KHNPDCDRAIsPEm Resource

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Sent: Monday, September 14, 2015 7:58 AM
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Subject: APR1400 Design Certification Application RAI 208-8245 (03.08.03 - Concrete and Steel Internal Structures of Steel or Concrete Containments)
Attachments: APR1400 DC RAI 208 SEB1 8245.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.

Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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REQUEST FOR ADDITIONAL INFORMATION 208-8245

Issue Date: 09/14/2015
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 03.08.03 - Concrete and Steel Internal Structures of Steel or Concrete Containments
Application Section: 3.8.3

QUESTIONS

03.08.03-1

10 CFR 50.55a and Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 1, 2, 4, 16 and 50, provide the regulatory requirements for the design of structures including the in-containment refueling water storage tank (IRWST). Standard Review Plan (SRP) 3.8.3, Section II discusses the loads and load combinations normally applicable to internal concrete structures, particularly the IRWST, 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.

In DCD Tier 2, Section 3.8.3.1.8, "In-Containment Refueling Water Storage Tank," the applicant stated that, "The design of the IRWST considers pressurization as a result of the reactor containment building systems design basis accident." The applicant further stated in Appendix 3.8A, Section 3.8A.1.4.3.1.3, "Structural Design Summary," report that, "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." The staff noted that additional information is needed in order to better understand the hydrodynamic pressure loads that are being considered for the analysis and design of the IRWST. Per 10 CFR 50.55a, Appendix A to 10 CFR Part 50, GDCs 1, 2, 4, 16 and 50; and SRP 3.8.3, the applicant is requested to provide additional information that fully describes the hydrodynamic pressure loads that are being considered in the analysis and design of the IRWST. The additional information should include a description, as to:

a. How the pressure transient (including the steady state portion) for a single sparger activation was developed (a figure of this load over time should be provided).
b. What cases of activation of the spargers may occur, i.e., one sparger alone, two spargers simultaneously, and/or some lag between the two activations.
c. How the total pressure transients on the walls and floor were developed based on the cases identified in item (b) above (a figure showing the pressure distributions on the walls and floor should be provided).
d. How the dynamic impact factor, discussed in DCD Section 3.8A.1.4.3.1.3, was determined, and explain why applying this as a static load is considered conservative.

03.08.03-2

10 CFR Part 50.65 provides the regulatory requirements for monitoring the effectiveness of maintenance of the containment internal structures (CIS). Standard Review Plan (SRP) 3.8.3, Section II.7, provides guidance for testing and examining the preservice, in-service, and repair/replacement requirements of containment internal

structures. DCD Tier 2, Section 3.8.3.7, "Testing and Inservice Inspection Requirements," refers to DCD Section 3.8.4.7, "Testing and Inservice Inspection Requirements," which describes the testing and in-service requirements for other seismic Category I structures.

The staff reviewed DCD Section 3.8.4.7 and noted that the applicant did not identify and discuss the requirements for monitory the effectiveness of maintenance of the CIS and other structures, specifically, the requirements of 10 CFR 50.65 and the supplemental guidance of Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance and Nuclear Power Plants." Per 10 CFR 50.65 and SRP 3.8.3, the staff is requesting the applicant to include this information in the DCD.

03.08.03-3

Appendix A to 10 CFR Part 50, General Design Criteria 1, 2, 4, 16 and 50, provides the regulatory requirements for the design of the containment internal structures. Standard Review Plan (SRP) 3.8.3, Section II discusses the loads and load combinations normally applicable to containment internal 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 3.8.3.3, "Loads and Load Combinations," indicates that the typical loads and load combinations used for the internal structures are detailed in Section 3.8.4.3. Then it lists loads that the internal structures are designed for. A comparison of these loads listed in DCD Section 3.8.3.3 with those of DCD Section 3.8.4.3 shows that some loads are not included like operating pressure loads, construction loads, and internal flooding. Per 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 confirm that all applicable loads described in DCD Section 3.8.4.3 are used for internal structures, or explain why not. This issue of consideration of all applicable loads also applies to the list of loads identified in DCD Section 3.8.3.4.1 – Analysis Procedures.

In addition, DCD Appendix 3.8A.1.4.3.1.2, "Load Combinations Considered," identifies load combinations that are critical for the analysis and design of the primary shield wall (PSW). It is not clear to the staff whether only these load combinations were evaluated or all load combinations were evaluated, and only these were critical. This should be explained. In addition, there are some loads that are not included in these critical load combinations such as R_o in load combination labelled a. and c., R_a in load combination b., and R_a, Y_j, Y_m, and Y_f in load combination d. This should be explained. The applicant is also requested to address the above items for the other containment internal structures (e.g., IRWST and SSW).

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.

03.08.03-5

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.

DCD Tier 2, Section 3.8A.1.4.3.1.3, "Analysis Methods and Results"

A. APR1400 DCD Tier 2, Section 3.8A.1.4.3.1.3 describes the analysis methods and results for the containment internal concrete structures. It states that, "Operating concrete floor slabs are modeled to mass in a finite element model (FEM), such as slabs between the SSWs and containment shell." 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 clarify if this meant to say that the operating floor slabs between the secondary shield walls (SSWs) and the containment shell are included as masses in the FEM. If this is the case, then explain why it is acceptable to decouple these slabs from the overall FEM analysis of the internal structures and how is the analysis and design for such subelements performed for all of the various loadings.

B. Additionally, DCD Section 3.8A.1.4.3.1.3 indicates that fifty percent of the weights and equipment weights on the floor between the containment shell and the SSW are assumed to be distributed to the containment shell and the SSW, respectively. This implies that there is a connection between the containment internal floors and the containment. 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 in what directions (radial, tangential, and/or vertical) are the connections made and the details of how they are designed. Also, identify the gap provided between the containment and the floor slabs/connections to prevent impact/interaction and describe how the relative displacements between the containment and the floor slabs/connections from all loads including thermal and seismic were determined to demonstrate the gap is adequate.

C. This DCD Section also indicates that Figure 3.8A-23 shows the full FEM for the containment internal structures and Figure 3.8A-24 shows the solid element model (PSW, IRWST, and fill concrete), shell element model (SSW), and beam element model (RCS). The staff notes that part (b) of Figure 3.8A-24, which is labelled Shell Element Model (SSW), does not show the shell elements of the SSW. The applicant is requested to clarify why not.

D. DCD Section 3.8A.1.4.3.1.3 states that, “An equivalent uniform temperature gradient is input directly in the ANSYS model at the appropriate nodes. The temperature profiles during normal operating condition are more severe than those of the accident condition, thus represent the limiting temperature for all the plant conditions.” 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 provide the technical basis for this conclusion.

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.

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.

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