markup attached in the RAI response, to include the statement that the AIA was performed by qualified personnel. Hence, the staff finds the response to

RAI 423-8526, Question 19.05-01, acceptable, and considers RAI 423-8526, Question 19.05-1, resolved and closed.

19.5.4.2 Key Design Features for Core Cooling

The key design features listed in the DCD Tier 2 Section 19.5.4.4, December 1, 2015 submittal (ML15335A017), have been designed specifically to perform core cooling functions during normal power operation and following design-basis events initiated during power operation. The staff's evaluation, documented in other sections of this report, was used to confirm that these features are also suitable for maintaining core cooling following impact of a large commercial aircraft. During the review, the staff also confirmed that all of these design features can be initiated and operated from the control room or an alternate location, and require little, if any, further operator intervention to maintain the core cooling function.

DCD Tier 2 Section 19.5.4.1,"RCB and SFP," December 1, 2015 submittal (ML15335A017), identifies key design features inside the reactor containment building necessary for core cooling; these include: the reactor pressure vessel (RPV), steam generators, reactor coolant loop piping, pilot operated safety relief valves, control element drive mechanisms, the safety injection and shutdown cooling system suction line motor operated valves, discharge line check valves, and associated instrumentation and control equipment.

DCD Tier 2 Section 19.5.4.4, December 1, 2015 submittal (ML15335A017), identifies core cooling features located outside the reactor containment building; these include: the design and physical separation of the safety injection system, shutdown cooling system, auxiliary feedwater system, main steam safety and atmospheric dump valves, and charging and auxiliary charging pumps. These are the key primary system design features for assuring core cooling following a reactor trip in response to an aircraft impact event. The design and physical separation of the component cooling water system, portions of the essential service water (ESW) system located in the ESW building, the Class 1E electrical power supply and distribution system, and the safety-related instrumentation and control (I&C) system are key supporting system design features for assuring core cooling following a reactor trip in response to an aircraft impact event.

In addition to the equipment discussed above, DCD Tier 2 Section 19.5.4.4, December 1, 2015 submittal (ML15335A017), specifies that physical separation between the main control room, remote shutdown room, and remote control console; and the ability to power the safety injection pumps, charging pumps, containment spray pumps, and shutdown cooling pumps from the AAC GTG are key design features for core cooling following a reactor trip. In addition, DCD Tier 2 Section 19.5.4.4, December 1, 2015 submittal (ML15335A017), indicates that, if all Class 1E power from the EDGs is unavailable due to an aircraft impact on the ultimate heat sink, the AAC GTG is utilized to power equipment to provide make-up water for the reactor coolant system. The capability of the safety injection pumps to inject water from the in-containment refueling water storage tank, and the charging pumps and auxiliary charging pump to take suction from the volume control tank or boric acid storage tank are both considered key design features.

During the review, the staff needed additional information in order to ensure compliance with 10 CFR 50.150. Therefore, the staff issued RAI 424-8532, Question 19.05-2 (ML16061A143), requesting that the applicant provide additional information associated with the key design features credited for core cooling. In RAI 424-8532, Question 19.05-2, item a, the staff requested that the applicant ensure that the key components inside the RCB listed in the DCD Tier 2 Section 19.5.4.1, December 1, 2015 submittal (ML15335A017), are adequately described. In response to RAI 424-8532, Question 19.05-2 dated April 16, 2016

(ML16107A076), the applicant provided a DCD markup that states the design of the reactor pressure vessel and associated reactor coolant system components in the RCB, as described in DCD, Tier 2, Sections 5.3 and 5.4, are key design features. The staff finds the applicant's response to RAI 424-8532, Question 19.05-2, item a, acceptable because it modifies the DCD to contain a description of the design features and functional capabilities of key components inside the RCB as required by 10 CFR 50.150(b).

In RAI 424-8532, Question 19.05-2, item b, the staff requested the applicant verify that the DCD contains a complete list of structures or buildings credited for protecting core cooling equipment, or spent fuel pool integrity. In a response dated April 16, 2016 (ML16107A076), the applicant stated that the DCD Tier 2, December 1, 2015 submittal (ML15335A017), Section 19.5.4.2, "Plant Arrangement," item d will be modified to include the emergency diesel generator building and specify that the concrete properties and reinforcement bars meet the minimum requirements for physical damage rule sets, as shown in Table 3-2, "Representative Structure Used to Develop Physical Damage Rule Sets" of NEI 07-13. The staff finds the applicant's response to RAI 424-8532, Question 19.05-2, item b, acceptable because it modifies the DCD in order to identify design features and functional capabilities credited for protecting Structures, Systems and Components (SSCs) necessary for core cooling and SFP integrity consistent with SRP 19.5.

In RAI 424-8532, Question 19.05-2, item c, the staff requested that the applicant adequately describe the AAC GTG source and the separation distance necessary to protect the AAC GTG and its components. The staff also requested the separation distance be specified as a key design feature. In response to RAI 424-8532, Question 19.05-2 dated April 16, 2016 (ML16107A076), the applicant stated that a statement in DCD Tier 2 Section 19.5.4.2 will be added to specify that the AAC GTG will be located 100 yards (91.4 m) from the auxiliary building. In addition, the applicant stated that the AAC GTG is described in DCD Tier 2 Section 8.4.1.3, and included that information in DCD Tier 2 Section 19.5.4.4. The staff finds the applicant's response to RAI 424-8532, Question 19.05-2, item c, acceptable because it modifies the DCD to provide an adequate description of the design features and functional capabilities, to include location and separation, of the AAC GTG, as required by 10 CFR 50.150(b).

In RAI 424-8532, Question 19.05-2, item d, the staff requested that the applicant confirm that no additional support systems are credited in the assessment for maintaining core cooling, such as the essential chilled water system or ultimate heat sink, which are typically credited as key design features for core cooling. In response to RAI 424-8532, Question 19.05-2 dated November 2, 2017 (ML17306A144), the applicant stated the essential chilled water system, including the pipe routing, the ultimate heat sink and cables routing between A and B I&C equipment and RCC are required for maintaining core cooling. The applicant adds that design and physical separation of these redundant systems and equipment is credited for core cooling, and DCD Tier 2 Section 19.5.4.4 will be revised to include these design features. The staff finds the applicant's response to RAI 424-8532, Question 19.05-2, item d, acceptable because it modifies the DCD to contain a complete description of the design features and functional capabilities credited for core cooling as required by 10 CFR 50.150(b).

In RAI 424-8532, Question 19.05-2, item e, the staff requested that the applicant clarify if the AIA accounted for the effect of smoke on ventilation systems, diesel generators, or other components as specified in NEI 07-13. In response to RAI 424-8532, Question 19.05-2 dated April 16, 2016 (ML16107A076), the applicant stated that the separation between the electrical divisions is adequate to preclude the failure of both electrical divisions due to smoke and proposes to include this statement in DCD Tier 2 Section 19.5.4.2. The staff finds the applicant's response to RAI 424-8532, Question 19.05-2, item e, acceptable because it confirms that the AIA considered the effects of smoke consistent with NEI 07-13.

The incorporation of all DCD markups provided in the response to RAI 424-8532, Question 19.05-2, items a through e, was identified as Confirmatory Item 19.5-2. The staff confirmed that the RAI response markups, except for item d, were incorporated into Revision 1 of the DCD. The staff confirmed that the DCD markup for item d was incorporated into Revision 2 of the DCD. Therefore, the staff considers Confirmatory Item 19.5-2, to be resolved and closed except for item d. Based on the review of the DCD to verify that it incorporated the text as shown in the RAI responses, the staff has confirmed incorporation of the changes described above; therefore, RAI 424-8532, Question 19.05-2, is resolved and closed.

DCD Tier 2 Section 19.5.4.4, December 1, 2015 submittal (ML15335A017), also provides a description of the design features and functional capabilities of the APR1400 that assure long term cooling following a large, commercial aircraft impact while the plant is shutdown with the reactor head removed and the reactor water level at or near the reactor vessel (RV) head flange. Specifically, the reactor core is cooled by the shutdown cooling system. In the event that the shutdown cooling system is unavailable, the ability to maintain core cooling is performed by the safety injection system or charging pumps. Further, administrative controls require that no trains of the safety injection system or shutdown cooling system be out of service for maintenance when the reactor vessel head is untensioned and the reactor vessel water level is at or near the reactor vessel head flange.

DCD Tier 2 Section 19.5.4.4, December 1, 2015 submittal (ML15335A017), specifies that, following reactor shutdown, decay heat be discharged via the main steam safety valves or main steam atmospheric dump valves, and, under this condition, additional boration is unnecessary to maintain the reactor subcritical. However, it was unclear to the staff whether there are certain strike locations that would limit secondary heat removal and, therefore, necessitate borated makeup water. In RAI 424-8532, Question 19.05-4 (ML16061A143), the staff requested the applicant explain if there are certain strike locations in which borated water is necessary for maintaining adequate shutdown margin. In response to RAI 424-8532, Question 19.05-4 dated April 16, 2016 (ML16107A076), the applicant states that the primary system is maintained at operating pressure and temperature by adjusting auxiliary feedwater flow to match the decay heat rate from the reactor core, and there are no scenarios which would require borated water for maintaining adequate shutdown margin. The staff finds the applicant's response to RAI 424-8532, Question 19.05-4 acceptable because it confirms that borated water is not necessary for maintaining core cooling. The staff considers RAI 424-8532, Question 19.05-4 resolved and closed.

In the event of a threatened aircraft impact while the reactor is at full power operation, it is assumed, per NEI 07-13, Revision 8, that the operators will successfully shutdown the reactor unless equipment essential for shutting down the reactor is within the physical damage footprint. For the APR1400, the only equipment necessary for shutting down the reactor is the control element drive mechanism, which is protected by the RCB from impact of a large, commercial aircraft. In addition, DCD, December 1, 2015 submittal (ML15335A017), Tier 2 Section 19.5.4.1, states that the control element drive mechanism is a key design feature for tripping the reactor, and is located inside of the RCB on top of the reactor vessel closure head and, upon a loss of internal power distribution, the control rods drop into the reactor core by gravity. The action of shutting down the reactor and maintaining shutdown margin ensures that the fuel in the reactor is kept subcritical.

Based on the staff's review of DCD Tier 2 Section 19.5, December 1, 2015 submittal (ML15335A017), discussed above, and the applicant's use of the NRC endorsed guidance document of NEI 07-13, Revision 8, the staff finds that the applicant has performed a reasonably formulated analysis within the aircraft impact assessment in order to identify key design features necessary for core cooling. Also based on the above, the staff finds the applicant's description of the key design features for maintaining core cooling to be adequate and acceptable, and therefore meets the requirements of 10 CFR 50.150(b).

The staff compiled a complete list of the APR1400 key design features in the table below.

Table 19.5-1: Key Design Features

Design Feature Location and Design	Modes	DCD Reference Sections	Function
Control Element Drive Mechanism	Power operations	3.9.4 and 4.6	Shutdown reactor
RPV and associated RCS components	Power operations and shutdown modes	5.3 and 5.4	Core cooling
Fire Barriers in Auxiliary Building and Emergency Diesel Generator Building: 3-hour fire-rated	Power operation and shutdown modes	9.5.1 9.5A Figures 9.5A-1 – 9.5A-11	Protect core cooling equipment from fire damage
Fire Barriers in AB and EDGB: 3-hour fire-rated, 5-psid rated	Power operation and shutdown modes	9.5.1 9.5A Figures 9.5A-1 – 9.5A-11 Figures 19.5-1 – 19.5-10	Protect core cooling equipment from fire damage
Safety Injection (SI)	Following a reactor trip	6.3	Core cooling (including if reactor coolant pump seal injection is lost or shutdown cooling is unavailable)
Shutdown Cooling	Following a reactor trip or with reactor head removed and water level at or near the reactor flange	5.4.7	Core cooling
Auxiliary Feedwater	Following a reactor trip	10.4.9	Core cooling
Main Steam Safety and Atmospheric Dump Valves	Following a reactor trip	10.3.2,2,3 and 10.3.2.2.4	Core cooling
Component Cooling Water System (CCWS)	Following a reactor trip	9.2.2	Core cooling (support system)

Design Feature Location and Design	Modes	DCD Reference Sections	Function
Essential Service Water System	Following a reactor trip	9.2.1	Core cooling (support system)
Essential Chilled Water System	Following a reactor trip	9.2.7	Core cooling (support system)
Ultimate Heat Sink	Following a reactor trip	9.2.5	Core cooling (support system)
Class 1E Power and Distribution (including EDGs)	Following a reactor trip	8.3	Core cooling (support system)
AAC GTG	Following a reactor trip	8.4.1.3	Core cooling (support system)
Safety-related I&C, and RCC	Following a reactor trip	Chapter 7	Core cooling (support system)
Charging and Auxiliary Charging Pumps for RCP Seal Cooling or Inventory Control	Following a reactor trip	9.3.4.2.1 9.3.4	Core cooling (support if CCWS is unavailable or shutdown cooling and SI are unavailable)
Containment Spray Pumps and Heat Exchanger	Following a reactor trip	6.2.2	Containment heat removal if RCP seal injection is lost
Spent Fuel Pool and Support Structures	Power operation and shutdown modes	3.8A.2 9.1.2	Spent Fuel Pool integrity
RCB	Power operation and shutdown modes	3.8.1 3.8.2	Protect core cooling equipment from physical damage
AB	Power operation and shutdown modes	3.8.4	Protect portions of the RCB from physical damage (intervening structure) and protect core cooling equipment inside the AB from physical damage
EDGB	Power operation and shutdown modes	3.8.3 8.3 Fig. 1.2-14 Fig. 1.2-21	Protects portions of AB from physical damage (intervening structure) and protect core cooling equipment inside EDGB from physical damage

19.5.4.3 Key Design Features that Protect Core Cooling Design Features

The key design features and functional capabilities that protect the core cooling design features are described below as follows: fire barriers and fire protection features, plant arrangement and plant structural design features, and ability to survive shock-induced vibrations.

19.5.4.3.1 Fire Barriers and Fire Protection Features

The fire protection key design features that protect and allow for maintaining core cooling are identified and described in the DCD Tier 2, December 1, 2015 submittal (ML15335A017), Section 19.5.4.3, “Fire Barriers and Fire Protection Features.” These include the design and location of 3-hour fire rated barriers, including fire doors that separate the safety divisions within the AB. The applicant states that the assessment credited the design and location of fire barriers (including doors, penetration seals, and dampers) as depicted in DCD, Tier 2, Figures 9.5A-1 through 9.5A-11, to limit the effects of internal fires created by the impact of a large, commercial aircraft. In addition, certain fire barriers (including doors, fast-acting blast dampers, and penetration seals) are credited for 5 psid (34.5 kPa) and stated to be identified in DCD, Tier 2, Figures 9.5A-1 through 9.5A-9. However, the staff noted during its initial review of these above key design feature descriptions that it was not clear whether all or only some of the fire barriers in DCD Tier 2, Figures 9.5A-1 through 9.5A-11, were designated as key design features for fire protection. Because additional information in the referenced figures was needed to determine whether the design meets the requirements in 10 CFR 50.150(b), the staff issued RAI 427-8444, Question 19.05-5 (ML16061A265), to request additional information.

In RAI 427-8444, Question 19.05-5, item 1, the staff requested that the applicant clarify DCD Tier 2, December 1, 2015 submittal (ML15335A017), Subsection 19.5.4.3, by clearly indicating whether all fire barriers on the referenced figures, or only those that separate safety divisions, are the key design features. In response to RAI 427-8444, Question 19.05-5 dated April 16, 2016 (ML16196A361), the applicant stated the fire barriers depicted in the fire area drawings in DCD Tier 2 Section 9.5A, are key design features. The applicant proposed a markup to the DCD Tier 2, December 1, 2015 submittal (ML15335A017), Subsection 19.5.4.3, containing this clarification. The staff finds this response acceptable because the applicant clearly identifies and adequately describes the credited key design features in accordance with 10 CFR 50.150(b). Incorporation of the DCD markup information was being tracked as a confirmatory Item 19.5-3. The staff confirmed that the RAI response markup was incorporated into Revision 1 of the DCD. Therefore, the staff considers Confirmatory Item 19.5-3 to be resolved and closed.

In RAI 427-8444, Question 19.05-5, item 2, the staff requested that the applicant clearly identify which EDGB fire protection features are credited. In response to RAI 427-8444, Question 19.05-5 dated April 16, 2016 (ML16196A361), the applicant stated the 3-hour fire rated barriers in the EDGB are key design features. These are shown on DCD Tier 2, Figures 9.5A-10 and 9.5A-11. The applicant proposed a markup to the DCD Tier 2, December 1, 2015 submittal (ML15335A017), Subsection 19.5.4.3, containing this clarification. The staff finds this response acceptable because the applicant clearly identifies and adequately describes the credited key design features in accordance with 10 CFR 50.150(b). The RAI is therefore considered closed. The incorporation of the DCD markup provided was being tracked as confirmatory Item 19.5-4. The staff confirmed that the RAI response markup has been incorporated into Revision 1 of the DCD. Therefore, the staff considers Confirmatory Item 19.5-4 to be, resolved and closed.

In RAI 427-8444, Question 19.05-5, Item 3, the staff requested that the applicant clearly depict the credited 5 psid (34.5 kPa) rated fire barriers on DCD, Tier 2, Figures 9.5A-1 through 9.5A-11. In response to RAI 427-8444, Question 19.05-5, dated April 16, 2016 (ML16196A361), the applicant stated that the sentence stating the “5 psid (34.5 kPa) barriers are identified on Figures 9.5A-1 through 9.5A-9,” will be deleted because these particular features are security-related, and identifying them within the DCD is not appropriate. The staff finds this response unacceptable because the identification and description of key design features credited in the aircraft impact assessment must be included in the FSAR, as required by 10 CFR 50.150(b). Additionally, security-related information is allowed to be submitted within the DCD, with the appropriate information security markings. In a supplemental response to RAI 427-8444, Question 19.05-5, dated February 9, 2017 (ML17041A211), the applicant provided a DCD mark-up detailing the 5-psid (34.5 kPa) fire barrier locations (walls and floors) in new Figures 19.5-1 through 19.5-10. The response also modifies the pointer (originally proposed to be deleted) to these new 19.5 figures. These 19.5 figures, showing the 5-psid barriers, are to be used in combination with the 9.5A fire barrier figures. The staff finds this response, in conjunction with the existing description of fire barriers and 5-psid fire barriers, to be acceptable, as this completes the reasonable description and identity of the credited 5-psid fire barrier key design features in accordance with 10 CFR 50.150(b). Incorporation of the DCD markup provided was being tracked as a Confirmatory Item 19.5-5. The staff confirmed that the RAI response markup was incorporated into Revision 1 of the DCD. Therefore, the staff considers Confirmatory Item 19.5-5 to be resolved and closed.

These key design features ensure at least one complete train of secondary heat removal equipment and necessary support systems, to include cooling water, electrical power supply and distribution, and instrument and control within the AB and EDGB, is available to provide core cooling following the impact of a large, commercial aircraft. Based on the above review, the staff finds the applicant's description of the fire protection key design features for protecting core cooling equipment to be adequate and acceptable.

19.5.4.3.2 Plant Arrangement and Plant Structural Design Features

In the DCD Tier 2, December 1, 2015 submittal (ML15335A017), and December 1, 2017 letter (ML17335A072), Section 19.5.4.2, the applicant states that the APR1400 plant design and arrangement of major structures as described in DCD, Tier 2 Section 1.2.14, "Plant Arrangement Summary," and Figure 1.2-1, "Typical APR1400 Site Arrangement Plan," through Figure 1.2-27, "General Arrangement Compound Building EL 100 ft.-0 in." are key design features.

Specifically, the applicant states that the AIA credited the arrangement and design of building features that limit the location and effect of potential aircraft strikes on the RCB, AB and EDGB, as detailed in Sections 19.5.4.3.2.1 through 19.5.4.3.2.5, below. The staff's review of the design features and functional capabilities of those individual buildings to demonstrate that the acceptance criteria of 10 CFR 50.150(a)(1) can be met are documented in the following subsections.

The staff reviewed DCD, Tier 2, December 1, 2015 submittal (ML15335A017), subsection 19.5.4.2, DCD, Tier 2, Figures 1.2-1, "Typical Site Arrangement Plan," and 1.2-14, "General Arrangement AB EL. 100'-0," and found that the applicant described the locations of plant structures in items b and c. of Subsection 19.5.4.2 using "the cardinal directions" (e.g.; east and west). The staff could not easily associate the locations of plant structures, since "the cardinal directions" were not identified in the site and plant arrangement figures.

Therefore, the staff issued RAI 450-8528, Question 19.05-6 (ML16084A125), requesting that the applicant either provide "the cardinal directions" in the site and plant arrangement figures, or describe the locations of plant structures without using "the cardinal directions."

In response to RAI 450-8528, Question 19.05-6, dated April 18, 2017 (ML17108A837), the applicant provided mark-up language for DCD Tier 2, Subsection 19.5.4.2, which revised the descriptions of locations of plant structures in order to remove references to cardinal directions. The staff found the mark-up language acceptable, as it describes the location of plant structures without using cardinal directions. Based on the review of the DCD to verify that it incorporated the text as shown in the RAI response, the staff has confirmed incorporation of the changes described above; therefore, RAI 450-8528, Question 19.05-6, is resolved and closed.

19.5.4.3.2.1 Reactor Containment Building Structure.

The staff reviewed the DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the RCB to protect core cooling equipment.

In the DCD Tier 2, December 1, 2015 submittal (ML15335A017), Section 19.5.4.1, the applicant stated that the RCB structure, as described in DCD, Tier 2, Sections 3.8.1 and 3.8.2 and shown on DCD, Tier 2, Figures 3.8-1 and 3.8-2, is a key design feature for the protection of the safety systems, located inside containment, from the impact of a large, commercial aircraft. The staff reviewed general arrangement drawings in DCD, Tier 2, Figures 1.2-1, "Typical APR1400 Site Arrangement," 1.2-2, "General Arrangement Reactor Containment Building Section A-A," and 1.2-3, "General Arrangement Reactor Containment Building Section B-B"; Section drawings (DCD, Tier 2, Figures 3.8-1, "Typical Section of Containment Structures (Looking North)" and 3.8-2, "Typical Section of Containment Structures (Looking East)"; and DCD, Tier 2, Sections 3.8.1, "Concrete Containment," and 3.8.2, "Steel Containment," to verify the adequacy of the description. The staff reviewed the DCD Tier 2 Section 3.8.1, descriptions and figures and finds that the RCB wall is a post-tensioned concrete wall with thinnest thickness at dome being 4.0 feet (1.22 m). The applicant stated, in the DCD Tier 2, December 1, 2015 submittal (ML15335A017), Section 19.5.4.1, that their assessment concluded that a strike upon the RCB would not result in the perforation of the containment, such as to cause direct damage or exposure to jet fuel of the systems within the containment. The assessment also determined that key components located inside the RCB are unaffected by shock-induced vibrations resulting from the impact of a large commercial aircraft.

Based on the above review, the staff finds the applicant's description of the RCB as a key design feature for providing physical protection for maintaining core cooling to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b).

In a letter dated September 29, 2017 (ML17272A599), the applicant provided a markup of DCD Tier 2 Section 19.5.4.1 that described that the nominal compressive strength of the RCB concrete reaching 6,900 psi at 28 days is a key design feature. Section 2.3.1, "Material Properties," of NEI 07-13, Revision 8 uses concrete test strength at 28 days as the basis to consider a concrete strength increase due to the effects of concrete aging. The nominal compressive strength of the RCB concrete at 28 days conforms to Section 2.3.1, "Material Properties," in NEI 07-13, Revision 8. On this basis, the staff finds the applicant's description of the nominal compressive strength of the RCB concrete reaching 6,900 psi at 28 days as a key design feature to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b). Adding this design feature for the nominal compressive strength of the RCB concrete to the DCD was being tracked as Confirmatory Item 19.5-7. Based on the review of the DCD to verify that it incorporated the text as described, the staff has confirmed incorporation of the changes described above; therefore, Confirmatory Item 19.5-7 is resolved and closed.

In letters dated September 29, 2017 (ML17272A599) and December 1, 2017 (ML17335A072), the applicant provided a markup of DCD Tier 2 Section 19.5.4.1 that described that the polar crane bracket and rail girder shown in Figure 19.5-11 are key design features that protect the safety system located inside containment from the drop of polar crane due to the impact of a large, commercial aircraft. The staff reviewed the DCD Tier 2 Section 19.5.4.1 markups and Figure 19.5-11, and finds that the design of polar crane and rail girder would maintain adequate design margins. A summary of design results and material properties is shown in DCD Tier 2 markups as Table 19.5-1. The applicant accounted for the damage to the polar crane as specified in Section 3.3.1, "Damage Rule Sets for Containment Structures," of NEI 07-13, Revision 8. On this basis, the staff finds the applicant's description of the polar crane bracket and rail girder as a key design feature to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b). Adding design features for the polar crane bracket and rail girder to the DCD is being tracked as Confirmatory Item 19.5-8. Based on the review of the DCD to verify that it incorporated the text as described, the staff has confirmed incorporation of the changes described above; therefore, Confirmatory Item 19.5-8 is resolved and closed.

19.5.4.3.2.2 Auxiliary Building Structure

The staff reviewed the DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the AB to protect portions of the RCB, and core cooling equipment.

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Based on the above review, the staff finds the applicant's description, including location and design of the AB as a key design feature for protecting a portion of the RCB from the impact of a large, commercial aircraft to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b).

In letters dated September 29, 2017 (ML17272A599) and December 1, 2017 letter (ML17335A072), the applicant provided a markup of DCD Tier 2 Section 19.5.4.2 that described that the configuration of the AB shown in DCD Tier 2 markups, Table 19.5-2 and Figures 19.5-12 through 19.5-17 is a key design feature for protecting key equipment in the AB. The staff reviewed DCD Tier 2 markups, Table 19.5-2 and Figures 19.5-12 through 19.5-17, and finds that the applicant performed the assessment for the AIA using the NEI 07-13, Revision 8 methodology. Based on results of the assessment, the applicant strategically strengthened measures for some exterior and interior walls in the AB, and the floor slab of the SFP. The staff finds the applicant's description of configuration of the AB as a key design feature to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b). Adding this design feature for the configuration of the AB to the DCD was being tracked as Confirmatory Item 19.5-9. Based on the review of the DCD to verify that it incorporated the text as described, the staff has confirmed incorporation of the changes described above; therefore, Confirmatory Item 19.5-9 is resolved and closed.

In a letter dated September 29, 2017 (ML17272A599), the applicant provided a markup of DCD Tier 2 Section 19.5.4.2 that described that the nominal compressive strength of the AB concrete reaching 5,900 psi at 28 days is a key design feature. Section 2.3.1, "Material Properties," of NEI 07-13, Revision 8 uses concrete test strength at 28 days as the basis to consider a concrete strength increase due to the effects of concrete aging. The nominal compressive strength of the AB concrete at 28 days conforms to Section 2.3.1, "Material Properties," in NEI 07-13, Revision 8. On this basis, the staff finds the applicant's description of the nominal compressive strength of the AB concrete reaching 5,900 psi at 28 days as a key design feature to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b). Adding this design feature for the nominal compressive strength of the AB concrete to the DCD is being tracked as Confirmatory Item 19.5-10. Based on the review of the DCD to verify that it incorporated the text as described, the staff has confirmed incorporation of the changes described above; therefore, Confirmatory Item 19.5-10 is resolved and closed.

19.5.4.3.2.3 Emergency Diesel Generator Building Structure

The staff reviewed the DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the EDGB to protect portions of the AB, and core cooling equipment.

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Based on the above review, the staff found the applicant's description, including location and design, of the EDGB as a key design feature for protecting a portion of the AB from the impact of a large, commercial aircraft to be acceptable because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b). Adding Table 19.5 and Figure 19.5-18 to the design feature was being tracked as Confirmatory Item 19.5-11. Based on the review of the DCD to verify that it incorporated the text as described, the staff has confirmed incorporation of the changes described above; therefore, Confirmatory Item 19.5-11 is resolved and closed.

19.5.4.3.2.4 Physical Separation of Emergency Diesel Generator.

The staff reviewed the DCD to ensure that the applicant describes the physical separation of EDGs and identifies the physical separations as key design features in accordance with 10 CFR 50.150(b).

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The staff reviewed DCD, Tier 2, Sections 3.8.4.1.1, "Auxiliary Building," 3.8.4.1.2, "Emergency Diesel Generator Building," 3.8A.2.1, "Structural Description and Geometry," and DCD, Tier 2, Figures 1.2-1, 1.2-14, 1.2-20, and 1.2-21. The staff verified that the AB houses two EDG areas at the far west corners of the AB. An independent EDGB structure housing two more EDGs is located on the outside of the AB, immediately adjacent to the outlying east wall. All EDGs are located at grade level. The staff confirmed that the applicant credited the interior wall by limiting the window size on viewing areas in the A and B diesel general control room based on the damage rule set in Section 3.3.2, "Damage Rule Sets for Reinforced Concrete Buildings," of NEI 07-13, Revision 8.

Based on the above review, the staff found the applicant's description of the physical separation and window size as key design features for providing electrical power for maintaining core cooling from an impact of a large, commercial aircraft to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b). Adding this design feature for the window size on viewing areas in A and B diesel generator control room to the DCD was being tracked as Confirmatory Item 19.5-12. Based on the review of the DCD to verify that it incorporated the text as described, the staff has confirmed incorporation of the changes described above; therefore, Confirmatory Item 19.5-12 is resolved and closed.

19.5.4.3.2.5 Properties of Concrete and Reinforcement Bars.

The staff reviewed the DCD to ensure that the applicant describes the properties of concrete and reinforcement bars that are used to protect core cooling capability in accordance with 10 CFR 50.150(b).

In the DCD Tier 2, December 1, 2015 submittal (ML15335A017), Section 19.5.4.2, item d, the applicant states that properties of concrete and reinforcement bars, as described in Appendix 3.8A, are key design features in protecting key safety equipment in the AB and EDGB.

The staff reviewed DCD, Tier 2, Appendices 3.8A.2.2, “Structural Materials,” and 3.8A.3.2, “Structural Materials,” and finds the materials used in the construction of the AB and EDGB are high strength concrete (91 days compressive strength is 5,000 psi) and American Society for Testing and Materials (ASTM) A615 Grade 60 high strength steel with 90 kilopounds per square inch (ksi) tensile strength and 60 ksi yield strength. The staff confirmed that these properties meet the minimum requirements for physical damage rule sets as shown in Table 3-2 of NEI 07-13, Revision 8.

However, the staff noticed that the DCD Tier 2, December 1, 2015 submittal (ML15335A017), Section 19.5.4.2, item d, only discussed the AB with regard to properties of concrete and reinforced bars providing protection of key design features. In RAI 424-8532, Question 19.05-2, item b, the staff requested that the applicant provide similar property information for the EDGB. See Section 19.5.4.2 of this SER for resolution of this inquiry. This issue was being tracked as Confirmatory Item 19.5-2. Based on the review of the DCD to verify that the DCD markup in the response to RAI 424-8532, Question 19.05-2, item d, was incorporated into the text as shown in the RAI response, the staff has confirmed incorporation of the changes described above; therefore, Confirmatory Item 19.5-2 is resolved and closed.

Based on the above review, the staff finds the applicant’s description of the material properties of concrete and steel used in the construction of the AB and EDGB as a key design feature for providing protection of key equipment inside the AB and EDGB for maintaining core cooling and spent fuel pool integrity from an impact of a large, commercial aircraft to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b).

19.5.4.3.3 Shock Damage

In the DCD, December 1, 2015 submittal (ML15335A017), Tier 2 Section 19.5.2, “Scope of Assessment,” the applicant states that the analysis assessed shock-induced vibration on systems, structures, and components (SSC) from a large, commercial aircraft impact. In addition, the applicant states in DCD Tier 2 Section 19.5.4.1 of the December 1, 2015 submittal (ML15335A017), “RCB and SFP,” that the assessment determined that key components located inside the RCB, including the reactor pressure vessel, steam generators, reactor coolant loop piping, pilot operated safety relief valves, control element drive mechanism, the safety injection and shutdown cooling system suction line motor operated valves, discharge line check valves, and instrumentation and control equipment associated with core cooling are not adversely affected by shock-induced vibrations resulting from the impact of a large commercial aircraft.

Based on the applicant’s use of the NRC endorsed guidance document NEI 07-13, Revision 8, and the assessment scope that includes shock vibration, the staff finds that the applicant has performed a reasonably formulated shock analysis within the aircraft impact assessment.

19.5.4.4 ***Spent Fuel Pool Integrity***

In the DCD, December 1, 2015 submittal (ML15335A017), Tier 2 Section 19.5.4.1, the applicant states that the design and location of the SFP and its supporting structures are key design features for protecting the SFP’s structural integrity. In addition, the applicant indicates that, based on the design as described in DCD, Tier 2, Sections 3.8A.2, “Auxiliary Building,” and 9.1.2, “New and Spent Fuel Storage,” including DCD, Tier 2, Figure 3.8A.45, the AIA performed by the applicant’s qualified personnel determined that there are no aircraft impact scenarios that will perforate the pool liner, and that all SFP piping attachments are configured such that they will not result in leakage from the spent fuel pool below the required minimum water level.

The staff also reviewed DCD, Tier 2 Section 9.1.3.1, “Design Bases,” item g, which states that to preclude loss of SFP minimum water level, which provides proper shielding, all piping that penetrates the pool is located approximately 10 feet (3.05 meters) above the top of the spent fuel assemblies, and all piping extending down into the pool have siphon breaker holes at or above this level. This description confirms that all spent fuel pool piping attachments are configured such that they will not allow water in the spent fuel pool to drain below the minimum water level, should the piping attachments suffer damage due to the aircraft impact.

Based on the above review, the staff finds the description of the key design features for ensuring spent fuel pool integrity to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR 50.150(b).

19.5.5 Combined License Items Identified in the DCD

Item No.	Description	Section
19.5(1)	When the reactor head is untensioned and before the refueling pool is flooded up, administrative controls will be in place to ensure that no trains of safety injection and shutdown cooling, including the necessary power and cooling water support systems, are out of service for maintenance.	19.5
19.5(2)	The piping layout of safety-related cooling water systems (CCW and ECW) are to be designed so that piping failure from an aircraft impact shall not cause the total loss of cooling capability.	19.5

The staff finds the above list of COL information items to be complete, and adequately describes the actions necessary for the COL applicant. No additional COL information items need to be included in DCD Tier 2 Table 1.8-2 related to aircraft impacts.

19.5.6 Conclusions

The staff finds that the applicant has performed an AIA that is reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the acceptance criteria 10 CFR 50.150(a)(1) are met.

The staff also finds that the applicant adequately described the key design features and functional capabilities identified and credited to meet 10 CFR 50.150, including descriptions of how the key design features meet the acceptance criteria in 10 CFR 50.150(a)(1). This includes describing how the facility can withstand the effects of a large commercial aircraft impact such that the reactor core remains cooled and spent fuel pit integrity is maintained. Therefore, the staff finds that the applicant meets the applicable requirements of 10 CFR 50.150(b) and 10 CFR 52.47.