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 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 the SECY-90-016 guidance, 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 APR1400 DCD lists the external reactor vessel cooling system (ERVCS) as one of the systems provided to mitigate the consequences of core damage and prevent containment failure. Although the applicant did not credit it in the baseline analysis and sensitivity analyses showed marginal benefit, the applicant proposed to update APR1400 DCD Section 19.1.4.2.2.6 to explain the marginal benefit from crediting ERVCS (ADAMS Accession No. ML15254A448).

Although, the ERVCS provided marginal benefit in the sensitivity analysis, the staff finds that it provides defense-in-depth.

The applicant has identified containment features, mitigating systems, and human actions that are available to prevent accidents and mitigate their consequences. The PRA is used to assess the associated risk. The staff finds that this meets the requirements of 10 CFR 52.47(a)(4).

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:

modify to "fission product **scrubbing**"

- containment spray system (CSS): In the event of a containment leakage, CSS provides fission product 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.

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 PRA Standard, to the extent possible (see discussion in Section 19.1.2.3). The APR1400 PRA was developed using a small-event-tree method supported by a linked fault tree approach.

The DCD Section 19.1.4.1 describes the APR1400 Level 1 at-power internal events PRA and results. The section references the APR1400 PRA Summary Report, which was submitted along with the application. It summarizes the APR1400 PRA development and results found in the APR1400 PRA Notebooks, which are KHNP's internal documentation of the APR1400 PRA. The staff reviewed Section 19.1.4.1 and the PRA Summary Report in their entirety. The staff reviewed in their entirety the applicant's PRA notebooks to understand the approach and process the applicant used in developing its PRA. Below is a description of the staff's review and findings for Initiating Events Analysis, Accident Sequence Analysis and Success Criteria Analysis.

Summary of Application

The DCD Section 19.1.4.1 of the application describes the Level 1 internal events PRA for operations at power, including results and insights:

- 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.

SSIEs were modeled as point estimates using generic data.

The applicant used NUREG/CR-6928, "Industry-Average Performance for components and Initiating Events at U.S. Commercial Nuclear Power Plants" (Feb. 2007) to identify potential initiating events as a starting point for a generic list of initiators. The list was further developed and screened to include design-specific initiating events. Support system initiating events were identified and modeled using the fault tree approach. The applicant used generic data from NUREG/CR-6928 and other generic sources to estimate the frequency of each initiating event. In cases where the applicants' calculated support system initiating events were higher than the generic data, the applicant used the calculated initiating event frequency. The applicant developed 25 event trees covering all the initiating events identified to be applicable to the APR1400 design. The event trees provided accident sequences that led to core damage by questioning the ability of mitigating systems and operator actions to respond to each initiating

event. The applicant showed that for all initiating events, the plant can 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 (MAAP and RELAP) to develop success criteria for mitigating systems and human actions required to respond to initiating events. The success criteria analysis defined key safety functions similar to 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 application reports the mean CDF from Level 1 internal events at power as 1.9×10^{-6}/year. Loss of offsite power and SBO initiating events strongly dominate the internal events CDF (40%). The next largest contributors to plant risk are the total loss of cooling water events (12%). The medium break loss-of-coolant accident contribution is also significant (10 percent).

Initiating Event Analysis

SSIEs were modeled as point estimates using generic data.

The staff reviewed the list of initiating events and compared it to the generic events list in NUREG/CR-6928 for completeness. The applicant did not identify any design-specific initiating events. The events were grouped into LOCA, Transients, ATWS and SGTR resulting in 25 initiating events. The grouping was accomplished by combining events that require similar plant responses and that have similar consequences. The initiating event list included support system initiating events. Because support systems are specific to each plant or design, the applicant performed analysis to determine the design specific support system initiating event. Each initiating event was assigned a frequency from NUREG/CR-6928, except for support system initiating events for which the applicant's analysis resulted in higher frequencies above the available generic data.

Accident Sequence Analysis

The staff reviewed each event tree used to describe design-specific scenarios that can lead to core damage to ensure they addressed mitigating system response, operator actions and recovery actions that support key safety functions within the defined mission time. The staff reviewed how the applicant modeled dependencies that could affect the ability of mitigating systems to operate and function. The staff also reviewed a sample set of accident sequences for completeness based on the significant contributors to risk and experience with PRA's for similar reactor designs. The sample set included sequences for the Loss of Offsite Power, Small LOCA, General transients and Loss of Main Feedwater.

Success Criteria Analysis

The staff reviewed the success criteria specified for the mitigating systems and human actions to ensure consistency with the design features, procedures and operating philosophy. The staff also reviewed the thermal-hydraulic results used to support the success criteria analysis and compared the success criteria to those determined for the design basis accidents documented in Chapter 15 of the DCD.

System Analysis

Six causes of system failure and unavailability modes were represented in the initiating events analysis and sequence definition. The causes were random component failures, outages for maintenance and testing, support systems, common cause failures (CCFs), 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 Technical Specifications in Chapter 16 of the DCD. Supporting requirement SY-A15 of the ASME/ANS PRA Standard was used to screen components or specific component failure modes.

The applicant lists the systems modeled in the APR1400 design-specific PRA in DCD Table 19.1-9. Simplified diagrams of the major systems are shown in DCD Figures 19.1-1 through 19.1-14.

System dependencies were explicitly considered. The applicant included two tables to summarize initiator-to-system dependencies. DCD Table 19.1-10 summarizes the dependency between initiating events and front line systems. DCD Table 19.1-11 summarizes the dependency between initiating events and support systems.

change to "DCD Table 19.1-10a summarizes initiating event/front line system dependencies, DCD Table 19.1-10b summarizes initiating events/support systems, DCD Table 19.1-11a summarizes front line/support systems, and DCD Table 19.1-11b summarizes support system/support systems."

Data Analysis

Data analysis performed by the [illegible] unavailability data, common cause analysis, and special event data. For each component type and failure mode identified in the system analysis, the failure rates were extracted from available generic data sources listed in Section 19.1.4.1.1.5 of the DCD.

Table 19.1-14 of the DCD provides the component failure data for the APR1400 PRA. The majority of this 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."

Component Unreliability Data: The applicant used generic data to evaluate component unreliability derived from NUREG/CR-6928.

Component Unavailability Data: Component unavailability means the component is in an out-of-service state due to test and maintenance (T&M). T&M events are modeled in the PRA using generic component unavailability data derived from NUREG/CR-6928 in the absence of plant data.

Common Cause Analysis: The applicant applied the Alpha Factor methodology to calculate the probability of common cause events. 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.

Special Event Data: Special events were developed to reflect potential plant scenarios. The special events and their associated probabilities are listed in DCD Table 19.1-16. The special

events are no adverse moderator temperature coefficient, reactor cooling pump (RCP) seal LOCA probability after success of secondary heat removal, non-recoverable probability of offsite power within 16 hours after switchyard-centered LOOP, non-recoverable probability of offsite power within 16 hours after grid-related LOOP, non-recoverable probability of offsite power within 16 hours after weather-related LOOP, conditional LOOP upon transients, and conditional LOOP upon LOCA initiators.

Non-recovery probability was also calculated for plant-centered LOOP.

Human Reliability Analysis

Human reliability analysis (HRA) performed by the applicant was done in conformance with the ASME/ANS PRA Standard requirements, as clarified by RG 1.200. The assessment included pre-initiator HFEs and post-initiator HFEs.

Pre-Initiator Actions - The pre-initiator HRA for this design are Type A, or latent, HFEs. Latent HFEs are events that take place prior to an initiating event. The applicant developed pre-initiator actions consistent with the Accident Sequence Evaluation Program (ASEP) framework described in NUREG/CR-4772. The applicant defined significant HFEs in accordance with RG 1.200. Three levels were used to assess dependency between multiple human errors. The applicant utilized an enhanced methodology in accordance with SR HR-A3 and HR-B2 from the ASME/ANS PRA Standard by assigning low dependence (LD) rather than zero dependence (ZD) to activities that impact more than one system train.

Post-Initiator Actions - Post-initiator actions were identified using plant-specific emergency operating guidelines (EOGS) along with information from previous and similar reactor designs. The post-initiator actions considered for the applicant's PRA primarily involved in-control-room actions. The applicant used cause-based decision tree methodology (CBDTM), human cognitive reliability / operator reliability experiment (HCR/ORE) methodology, and annunciator response model (ARM) depending on the particular scenario.

The applicant used the HCR/ORE methodology for HFEs in situations where cognition must occur within a short time frame and for which the proposed scenarios were similar to the original scenarios. This technique implicitly includes performance shaping factors (PSFs). The applicant used CBDTM for HFEs in which cognition can occur at some time greater than 30 minutes. ARM is suggested for parts of HRA where the primary interest is in responding to annunciators. THERP was used to estimate the failure probability for the execution steps.

Quantification

SAREX and CAFTA computer codes

The applicant used the SAREX computer code to develop event trees and fault trees and used the FTREX computer code to solve the PRA model for the APR1400 design CDF. The EPRI HRA calculator was used to calculate the Human Error Probabilities. The results were analyzed to identify the significant contributors to the CDF from initiating events, accident sequences and basic events. The applicant performed post-processing to address cutsets with multiple HEPs.

to address HEP dependency within cutsets

HEPs were calculated by hand using techniques used in the HRAC. HRAC is only being credited in the PRA update which is not yet published.

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."

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 using the relevant sections of the ASME/ANS PRA standard. The staff notes 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 review focused on ensuring that the internal flooding risk is not underestimated and that sufficient risk insights are identified to support the DC application. In particular, the staff emphasized the review of the flooding protection design features assumed in the internal flooding PRA. The staff also focused on the completeness of the applicant's internal flooding PRA by reviewing the development of the internal flooding sources including the screening process.

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 78 foot elevation

the barriers are sealed for a flood only to the 64-foot elevation.

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 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 staff review found that the descriptions for the top internal flooding sequences are not readily available in the DCD for the staff to check for reasonableness. On February 23, 2016, the staff issued RAI 8348, Question 19-43 to ensure that the top flooding sequences are clearly documented in the DCD (ADAMS Accession No. ML16054A291). **This is an open item being tracked as Open Item 19.1-43.**

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 the isolation of fire protection line break in less than 20 minutes and operation of the ECW pumps.

up to the 64-foot elevation only

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 elevation 78 foot level 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.

Due to Open Items 19.1-43, 19.1-45, 19.1-47, and 19.1-91, the staff is unable make a finding on the applicant's internal flooding risk results and insights.

19.1.5.4 Other External Events Risk Evaluation

Summary of Application

The applicant identified site-specific attributes for other external events, described in Chapter 2 of the DCD. 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.

hierarchy and how this assumption may impact the PRA (ADAMS Accession No. ML16054A291). The applicant provided a response 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 (ADAMS Accession No. ML16231A501). The staff finds that the proposed approach reasonable. **RAI 8348, Question 19-50 is being tracked as Confirmatory Item 19.1-50.**

Similar to the calculation of LRF for LPSD internal events, the uncertainty calculation is only performed on POSs 4B-12A. The uncertainty results shown in DCD section 19.1.6.3.2.4 is only for POSs 4B-12A.
The total LRF for LPSD fires is 1.3 x 10-7 per year, as presented in DCD section 19.1.6.3.2.

LPSD Internal Fire Risk Results and Insights

The applicant reported the following LPSD internal fire CDF results:

5 percent value:	8.7×10^{-7} per year
Mean value:	1.7×10^{-6} per year
95 percent value:	3.6×10^{-6} per year

The applicant reported the following LPSD internal fire LRF results:

5 percent value:	2.5×10^{-8} per year
Mean value:	7.1×10^{-8} per year
95 percent value:	1.7×10^{-7} per year

The applicant estimated the above CDF and LRF using reasonably conservative assumptions and crediting the key design features of the APR1400 design. The applicant provided sufficient evaluation of the uncertainties associated with key assumptions. The staff also found the applicant's FPRA model to be sufficiently consistent with guidance in NUREG-6850 applicable to design certification and SRP Section 19.0, and therefore, acceptable. Therefore, the staff finds that the APR1400 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 30 percent to the LPSD internal fire LRF. Other risk-significant fire compartments include corridor F078-A19B and general access area F120-AGAD. Fires in the essential service water building and the component cooling water heat exchanger building are moderately risk-significant. The staff review found that the descriptions for the top internal fire sequences are not readily available in the DCD for the staff to check for reasonableness. On February 23, 2016, the staff issued RAI 8348, Question 19-43 to ensure that the top internal fire sequences are clearly documented in the DCD (ADAMS Accession No. ML16054A291). **This is an open item being tracked as Open Item 19.1-43.**

In terms of the plant operating state, hot mid-loop condition (POS5) contributes about 7.7×10^{-7} per year or about 46 percent to the LPSD internal fire CDF and about 2.3×10^{-8} per year or about 20 percent to the LPSD internal fire LRF.

The applicant's LPSD internal fire PRA identified the risk-significant equipment failure events and operator actions. The results showed that the IRWST sump plugging, 125 VDC bus faults, and failure of barriers between fire compartments F120-AGAC and F120-AGAD or between

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 Chapter 6 of the DCD.

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, Rev. 3 states that the expectation in SECY-93-087 with respect to the deterministic containment performance assessment is as follows:

> The containment should maintain its role as a reliable, leaktight barrier (e.g., by ensuring that containment stresses do not exceed American Society of Mechanical Engineers (ASME) Service Level C limits for metal containment or factored load category for concrete containments) for approximately 24 hours following the onset of core damage under the most likely severe accident challenges, and following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

Consistent with staff guidance, the APR1400 containment consists of design features to mitigate severe accidents and meet Commission's deterministic containment performance assessment goal of 0.1 CCFP.

The APR1400 DCD references different containment pressure capacities without justification: 163, 123, and 184 psig in Sections 19.1, 19.2 and 19.3, respectively. Therefore, in RAI 8325, Question 19-26 the staff asked the applicant to justify these differences. In the responses, dated April 16,2016 (ADAMS Accession No. ML16107A067) and August 4, 2016 (ADAMS Accession No. 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 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

- 184 psia referenced in DCD Section 19.3 (Beyond Design Basis External Event) is the ultimate pressure capacity of the containment as described in DCD Section 3.8.1.4.11, which is in conformance with RG 1.216 Position 1.

Table 19.2-1 Summary of Severe Accident Phenomena

Severe Accident Phenomena	Severe Accident Mitigation Features	Key Results and Conclusions from Accident Analysis
External Reactor Vessel Cooling	Cavity Flooding System (CFS) 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 Passive Autocatalytic Recombiners (PARs) Igniters installed in the containment 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%.
MCCI and Core Debris Coolability	Cavity Flooding System Reactor Cavity Design Holdup volume tank (HVT) and flooding valves	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).

Severe Accident Phenomena	Severe Accident Mitigation Features	Key Results and Conclusions from Accident Analysis
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).
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	Power Operated Safety Relief Valves (POSRV) Design features to address ISLOCA are summarized above in Section 19.2.2.6 of this safety evaluation report.	Manual actuation of POSRVs will reduce pressure in the reactor vessel below 17.6 kg/cm^2 (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: 1200 K (1,700 °F) Temperature–long term: 460 K (368 °F) Pressure 7.75 kg/cm^2 (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.

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

MCCI and Core Debris Coolability

cavity

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 lower drywell depends upon the amount of core debris, its spreadability, and the lower drywell floor area. Once on the drywell 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:

- Provide reactor cavity floor space to enhance debris spreading
- Provide a means to flood the reactor cavity to assist in the cooling process
- Protect the containment liner and other structural members with concrete, if necessary
- 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 Reactor 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.

APR1400 design consists of two features, reactor cavity design and cavity flooding system (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 finds this approach consistent with the SECY-90-016 guidance and acceptable.

The containment liner plate in reactor cavity area is embedded 0.91 m (3 ft.) below from the cavity floor at the minimum. Actuation of CFS can provide flooding up to 6.4 m (21 ft) above the reactor cavity floor (EL. 69 ft 0 in) enabling cooling of core debris if spread on the reactor cavity.

b. Reasons for second deceleration and subsequent acceleration of the melt jet shown in APR1400-E-P-NR-14003-P, Figure 3-2(a),

c. The initial void fraction assumed for the melt jet, and

d. Justification for the triggering time assumed.

The applicant responded on February 27, 2017 (ADAMS Accession No. ML17058A241). In response to Part (a) of the RAI, 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 adjusting the radial mixing zone.

ARIY

The applicant chose the nodal area for the calculations (i.e., AIRY) 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. The applicant used the AIRY value corresponding to the energy index resulting in highest pressure and impulse load shown on the figure, which the staff finds acceptable. The staff finds the applicant's use of one-dimensional TEXAS computer code conservative and acceptable.

In response to Part (b) of the RAI 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 applicant responded on January 23, 2017 (ADAMS Accession No. ML17023A329). In response to Parts (a) and (b) of the RAI, the applicant stated that it chose the nodal area for the calculations (i.e., AIRY) 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. The applicant did not use the AIRY 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 not choosing lower than optimal AIRY value:

ARIY

> 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-P, 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 APR1400 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.

For reference, the applicant performed a structural integrity assessment for the reference plant under higher peak pressure from steam explosion than peak pressure in the transition region. The results showed that the maximum stress of the liner plates was 55.8 ksi, which is less than the ultimate tensile strength (75 ksi); and the maximum effective plastic strain was approximately 1.4%, which is less than the plastic strain criteria (5%).

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, LBLOCA 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 LBLOCA scenario, and lower short term and long term temperatures for the TLOFW scenario. The results of the LBLOCA analysis is found in ERI/NRC 16-208 report (ADAMS Accession No. ML16314E431), "Assessment of Combustible Gas Control during Severe Accidents in APR1400," Figure 4.1 through Figure 4.6, temperatures in various locations in containment during large scale combustion. The results of the TLOFW analysis is 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 LBLOCA, 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 (LBLOCA, 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 LBLOCA 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 110 psia, the staff finds the containment pressure meets the equipment survivability acceptance criteria of 10 CFR 50.44(c)(3).

123.7 psia

To determine the severe accident radiation conditions, the applicant selected the bounding radiation dose of 4.4E+05 Gy (4.4E+07 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, LBLOCA, and TLOFW. Results from these calculations at 24 hours after the accident range from 0.3E+05 to 4.4E+05

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 Report (Reference 44) 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 Report (Reference 44) 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 (ADAMS Accession No. ML16309A031). **This is being tracked as Open Item 19.2-53.**

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.

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 or 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 92-087 and RG 1.216, the applicant must 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

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In Appendix B of APR1400-E-P-NR-14005-P, 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-P, 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."

and

delete

The DCD Tier 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 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 10 ft above the fuel assemblies. The level instrument, as described in DCD Tier 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-P, 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 50, Appendix A, GDC 2. 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-P shows the connection for SFP makeup and spray line. Figure 6-4 of APR1400-E-P-NR-14005-P 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 to a location above the SFP at EL. 156 ft. 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(5). Establishing procedures and guidance is included in COL 19.3(5). 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 NRC staff reviewed the DCD Tier 2 and APR1400-E-P-NR-14005-P regarding SFP mitigation strategies including onsite equipment, FLEX equipment, connections to the portable equipment as discussed above, and identified the following RAI.

<u>Non-seismic Piping Connected to the SFP</u>

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In Appendix B of APR1400-E-P-NR-14005-P, 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 7991 Question 09.01.03-1 and Question 09.01.03-~~4~~**) indicated that additional justification may be needed for the above statement, which resulted in a follow-up question, **RAI 8670, Question 19.03-~~40~~.**

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