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.:	**208-8245**
SRP Section:	**03.08.03 – Concrete and Steel Internal Structures of Steel or Concrete Containments**
Application Section:	**03.08.03**
Date of RAI Issue:	**09/14/2015**

Question No. 03.08.03-4

10 CFR 50.55a and Appendix A to 10 CFR Part 50, General Design Criteria 1, 2, 4, 16 and 50, provide the regulatory requirements for the design of the containment internal structures. Standard Review Plan (SRP) 3.8.3, Section II specifies analysis and design procedures normally applicable to internal concrete structures, with emphasis on the extent of compliance with American Concrete Institute (ACI) 349-01, “Code Requirements for Nuclear Safety Related Concrete Structures,” with additional guidance provided in Regulatory Guide 1.142, and ANSI/AISC N690-1994, “Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities,” including Supplement 2.

APR1400 DCD Tier 2, Section 3.8.3.4, “Design and Analysis Procedures,” states, “The thermal stress analysis is carried out by inputting the normal operating thermal load into the corresponding FEM for the internal structure.” In reviewing this section, the staff noted that no description about accidental thermal loads - loads generated by a postulated pipe break - was provided. Per 10 CFR 50.55a; Appendix A to 10 CFR Part 50, General Design Criteria1, 2, 4, 16 and 50; and SRP 3.8.3, the applicant is requested to confirm that accident thermal loads were considered; and describe how the accident thermal loads were evaluated in the analysis and design of the internal structures.

Response – (Rev. 1)

Equivalent linear temperature profile of normal operating conditions represent the limiting temperature profile for all plant conditions. Therefore, normal operating thermal loads were considered for containment internal structure design instead of accident thermal loads. Figure 1, below, compares normal operating thermal loads and accident thermal loads.

According to ACI 349, the actual non-linear temperature distribution can be converted to an equivalent linear temperature distribution for use in design of concrete structures. In the containment internal structure, the equivalent linear temperature profile for normal operating conditions is more severe than those of the accident conditions since the temperature difference between the inside and the outside surface of the containment internal structure during accident conditions is negligibly small. Figure 1 for the PSW shows the example of differential temperatures for normal operating conditions and accident conditions. The temperature for accident condition is the temperature at 1,000 sec which is the worst case among the accident conditions. For the worst case of temperature analysis of internal structure, normal operating loads are considered to the wall located close to heat source is selected.

TS

Figure 1. Example of Temperature Profile

Figure 2. Primary Shield Wall Temperature Profile – Equivalent Linear Curve

TS

Figure 3. Primary Shield Wall - Concrete Temperature Profile During Accident

Table 1. Primary Shield Wall - Concrete Temperature Profile During Accident

TS

TS

Impact on DCD

DCD Tier 2, Section 3.8.3.4 will be revised, as indicated in the attached markup.

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 Environmental Report.

RAI 208-8245 - Question 03.08.03-4_Rev.1

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Analysis Procedure

The internal structure is designed for the loads and loading combinations specified in Subsection 3.8.3.3. The internal reinforced concrete structure including the reactor coolant system (RCS) is modeled with eight-node solid elements, three-node or four-node shell elements, and two-node beam elements using the ANSYS computer program. The design loads for analysis of internal structures are classified dead load, live load, hydrostatic and dynamic loads, temperature load, accident pressure load, pipe break load, and seismic load.

Dead loads include the self-weight of the PSW, SSW, IRWST, and fill concrete; equipment; and intermediate steel floor framing. Large equipment loads (e.g., reactor drain tank, letdown/regenerative heat exchanger, safety injection tank, recirculation fan) are treated as dead loads. In addition, potential loads during construction or operating periods are treated as live loads.

The vertical and lateral pressures of liquids inside containment are treated as dead loads. Structures supporting fluid loads during normal operation and accident conditions are designed for the hydrostatic as well as hydrodynamic loads.

The thermal stress analysis is carried out by inputting the normal operating thermal load into the corresponding FEM of internal structure. During the thermal analysis, the equivalent uniform temperature gradient is input directly in the ANSYS model at the appropriate nodes.

The compartment pressures on internal structures are the result of a pipe break inside containment. In addition, branch line pipe break (BLPB) loads are dynamic reactions caused by the combined effects of branch line nozzle reactions or thrust due to pipe break, jet impingement on RCS equipment, or subcompartment pressure effects on RCS equipment. These loads are applied to the ANSYS model with pressure and concentrated loads.

In the containment internal structure, the equivalent linear temperature profile for normal operating conditions is more severe than those of the accident conditions since the temperature difference between the inside and the outside surface of the containment internal structure during accident conditions is negligibly small. Therefore, the normal operating thermal load is conservatively applied for design of internal structures in case of normal operating and accident conditions.

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.:	**208-8245**
SRP Section:	**03.08.03 – Concrete and Steel Internal Structures of Steel or Concrete Containments**
Application Section:	**03.08.03**
Date of RAI Issue:	**09/14/2015**

Question No. 03.08.03-6

10 CFR 50.55a and Appendix A to 10 CFR Part 50, General Design Criteria 1, 2, 4, 16 and 50, provide the regulatory requirements for the design of the containment internal structures. Standard Review Plan (SRP) 3.8.3, Section II specifies analysis and design procedures normally applicable to internal concrete structures, with emphasis on the extent of compliance with American Concrete Institute (ACI) 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures," with additional guidance provided in Regulatory Guide 1.142.

APR1400 DCD Tier 2, Section 3.8A.1.4.3.1.3, "Analysis Methods and Results," describes analysis parameters used for the in-structure refueling water storage tank (IRWST) in the containment internal structure FEM. It indicates that the damping ratio for water in the IRWST or refueling pool is the same as that for reinforced concrete structures: the seismic response of water is only considered as impulsive (rigid) mode for structural analysis. Per 10 CFR 50.55a; Appendix A to 10 CFR Part 50, General Design Criteria 1, 2, 4, 16 and 50; and SRP 3.8.3, the applicant is requested to explain why the damping ratio for water is included in the model unless the water is included using finite elements to represent the water. If so, then describe the methodology for representing the water as finite elements.

Based on the above statement from DCD Section 3.8A.1.4.3.1.3, explain why only the impulsive (rigid) mode is considered in the analysis. To enable the staff to fully understand how water in pools are evaluated to design the pool walls and slabs, the applicant is also requested to provide a full description of how water in the various pools is modeled in the FEM and how member forces are determined to design the walls and floors of the pools. Also, explain if the approach followed the methodology presented in ASCE 4-98 Section 3.5.4, or what alternative methods were used, and the basis for those methods.

Response – (Rev. 1)

1. In the IRWST analysis, the water is not included using finite elements to represent the water. The water is only considered as mass, and the damping ratio for the IRWST is equal to the value used for reinforced concrete structures. DCD Section 3.8A.1.4.3.1.3 will be revised as shown in the attached markup to clarify the value used for the damping ratio.

2. The hydrodynamic pressure in the IRWST which results from seismic excitation can be considered as impulsive and convective modes depending on the depth, simultaneously, but not in phase with each other (Refer to ACI 350.3-06). The impulsive pressure is associated with inertial force produced by acceleration of the wall, and the convective pressure is produced by the oscillations of the fluid. The impulsive mode primarily acts to stress the wall, whereas the convective mode acts primarily to uplift the wall. The sloshing due to the convective mode could increases and decreases the fluid pressure on the wall and the fluid pressure due to the sloshing effect is smaller than that due to the impulsive effect. Therefore, considering the impulsive mode over the water level is more conservative than considering both impulsive and convective modes. Here, with regard to the impulsive mode over the water level, the 50% mass of total water is applied to the inner wall and outer wall, respectively, in the horizontal (radial and tangential) direction and the 100% mass of total water is applied to the bottom slab in the vertical direction.

3. There are two water tanks in the RCB, the IRWST and the refueling pool. Water is not in both tanks simultaneously. The hydrostatic pressure loads are applied as surface pressure on the IRWST and refueling pool walls and bottom slabs. For the hydrodynamic pressure, a mass calculated in accordance with ASCE 4-98, Section 3.5.4 is applied to the IRWST and refueling pool, and then a response spectrum analysis was performed. Member forces and stresses for design were computed using modal combinations (SRSS) in accordance with Regulatory Guide 1.92.

4. In the IRWST analysis, the methodology presented in ASCE 4-98, Section 3.5.4 was used.

Impact on DCD

DCD Section 3.8A.1.4.3.1.3 will be revised as indicated on the attached markup.

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 Environmental Report.

c. SG compartment – SG blowdown nozzle

d. PZR compartment – PZR spray nozzle

e. PZR compartment – POSRV nozzle

f. PZR spray valve room – PZR spray line

Branch line pipe break (BLPB) loads are dynamic reactions caused by the combined effects of branch line nozzle reactions or thrust due to pipe break, jet impingement on RCS equipment, or subcompartment pressure effects on RCS equipment. The RCS support reactions due to BLPB are applied as nodal forces at the support locations.

The hydrodynamic pressure load, which is generated by the expulsion of air in the pilot-operated safety relief valve (POSRV) discharge, is applied to the wall and bottom slab of the IRWST through the two spargers. For the hydrodynamic pressure load, by multiplying the dynamic impact factor (DIF), the maximum pressure is conservatively considered as the static load in the analysis. In addition, the normalized factor is considered for the spatial distribution due to the location of spargers.

The seismic analysis for structures is performed using response spectrum analysis. A 7 percent damping ratio for reinforced concrete structures (SSE) and 3 percent damping ratio for the RCS model are used. In addition, the damping ratio for ~~water in~~ the IRWST or refueling pool is the same as that for reinforced concrete structures: the seismic response of water is only considered as impulsive (rigid) mode for structural analysis. Figure 3.8A-5 (a) and (c) show the in-structure response spectrum (ISRS) of the SSE level at El. 78 ft 0 in with 3 percent and 7 percent damping.

Three sections are selected in the PSW as critical sections. Each section is thinnest in the directions of north, south, and east. The design forces and moments for PSW critical sections are presented in the Table 3.8A-18. Table 3.8A-22 presents the margins of safety of rebar stress in the primary shield wall. The margin of safety is the ratio of allowable stress and actual stress.

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.:	**208-8245**
SRP Section:	**03.08.03 - Concrete and Steel Internal Structures of Steel or Concrete Containment**
Application Section:	
Date of RAI Issue:	**09/14/2015**

Question No. 03.08.03-7

10 CFR 50.55a and Appendix A to 10 CFR Part 50, General Design Criteria 1, 2, 4, 16 and 50, provide the regulatory requirements for the design of the containment internal structures. Standard Review Plan (SRP) 3.8.3, Section II specifies analysis and design procedures normally applicable to internal concrete structures, with emphasis on the extent of compliance with American Concrete Institute (ACI) 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures," with additional guidance provided in Regulatory Guide 1.142.

APR1400 DCD Tier 2, Section 3.8.3.1.11, "Interior Concrete Fill Slab," and Appendix 3.8A.1.4.3, "Internal Structures," indicate that the containment internal structures include concrete fill located on the surface of the liner plate of the reactor containment building basemat for protection of the pressure boundary structures. The staff notes that this concrete fill also provides support to the containment internal structures. Therefore, per 10 CFR 50.55a; Appendix A to 10 CFR Part 50, General Design Criteria 1, 2, 4, 16 and 50; and SRP 3.8.3, the staff requests the applicant to explain if the concrete fill is reinforced concrete. If not, then explain how the structural adequacy of the concrete fill is demonstrated. In addition, describe the connection details of the concrete fill to the reactor containment basemat to demonstrate its capability to withstand the various loads including seismic.

Response – (Rev. 1)

The detail of the concrete fill slab in CIS is shown in the following figure. As shown in the figure, the concrete fill slab is to protect the liner plate on the basemat and it is not required as a structural member. Since the reinforcements, which are calculated for all applicable loads and load combinations of PSW and SSW, are developed in the basemat, the load of PSW and SSW can be transferred to the basemat without the concrete fill slab. The reinforcement of the concrete fill slab is calculated based on the maximum moment (around the circumference) of the SSW at the elevation of top of concrete fill. This maximum moment is to calculate the reinforcement of the concrete fill using the height of the concrete fill as the depth of the section.

Therefore, the concrete fill slab is designed for more conservative load than the actual load on it. Because the vertical g-value of fill concrete slab is less than 0.5, there is no uplift force.

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

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