REVISED 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. 19-64

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. 3)

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.

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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. Based on this selected ARIY of 7.0 m^2, energy index was calculated to be [6.5 %], as below.

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

In addition, regarding In-Vessel Retention and External Reactor Vessel Cooling (IVR-ERVC) strategy, we did not give credit to the IVR-ERVC system. The adoption of this strategy is related to Accident Management (AM). Severe Accident Management Guideline (SAMG) continent to activation of IVR-ERVC is also constructed in the AM procedure. Therefore the evaluation of the steam explosion load and consequential structural integrity assessment under the IVR-ERVC situation will be performed as a COL item. To clarify this concern, DCD Section 19.2.7 is revised as shown in Attachment 5.

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 Attachment 1 associated with this response and Section 19.2.3.3.3.3, 19.2.3.3.4.3, 19.2.3.3.5.2.1, 19.2.3.3.5.2.2 as shown in the Attachment 4.

Section 19.2.7 will be revised as shown in Attachment 5.

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 and Section 4.4.2.1 is revised as shown in Attachment 3.

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

The 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,

Figures 4.7(b), (c) and (d) show more insights of the explosion phase of the steam explosions in RPV that include the energy partitioning, the coolant kinetic energy and the conversion ratio history during the explosion phase. In particular, the conversion ratio of the steam explosion reaches its maximum of near 1.8% after the triggering of explosion.

4.4.2.1. Evaluation of Dynamic Loads of EVSE at the Cavity Walls

Shock pressure generated from the steam explosion in the reactor cavity pool propagates. In the TEXAS code, however, due to its one-dimensionality of computational domain, the pressure generated at one location, x(z), can be tractable only in the vertical z-direction. Therefore, the impulse acts to the cavity wall in the radial direction requires additional analysis. The most recent version of the TEXAS-V code encompassed with the ANSYS CFD packages to analyze the radial shock propagation. On the other hand, the underwater shock propagation studied by Cole [Reference 46] known as a TNT method has been well applied for this purpose.

Figure 4-9 illustrates the reactor cavity arrangement in the APR1400 plant. The distances from the center axis of the RPV centerline to the near cavity walls are listed in Table 4-11. It is noted that the closest wall from the center has a distance of 2.159 m.

If the maximum explosion pressure at a known distance, for instance, $r=R_{mix}$, is ΔP_{mix}, the distance-dependent maximum explosion pressure, $\Delta P_m(r)$ becomes,

$$\Delta \mathrm{P_m} = \Delta \mathrm{P_{max}} \left(\frac{1}{r}\right)^{\alpha} \quad \text{(Eq. 4.1)}$$

where, α=1.13 and all units are the British units, ie., P [psia] and r [ft]. For instance, in the TEXAS-V analysis, it is difficult to evaluate the exact mixing zone for the steam explosion although the one-dimensional characteristic parameter, ARIY, was at to 7 m^2 that is the diameter of approximately 3 m. By assuming this diameter be the mixing zone and considering the distance from the outer mixing boundary to the near cavity wall, the maximum pressure propagation ~~along the lateral direction to the cavity wall can be estimated by Eq. (4.1) using ΔP_{max}=60.51 MPa. Table 4-11 shows the estimated maximum pressures at the cavity walls that significantly attenuated from the EVSE maximum pressure. These estimated values can be used for the structure analysis of cavity integrity due to the EVSE loadings.~~

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4.5. Sensitivity Study

For the sensitivity study, additional cases for issues associated with (a) vessel failure modes such as bottom failure due to penetration tube failure, and side vessel failure due to metallic layer focusing effect, (b) severe accident management strategies, and (c) key corium characteristics including the corium temperature, the velocity and diameter and the cavity water temperature are examined. Tables 4-12 to 16 show the result of the analyses in comparison to the base case. The details are discussed in the following sub-sections.

4.5.1. Reactor Vessel Failure Mode Issues

For the side vessel failure, the vessel failure location and break size are important parameters that determine the energetics of EVSE because it determines the mass of corium participated during EVSE and the distance between the mixing zone of steam explosion and the nearest cavity wall. In the case of a potential vessel failure due to the metallic corium layer focusing effect with assumption of the side vessel failure without IVR-ERVC (In-Vessel Core Melt Retention-External Reactor Vessel Cooling) SAM strategy, RPV is exposed to atmosphere and the location of side vessel failure occurs at 8.05 m above the cavity floor (~ 80^{O}) as shown in Figure 4-2(A).

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However, regarding the uncertainty of radial mixing zone length and, consequently, the radial exact distance from the explosion point to the cavity walls, we impose the peak pressure without attenuation on the near cavity wall regions. In other words it is assumed that the EVSE occurs at the lower cavity wall surface, as marked `explosion location' in Figure 4-18. Moreover the explosive load given by TEXAS-V calculation without attenuation is applied for the lower cavity wall segment `A' in the same figure. The input load on the rest of cavity wall segments is determined from the Eq. (4.1) according to the distance from the explosion location. The input load used for the cavity wall integrity calculation is given in Figure 4-17.

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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.

19.2.3.3.3.3 Analysis Result

(Reference 15)

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.

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 determined. The ejection characteristics of core debris are determined based upon the geometrical configuration of the containment. Probabilistic distribution functions for uncertainties in parameters such as core debris mass, degree of Zr oxidation, coherence ratio describing heat transfer between dispersed debris and gases in containment, and containment failure pressure are determined. A TCE analysis is performed by sampling inputs using 10,000 samples by LHS processing coupled with all generated data.

19.2.3.3.4.3 Analysis Result

Figure 19.2.3-14 shows the RCS pressure responses during the rapid depressurization. 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 (17.6 kg/cm^2 [250 psi]) for all sequences considered. Table 19.2.3-3 shows the summary of results for the rapid depressurization analysis about the Total Loss of Essential Service Water (TLOESW) sequence (Reference 15). The analysis results comply with SECY-93-087 (Reference 1).

For each of the three scenarios, no containment failure cases have resulted in 10,000 trials (Reference 15). Based on this outcome, the CFP in APR1400 due to DCH is estimated to be less than 0.01 percent (0.0001). This indicates that APR1400 meets the success criterion established in NUREG/CR-6338 (Reference 18) for PWR large dry containment, where DCH problem is considered resolved if CFP is less than 1 percent (0.01).

19.2.3.3.5 Fuel-Coolant Interactions

The containment integrity and function may be challenged by dynamic loads from a steam explosion resulting from FCI. For the evaluation of the risks associated with FCIs for the APR1400 design, in-vessel steam explosions (IVSEs) and ex-vessel steam explosions (EVSEs) are described and analyzed in accordance with 10 CFR 52.47(a)(23) (Reference 20).

19.2.3.3.5.2 Analysis Result

19.2.3.3.5.2.1 In-Vessel Steam Explosion

The key physical processes that can influence in-vessel steam explosions for PWRs are (a) melt relocation into the lower plenum, (b) corium jet breakup and coarse mixing formation in the lower plenum, (c) triggering of coarse mixing, (d) energetic FCIs, and (e) pressure loads to the upper and lower vessel heads and their responses.

Both NUREG-1116 and NUREG-1524, written by the NRC-sponsored Steam Explosion Review Group, concluded that the potential for alpha-mode failure is vanishingly small or physically unreasonable. The OECD/Committee on the Safety of Nuclear Installations (CSNI) also confirmed the conclusion of NUREG-1524 and concluded that the alpha-mode failure issue was resolved from a risk perspective.

Because the APR1400 design is not significantly different from current PWRs, the NUREG-1524 conclusions are applicable to the APR1400 design, thus no mitigation features are provided to prevent or mitigate IVSE. (Reference 15)

19.2.3.3.5.2.2 Ex-Vessel Steam Explosion

The initial and boundary conditions for EVSE are largely dependent upon the in-vessel severe accident progression, severe accident management procedure, and vessel failure modes. Thirteen severe accident sequences were chosen to cover the spectrum of key variable parameters and thus characterize the initial and boundary conditions for EVSE analysis. The key parameters considered include corium discharge rates, corium thermal conditions, cavity conditions, and related parameters.

The result of analysis using the MAAP code provided the initial conditions for the TEXAS-V code. TEXAS-V was then used to calculate the peak pressure due to EVSE. (Reference 15) The pressure at the nearest cavity wall was then estimated by the TNT method (Reference 25).

The reactor cavity and RPV column support have to maintain structural integrity in events such as an ex-vessel steam explosion. 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.

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.

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 (Reference 15).

19.2.3.3.6 Containment Bypass

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 Steam Generator Tube Rupture

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

19.2.6.7 Conclusions

The analyses described in the previous sections analyzed conceptual alternatives for mitigating severe accident impacts in the APR1400 design. Preliminary screening eliminated all SAMDA candidates from further consideration, based on inapplicability to the design, design features that have already been incorporated into the design, inapplicability to a design certification stage, or extremely high cost of the alternatives considered.

The analysis using a 7% discount rate showed that no design changes to reduce risk associated with contributors to plant risk would be cost-beneficial to implement. A second baseline maximum benefit calculation using a 3% discount rate showed only minor variations in the calculated benefits. Therefore, it is concluded that no design changes would provide a positive cost-benefit if included in the APR1400 design.

19.2.7 Combined License Information

COL 19.2(1) The COL applicant is to perform and submit site-specific equipment survivability assessment in accordance with 10 CFR 50.34(f) and 10 CFR 50.44.

COL 19.2(2) ~~The COL applicant is to develop and submit an accident management plan.~~

The COL applicant is to develop and submit an accident management plan including the evaluation of the effect of higher water level in the cavity on steam explosion loading when using In-Vessel Retention and External Reactor Vessel Cooling for accident management.

19.2.8 References

1. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” U.S. Nuclear Regulatory Commission, April 1993.

2. 10 CFR Part 100, “Reactor Site Criteria,” U.S. Nuclear Regulatory Commission.

3. 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” Title 10, Code of Federal Regulations, ,U.S. Nuclear Regulatory Commission.

4. 10 CFR 50.34, “Contents of Applications; Technical Information,” U.S. Nuclear Regulatory Commission.