RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**432-8377**
SRP Section:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation Section: 19**
Application Section:	**19**
Date of RAI Issue:	**03/08/2016**

Question No. 19-65

10 CFR 52.47(a)(23) states that a design certification (DC) application for light-water reactor designs must contain an FSAR that includes 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. Revise the design control document (DCD) accordingly.

APR1400 DCD Rev. 0, Section 19.2.3.3.3.3.2 states the following:

> The limiting case for MCCI analysis is large-break LOCA with 100 percent core relocation into the reactor cavity. For the large-break LOCA scenario, corium is predicted to be quenched in the reactor cavity sump before the depth of concrete ablation reaches the buried containment liner. This sequence conservatively assumes early relocation of 100 percent of the core inventory into the containment.

a. Add text to the DCD to describe the molten core-concrete interaction (MCCI) analysis for the cavity sump including key assumptions, methodology, and key results.

b. The applicant provided for staff audit “Ex-Vessel Severe Accident Analysis for the APR1400 with the MELTSPREAD and CORQUENCH Codes,” dated August 28, 2012, which attributes the cooling and eventual quenching of debris in the reactor cavity sump to the cooling mechanism of melt eruption. This cooling mechanism involves gasses generated by corium-concrete interaction causing the surface crust to erupt allowing water to penetrate and cool the debris. Justify the applicability of the melt eruption models used for the APR1400 debris coolability calculations for the cavity sump considering that the APR1400 cavity sump has larger dimensions than the testing on which the models are based.

Response

a. The DCD will be revised. The markups and revised text are included in Attachment.

b. CORQUENCH has been validated against the OECD/MCCI 2-D test database which consists of 400 – 1,000 kg tests with H/D ratios ranging from 0.43 to 0.6, a little less than that of the H/D ratio for the APR1400 sump of 0.86. The best test for scaling to the APR1400 conditions is the CCI-6 experiment (see Appendix C of the CORQUENCH3.03 manual, Reference 16 in DCD 19.2.8); this experiment involved 1,000 kg melt interacting with low gas content siliceous concrete (worst case for the melt eruption cooling mechanism). In that experiment, a floating crust boundary condition was obtained, which is the expected boundary condition at plant scale. In terms of validation, CORQUENCH under-predicts the amount of core debris stabilized by the melt eruption cooling mechanism by 20 % for this experiment using the same set of modeling assumptions that are applied in the APR1400 analysis (i.e., Ricou-Spalding melt entrainment correlation with the empirical constant set at 0.08). The fact that the code under-predicts the coolability test data for the experiment indicates that the prediction for APR1400 is likely conservative insofar as the extent of cavity ablation is concerned. It is not feasible to consider testing at the scale involved with the APR1400 sump which involves ~35,000 kg of core debris at a depth of ~1.6 m and area of ~2.9 m^2 (equivalent diameter of ~1.9 m), with a height/diameter (H/D) ratio of 0.86.

Impact on DCD

The DCD Tier 2 Section 19.2.3.3.3.2 will be revised as shown in Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

sequences as well as a LBLOCA sequence. Each sequence is run with a flooded reactor cavity.

Debris coolability in the sump is evaluated using CORQUENCH for a conservative LBLOCA sequence.

Replace this text with the text "A" in the next page.

19.2.3.3.3.3 Analysis Result

The corium in the APR1400 reactor cavity is quenched, and the integrity of containment liners is maintained when the CFS is available, based on the analyses presented in this subsection. This is due to the ample corium spreading area in the reactor cavity, which allows for sufficient heat transfer from the corium pool into the overlying pool of water and thus prevents the ablation front from reaching the containment liner plate.

19.2.3.3.3.3.1 CORQUENCH Result for MCCI in the Reactor Cavity

For the MCCI analysis in the reactor cavity, the conservative large-break LOCA (LBLOCA) scenario is calculated by CORQUENCH. This sequence conservatively assumes early relocation of 100 percent of the core inventory into the containment and that no jet breakup occurs when the core debris relocates into the flooded reactor cavity. The depth of concrete ablation in the reactor cavity for the conservative LBLOCA scenario was predicted to be 0.27 m (0.86 ft) by CORQUENCH.

19.2.3.3.3.3.2 CORQUENCH Results for MCCI in the Reactor Cavity Sump

The limiting case for MCCI analysis is large-break LOCA with 100 percent core relocation into the reactor cavity. For the large-break LOCA scenario, corium is predicted to be quenched in the reactor cavity sump before the depth of concrete ablation reaches the buried containment liner. This sequence conservatively assumes early relocation of 100 percent of the core inventory into the containment.

Replace this text with the text "B" in the next page.

19.2.3.3.3.3.3 MAAP Results for MCCI in the Reactor Cavity

The largest amount of concrete erosion in the reactor cavity is predicted to occur for the large-break LOCA scenario. This scenario models a large-break LOCA with MAAP

A

MCCI in the reactor cavity sump was analyzed using the CORQUENCH code for a conservative LBLOCA sequence. The initial conditions used in these calculations were:

1) The sequence initiator is a Large LOCA. To add more conservatism, the core support plate is assumed to fail early, dumping the entire core into the RPV lower plenum and leading to early reactor vessel failure.
2) 100% of the core inventory of UO_2 and Zr is relocated from the vessel into the cavity at the time of vessel failure.
3) The oxidation fraction of Zr is about 52%.
4) The initial temperature of the melt is about 2500 K.

The CORQUENCH calculation is based on a very conservative assumption, which ignores the particle bed created during the initial jet breakup phase. This means that the particle bed can only be created through melt eruption when MCCI occurs. The CORQUENCH calculation investigates containment pressures of 2 bar and 4 bar for both Limestone and LCS (Limestone and common sand) concrete types.

B

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 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.

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Question No. 19-66

10 CFR 52.47(a)(23) states that a design certification (DC) application for light-water reactor designs must contain an FSAR that includes 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. Revise the design control document (DCD) as necessary.

The following refer to APR1400-E-P-NR-14003-P, "Severe Accident Analysis Report," Rev. 0, Appendix C-1, "Severe Accident Analysis Report for HPME/DCH:"

a. Section 4.2.1 states that "the initial mass of UO_2 in melt at vessel breach, the fraction of Zr oxidized and variations in the coherence ratio were quantified as probability density curves." However, APR1400-E-P-NR-14003-P does not provide any probability density curves used. Provide probability density curves used for the initial mass of UO_2 in melt at vessel breach, the fraction of Zr oxidized and variations in the coherence ratio.

b. Section 4.2.2 lists 26 input parameters without their values used for analysis. Provide input values used for these parameters.

Response

a. Figure 1 through 5 show the probabilistic density curves used for the initial mass of UO_2 in melt at vessel breach, the fraction of Zr oxidized and variations in the coherence ratio. The detailed information for each variable is described as below.

   1. Mass of UO_2 in the Lower Plenum

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TS

TS

2. Fraction of Zr Oxidized

TS

3. Variation in the Coherence Ratio

TS

b. The values of 26 input parameters listed in Section 4.2.2 are shown as Table 4.

Table 1 Distribution of Molten Mass of UO_2 in the Lower Plenum for Scenarios V and Va

TS

Table 2 Distribution of Molten Mass of UO_2 in the Lower Plenum for Scenarios VI

TS

Table 3 Distribution of Oxidation Fraction of Initial Zr Inventory in the Core

TS

Table 4 Input Parameters for V, Va and VI Scenarios in DCH Analysis

TS

TS

Figure 1 Distribution of Mass of UO2 in the Lower Plenum for Scenario V and Va

TS

Figure 2 Distribution of Fraction of Zr Oxidation for Scenario V and Va

TS

Figure 3 Distribution of Molten Mass of UO_2 in the Lower Plenum for Scenario VI

TS

Figure 4 Distribution of Fraction of Zr Oxidation for Scenario VI

TS

Figure 5 Variation in the Coherence Ratio for Scenario V, Va, and VI

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

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Question No. 19-67

10 CFR 52.47(a)(23) states that a design certification (DC) application for light-water reactor designs must contain an FSAR that includes 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. Revise the design control document (DCD) as necessary.

a. APR1400 DCD Rev. 0, Section 19.2.3.3.3.2, provides a list of phenomena for which CORQUENCH models were used to tune MAAP model parameters for analyzing molten core concrete interactions in the reactor cavity. Describe how these phenomena were captured in modeling with MAAP.

b. Provide molten core-concrete interaction (MCCI) results for a case with no overlying water present in the cavity.

c. APR1400 DCD Rev. 0, Figure 19.2.3-7 has a caption “Ablation Depth in Floor and Sidewall for the PRA Sequence of Loss of Essential Service Water.” However, the figure also has a title “Loss of AC power with short battery life.”

Response

a. MAAP 4.0.8 only considers two main phenomena related to ex-vessel corium coolability, initial jet breakup and water ingression.

   Initial jet breakup is the phenomenon occurring when a corium jet flows from the vessel into a deep water pool. Corium particles are stripped off from the corium jet due to the Kelvin-Helmholtz instability at the vapor/liquid interface. These particles can eventually settle on top of the upper crust of corium pool, forming a layer referred to as the particle bed. As a

result of this process, the mass of the molten corium pool can be significantly reduced, making the MCCI less severe.

Water ingression is the phenomenon occurring once corium is fully settled on the cavity floor. As the corium is covered by water, the large heat removal from the top surface can produce a thick upper crust. According to experimental data, the heat removal rate by water in this configuration far exceeds the maximum conduction heat transfer rate through the upper crust layer, if the layer were assumed to be continuous and impermeable. The upper crust is not a continuous, impermeable layer; instead, numerous deep cracks develop, as the solidified corium in the upper crust is subjected to a large temperature gradient. Water can ingress (infiltrate) into the corium through these cracks, contacting the hotter region of the corium. As the solidification progresses, both the temperature gradient and the cracks extend deeper into the corium pool.

The model in MAAP 4.0.8 evaluates the heat transfer and oxidation of metal during the process of the initial jet breakup. However, it does not simulate the formation of the particle bed on top of the upper crust. MAAP 4.0.8 assumes that the particles formed during the initial jet breakup are promptly merged into the corium pool. The rate of mass stripping from a jet is calculated using a formulation similar to the Ricou-Spalding model:

$$\frac{dr_{dj}}{dt} = \mathrm{ENT0C} \cdot \left(\frac{\rho_w}{\rho_{dj}}\right)^{1/2} u_{dj}$$

where r_{dj} is the radius of the corium jet, ρ_w is the density of water, ρ_{dj} is the density of corium jet, and u_{dj} is the velocity of the corium jet. Model parameter ENT0C is a coefficient multiplier to the total mass of stripped particles (the higher the coefficient, the larger the fraction is).

MAAP 4.0.8 models water ingression by assuming the heat flux from a corium pool to its overlying water is prescribed by the critical heat flux, or higher, because water always ingress into the hot region in the corium. The heat flux formulation is given by:

$$q''_{cr-wt} = \mathrm{FCHF} \cdot \left[\frac{g\sigma(\rho_l - \rho_g)}{\rho_g{}^2}\right]^{1/4} \rho_g (h_g - h_l)$$

where g is the gravity constant, ρ_l and ρ_g are densities of saturated water and steam, and h_l and h_g are enthalpies of saturated water and steam. Modeling parameter FCHF is the Kutateladze number for corium to water heat transfer, which controls the magnitude of the heat flux.

The results of the CORQUENCH 3.03 calculations were used to tune the MAAP model parameters ENT0C and FCHF for the purposes of MCCI analysis. ENT0C was set to a low value in order to effectively disable 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 reaching the concrete at high temperature, which increases the calculated ablation depth. FCHF was tuned so that the reactor cavity ablation depth predicted by MAAP 4.0.8 is 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.

b. The scenario used to represent the Basemat Melt-Through fission product release category in Section 6.2.11 of the DCD provides a conservative, but reasonable representation of MCCI results for a case with a dry cavity. The sequence presented in section 6.2.11 demonstrates that fission product release will occur due to breach of the containment liner if no systems are available to mitigate MCCI.

c. This figure will be replaced with the correct figure. See Attachment.

Impact on DCD

The DCD Tier 2 Figure 19.2.3-7 will be revised as shown in Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 19.2.3-7 Ablation Depth in Floor and Sidewall for the PRA Sequence of Loss of Essential Service Water

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Question No. 19-68

10 CFR 52.47(a)(23) states that a design certification (DC) application for light-water reactor designs must contain an FSAR that includes 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.

APR1400 DCD Rev. 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. Revise the design control document (DCD) to describe the process used to determine vessel failure, modes of vessel failure, and failure size.

Response

The DCD will be revised. The markups and revised text are included in Attachment.

Impact on DCD

The DCD Tier 2 Section 19.2.3.2.1 will be revised as shown in Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

eutectics results in a non-uniform melting and relocation process. Further differences in the melt relocation process can be attributed to asymmetric bundle heating that can increase upon zirconium alloy oxidation. This process begins when one area of the fuel bundle is initially at a temperature higher than the other areas. The higher temperature zirconium alloy oxidizes at a faster rate. The exponentially increasing oxidation reaction contributes to asymmetric bundle heatup and the potential for incoherent melt relocation behavior.

As the temperature of the core increases, vaporization and release of some fission products occur. Steam or hydrogen carries these fission products throughout the primary system where they can deposit on internal components. The deposition mechanisms include condensation on the heat sinks (diffusiophoresis), gravitational settling, and thermophoresis. The fission products that are not deposited remain airborne and are released to the containment, where the dominant removal mechanisms are gravitational settling and diffusiophoresis.

Core melt progression, including relocation and fission product release, is a very complex process, which becomes increasingly difficult to predict as the scenario unfolds. The core melt could relocate into the lower reactor vessel plenum. If water is present in the lower plenum, the potential could exist for in-vessel steam explosions, wherein molten core material rapidly fragments and transfers its energy, causing rapid steam generation and possible shock waves.

Insert the text in the next page at this location.

The in-vessel core melt progression contains considerable uncertainty. This uncertainty relates 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

The unfragmented, intact molten corium jet reaching the vessel bottom will spread out and melt off the top of instrument penetration tubes that contact the molten corium. The molten corium can flow into a tube and thermally attack the tube wall outside the reactor vessel, causing a breach. Or, the molten corium can freeze and form a plug, sealing off the tube. Also, the molten corium can thermally attack the penetration tube weld before the protective oxide crust is formed, causing tube ejection.

If the vessel survives the initial attack when the molten core material is relocated to the lower plenum, a protective oxide crust will be formed on the lower head and the core debris will be quenched by the water in lower plenum. In time, the water will boil off and core debris will start to heat up. Eventually, the lower head will fail by creep rupture. If ex-vessel cooling is available by submerging the lower head in water, the core debris will continue to heat up and form a circulating molten pool in the lower plenum. In this case, the likely vessel failure mechanisms are either ablation of the vessel wall by the molten metal layer overlying the oxide corium pool or ejection of a penetration tube.

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 tube 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.

After the lower head fails by any of the localized failure mechanisms, the core debris above the failure location in lower plenum will be discharged. The remaining debris will continue to heat up and the code will continue to calculate the accumulation of creep damage. If the creep damage fraction exceeds 0.4, the lower head is defined to be extensively failed and the failure is defined to be at the bottom of the vessel with a radius of 0.1 m. The extensive failure is intended to discharge any debris that may have been retained in the lower head after the initial localized failure.

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Question No. 19-69

10 CFR 52.47(a)(23) states that a design control (DC) application for light-water reactor designs must contain an FSAR that includes 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.

APR1400 design control document (DCD) Rev. 0, Section 19.2.3.3.6.1, states that the APR1400 design mitigates the possibility of a thermally induced steam generator tube rupture by operator actuation of the required pilot-operated safety relief valves (POSRVs). Provide sufficient information in DCD to confirm that after receiving an indication of core uncovery, as stated in DCD Section 19.2.3.3.3.1.2, the operator will have sufficient time to actuate POSRVs for mitigating thermally induced steam generator tube rupture.

Response

The evaluation of depressurization capability considering thermally induced SGTR was not performed in the APR1400 design. Instead, the rapid depressurization analysis was performed according to the time of POSRV manual open. Table 1 through 3 summarizes the analysis results of rapid depressurization. As shown in tables, even the operator manually open 4 POSRVs at the one and half hours after the onset of core damage, the RCS pressure could be decreased rapidly. If the operator can manually open 4 POSRVs regardless of timing, the RCS pressure would be decreased below 250 psia within one hour. Therefore, it is expected in the APR1400 design that the operator can have sufficient time to actuate POSRVs for mitigating thermally induced SGTR.

Table 1 R1_TLOESW003 Results of Rapid Depressurization Analysis Based on Operator Interventions

TS

Table 2 R3_LOOP004 Results of Rapid Depressurization Analysis Based on Operator Interventions

TS

Table 3 R4_SBO002 Results of Rapid Depressurization Analysis Based on Operator Interventions

TS

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

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