REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

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

Docket No. 52-046

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Question No. 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 control document (DCD) as necessary

a. Figure 4-2 shows one dimensional nodalization of TEXAS-V for the ex-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.

Response – (Rev. 1)

a. TEXAS-V code is a one-dimensional code and the user is expected to input the area of the node as a user-defined parameter, ARIY, which corresponds to the cross-sectional area of the cavity. This user parameter is used to specify the amount of coolant at given node and its cooling capacity, in consequently. ARIY plays an important role in determining the vapor fraction during the mixing phase as well as the numerical convergence.

   Instead of the actual cavity cross-sectional area (approximately 80 m^2), ARIY is set to give a maximum energetic load based on the energy index concept, i.e. when the ratio of the given melt's initial thermal energy and the coolant energy places in the optimal range the explosion pressure can be maximized. In other words, if user introduces the actual cavity floor area of APR1400 as ARIY, the excess of cooling capacity can produce the higher void fraction and eventually it can lead to the limited energetic load due to one-dimensional characteristics of TEXAS-V code. In contrary for the case with too small ARIY, the certain amount of the melt thermal energy may remain inside the melt and it can restrict the higher load.

   The influence of the large cross-sectional area of the cavity is eliminated in TEXAS-V study in this way from the conservatism standpoint.

b. As discussed in Response a., ARIY represents the node area not the mixing area. The editorial error will be revised as Attachment (“mixing” replaced with “node”).

c. For melt jet, the initial void fraction is set to be zero.

d. The steam explosion energetics depends largely upon the corium mass participated in the interaction. Therefore, it is assumed that the artificial trigger is provided by the corium jet contact at the bottom of the reactor cavity. The less conservative results will be obtained if the corium jet is triggered before or after the bottom contact of corium leading edge to the cavity floor.

e. The numbers of cracks described in Table 4-17 include all cracks having from a very small crack width to maximum 0.027 in crack width. In addition, there are no through cracks in concrete. It means that the possible concrete damage did not cause a threat to the cavity design even though cracks seem quantitatively much. In the scope of leakage, the damage of liner plate rather than concrete crack is more important. By ex-vessel steam explosion, the maximum stress in the liner plate is 54.9 ksi which is less than the ultimate tensile strength (75 ksi). In addition, the maximum effective plastic strain is around 1.1% which is less than the failure strain criteria of liner plate (5%). Therefore, it can be concluded that the APR1400 cavity structure remains intact from the ex-vessel steam explosion.

   As mentioned in APR1400-E-P-NR-14003-P/NP, “Severe Accident Analysis Report,” Rev. 0, Appendix D, “Severe Accident Analysis Report for FCI”, 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 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 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 clarification of the present APR1400 analysis results, the justification of application of the reference plant analysis results is reflected in Section 19.2.3.3.5.2.2 of APR1400 DCD, Tier 2.

Impact on DCD

DCD Tier 2, Section 19.2.3.3.5.2.2 will be revised, as indicated in the Attachment1 associated with this response.

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

Technical Report APR1400-E-P-NR-14003-NP and APR1400-E-P-NR-14003-P, “Severe Accident Analysis Report,” Rev. 0, Appendix D, “Severe Accident Analysis Report for FCI” Section 4.5.3 is revised as shown in Attachment 2.

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designed such that the cavity strength has an adequate capability to withstand the postulated pressure load during a severe accident.

For the assessment of reactor cavity structural integrity against the EVSE pressure loading, the concrete cracks of cavity walls and bottom slab, the stress of the RPV column support anchor bolts, reinforcement rebars, and liner plates were evaluated using LS-DYNA computer program. The results of evaluation confirm that the reactor cavity is capable of maintaining structural integrity when EVSE loads are applied.

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The requirements of ACI 349-97 (Reference 26) were used in determining the ultimate static pressure capacity and the dynamic pressure capacity of the reactor cavity wall (except no load factors were applied to the loads because of the highly unlikely occurrence of a severe accident and the one-time loading condition). As such, potential additional margins in reinforcing strength, concrete strength, and the material ductilities beyond those allowable by design code were not used in determining the aforementioned static and dynamic capacities of the structure. The evaluation of the cavity structural analysis indicates that the reactor cavity integrity is preserved during both static and dynamic EVSE loads.

19.2.3.3.6 <u>Containment Bypass</u>

Containment bypass events involve failure of the pressure boundary between the high-pressure reactor coolant system and a low-pressure auxiliary system. For PWRs, this can also occur because of the failure of the steam generator tubes, either as an initiating event or as a result of severe accident conditions.

These scenarios are important because, if core damage occurs, a direct path to the environment can exist. This can lead to an early release of fission products outside containment and public health risks. The following sections describe potential containment bypass events for the APR1400.

19.2.3.3.6.1 <u>Steam Generator Tube Rupture</u>

A thermally induced steam generator tube rupture (SGTR) can occur in severe accident sequences where the primary system is at high pressure during core melt. This condition

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The structural assessment of reactor cavity under EVSE loading was performed in the reference plant project. The results of reactor cavity structural assessment of reference plant are applicable to the APR1400 because design parameters such as geometry, material properties, rebar arrangement, and design codes are the same between the reference plant and the APR1400. In addition, the EVSE pressure time history curve obtained from the APR1400 is almost identical to that of the reference plant with small perturbation after peak pressure. It is noted that this difference is negligible because the dynamic structural response depends on the peak value and its time.

The analysis shows that the peak pressure and corresponding impulse of 60.35 MPa and 194.07 kPa-s, as shown in Table 4-12, are estimated. The results are similar to those from the base case. As described in Table 4-8, the initial conditions for the SVF case assume that the corium is 100% metallic composition with high superheat of corium but lower temperature. In addition, the corium injection velocity at the vessel breach location is low due to the small gravitational head of corium in the reactor vessel. Comparing to the base case, the peak pressure due to steam explosion is similar but the impulse generated by the steam explosion is higher. The steam explosion loadings to the cavity wall will be higher than that of the base case due to the location of the vessel failure.

4.5.2 SAMG Related Issues: In-Vessel Corium Melt Retention (IVR)

For the case of IVR/ERVC, the RPV is in a stage of submersion in the fully flooded cavity water up to EL114'-4" from the plant ground level, or 13.8 m from the plant cavity floor (see Figure 4-2), to provide the external cooling when the core meltdown and relocation to the bottom of the reactor vessel occurs. In this situation, there is two potential vessel failure modes; bottom and side vessel failures at the locations assumed to be 6.5 and 8.05 m, respectively.

Table 4-12 shows that the peak pressures and maximum impulses for both bottom and side vessel failures with IVR-ERVC are 69.79 MPa, 217.33 kPa-s and 48.84 MPa, 226.16 kPa-s. It is noted that for the bottom vessel failure in the case of fully-flooded (FF) case, the explosion peak pressure is slightly higher but the impulse becomes about 20% higher. For the side vessel failure, however, it was observed that the tendency of explosion pressure profile was opposite to one for the bottom vessel failure, resulting in about 20% lower peak pressure but 26% higher impulse. The result indicates that the energetics of the side vessel failure is slightly higher than one of the bottom vessel failure.

4.5.3 Effects of Key Physical Parameters on EVSE Energetics

In this sensitivity analysis, some of key parameters pertaining to the thermal and dynamic properties of corium and the conditions of cavity water are examined to investigate their uncertainties on the energetics of EVSE in the APR1400 design. In this sensitivity analysis, it is worth to note that the ~~mixing~~ area defined by the model parameter, ARYI value of 7 m^2, is maintained in most of cases (except corium jet diameter effects).

node

4.5.3.1 Corium Temperature Effects

The effect of the initial corium temperatures on the EVSE energetics with the minimum and the maximum temperatures of 2900 and 3150 K is analyzed as shown in Table 4-13. Those temperatures correspond to the corium superheats of 50 and 300 K respectively. The results show that the energetics of EVSE in terms of pressure impulse increases with the corium temperature; 168.27 and 216.69 kPa-s for 50 K and 300 K superheat of corium, respectively. However, it also shows that the peak pressures for three cases; minimum, base, and maximum, are in a similar range of approximately 57-67 MPa. It indicates that the increase of thermal contents of corium enhances the explosion pressure peaks and profiles.

4.5.3.2 Corium Ejection Velocity Effects

The corium ejection velocity influences directly to the mixing phase of steam explosion process, mainly to corium jet breakup. In general, jet breakup length depends on the Froude number, and the ratio of density ratios between jet and coolant as shown in Eq. (4-2) below, showing the linear increase of the jet breakup length with the jet velocity,

$$\frac{\mathrm{L}}{D_j} \propto \left(\frac{\rho_j}{\rho_c}\right)^{0.5} (Fr)^{0.5} \qquad \text{(Eq. 4.2)}$$

where,