## KHNPDCDRAIsPEm Resource

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| **From:** | Ciocco, Jeff |
| **Sent:** | Tuesday, March 08, 2016 9:43 AM |
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| **Subject:** | APR1400 Design Certification Application RAI 432-8555 (19 - Probabilistic Risk Assessment and Severe Accident Evaluation) |
| **Attachments:** | APR1400 DC RAI 432 SPRA 8377.pdf |

KHNP,

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form. Your licensing review schedule assumes technically correct and complete responses within 30 days of receipt of RAIs. However, KHNP requests, and we grant, the following RAI question response times. We may adjust the schedule accordingly.

19-54: 60 days
19-55: 60 days
19-56: 60 days
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19-58: 45 days
19-59: 45 days
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19-62: 60 days
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19-65: 30 days
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Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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# REQUEST FOR ADDITIONAL INFORMATION 432-8377

Issue Date: 03/08/2016
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section:

QUESTIONS

19-54

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

Section 19.1.4.2.1.2.1 of APR1400 design control document (DCD) Rev. 0 states the following:

> Detailed evaluation of phenomena that affect containment failure timing, fission product releases, or that may have an impact on downstream top events are treated through the use of decomposition event trees (DETs). The containment ultimate pressure capacity and severe accident phenomena analysis results are needed for quantification of the DETs. This CET/DET approach allows a relatively detailed treatment of the phenomena affecting containment performance while maintaining a relatively simple and easily understood CET.

APR1400 DCD Rev. 0 does not provide a description of DET analysis. Update the DCD providing a description of DETs.

19-55

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

APR1400 design control document (DCD) Rev. 0, Section 19.1.4.2.1.2.1, states that “The containment event trees [CETs] are shown in Figure 19.1-42 through Figure 19.1-46.” Of these, the DCD does not describe Figures 19.1-43 through -46 showing CETs for SGTR, ISLOCA, Containment Isolation Failure, and Containment Failure before Vessel Breach, respectively, nor does it describe how the top events of these CETs were evaluated. Update the DCD describing these CETs and how their top events were evaluated.

19-56

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

APR1400 design control document (DCD) Rev. 0, Tables 19.1-29 and -30, provide Summary of Source Term Evaluation and Source Term Category Frequencies and Contributions to large release frequency (LRF) for internal events. However, similar information is not provided in the

DCD for internal fire and internal flooding. Update the DCD providing similar tables for internal fire and internal flooding.

19-57

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

Section 19.1.5.2.2 of APR1400 design control document (DCD), Rev. 0, states that "[t]he internal fire risk evaluation is performed 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 explain how KHNP evaluated the systems and equipment that are important for Level 2 and affected by internal fire. Revise the DCD accordingly.

19-58

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

APR1400 DCD Section 19.1.4.2.1.1, Rev. 0, 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 kg/cm$^2$ (250 psia)."

The MAAP input file for Case-A01 in APR1400-K-P-NR-013601-P, Revision 0, shows that the large break LOCA break area was calculated for a double-ended guillotine break of a pipe with diameter "XDCL." However, the MAAP input file does not provide the value used for XDCL.

The large break LOCA (LBLOCA) stated in the design control document (DCD) appears to be a single-ended guillotine break, which is inconsistent with the double-ended guillotine break used for the LBLOCA MAAP analysis provided in APR1400-K-P-NR-013601-P.

Explain the apparent discrepancy between the DCD and APR1400-K-P-NR-013601-P and update the DCD clarifying the type of break assumed.

19-59

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.

The source term evaluation results listed in APR1400 DCD Rev. 0, Table 19.1-29, show that the cesium iodide release fraction for source term category (STC)-21 is 357 times higher than that for STC-17 (5.0 versus 0.014 percent of total core inventory). However, MAAP calculations documented in APR1400-K-P-NR-013603-P show that STC-21 has only a 10 times larger

release opening area than STC-17 (1.0 $ft^2$ versus 0.1 $ft^2$). Explain the significant variation in releases in two cases compared to the area assumed.

19-60

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

The process of PRA quantification for Level 2 event trees for internal fire events is not stated in the design control document (DCD). Update the DCD providing Level 2 methodology and quantification for internal fire events.

19-61

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design-specific PRA and its results.

Section 19.1.4.2.1.2.1 of APR1400 design control document (DCD), Rev. 0, states the following: “Containment event trees (CETs) are developed to model the containment response during severe accident progressions. These CETs depict the various phenomenological progress, containment conditions, and containment failure modes that could occur under severe accident conditions.”

Section 19.1.4.2.1.2.3 of APR1400 DCD Rev. 0 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 overpressurization was considered using the results of the containment ultimate capacity evaluation. Many other calculations were performed to support the CET.

However, APR1400 DCD Rev. 0 does not provide information on MAAP runs performed and how results of MAAP runs were used to support the CET phenomenological evaluations. The staff need this information to understand how containment response during severe accident progressions was addressed for the APR1400 design. Provide details of MAAP runs performed to support the APR1400 CET phenomenological evaluations. Revise the DCD as necessary.

19-62

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 design control document (DCD) Rev. 0, Section 19.2.3.3.5.1.1, states that in-vessel steam explosion analysis is performed to confirm the applicability of the NRC Fuel-Coolant Interactions (FCI) expert review group OECD/NEA FCI specialist conclusions to the APR1400 design. The applicant provided this in-vessel steam explosion analysis in APR1400-E-P-NR-14003-P, “Severe Accident Analysis Report.” Rev. 0, Appendix D, “Severe Accident Analysis Report for FCI.” Add text to the DCD to describe the in-vessel steam explosion analysis performed including key assumptions, methodology, and key results..

19-63

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.

Provide the following regarding the discussion on *in-vessel* steam explosion as provided in APR1400-E-P-NR-14003-P, “Severe Accident Analysis Report,” Rev. 0, Appendix D, “Severe Accident Analysis Report for FCI” and revise the design control document (DCD) to incorporate them:

a. Figure 3-1 shows one-dimensional nodalization of TEXAS-V for the in-vessel steam explosion in the APR1400 RPV. Explain and justify using one-dimensional analysis.

b. Section 3.4.1 states that “The 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.” Explain the reasons for a second deceleration and subsequent acceleration of the jet.

c. Provide the initial void fraction assumed for the melt jet.

d. Explosion energy generated depends on melt fraction and void fraction before triggering an explosion, which are functions of time after the initiation of premixing. Provide the timing and justify the time at which triggering was assumed.

19-64

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:

Provide the following regarding the discussion on *ex-vessel* steam explosion as provided in APR1400-E-P-NR-14003-P, "Severe Accident Analysis Report," Rev. 0, Appendix D, "Severe Accident Analysis Report for FCI" and revise the design certification document (DCD) as necessary

a. Figure 4-2 shows one dimensional nodalization of TEXAS-V for the in-vessel steam explosion in the APR1400 RPV. Explain and justify using one-dimensional analysis for the cavity which has a large cross sectional area.

b. TEXAS-V code being one dimensional, assumed diameter for the mixing region would significantly affect the premixing results as shown in Figures 4-3 and 4-4. As stated in Section 4.5.3, mixing has an area of 7 $m^2$, which is significantly larger than the cross-sectional area of the melt jet of 0.2 $m^2$. Justify using one-dimensional analysis.

c. Provide the initial void fraction of the melt jet.

d. Explosion energy generated depends on melt fraction and void fraction before triggering an explosion, which are functions of time after the initiation of premixing. Provide the timing and justify the time at which triggering was assumed.

e. Table 4-17 showing cavity structural analysis results lists number of cracks as "47,073 EA" and a maximum crack width of 0.027 in. with a remark of considerable concrete damage. However, Table 5-1 remarks that ex-vessel steam explosion has no threat to APR1400 design. Explain what is meant by EA in listing number of cracks and why a possible concrete damage with 47,073 cracks would not cause a threat to the APR1400 cavity design.

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.

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.

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.”

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.

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.

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