KHNPDCDRAIsPEm Resource

From: Ciocco, Jeff
Sent: Thursday, December 17, 2015 11:28 AM
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Cc: Drzewiecki, Timothy; McKirgan, John; Steckel, James; Lee, Samuel
Subject: APR1400 Design Certification Application RAI 339-8415 (15.01.05 - Steam System Piping Failures Inside and Outside of Containment (PWR))
Attachments: APR1400 DC RAI 339 SRSB 8415.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 response times to the RAI questions.

15.01.05-1: 45 days
15.01.05-2: 45 days
15.01.05-3: 45 days
15.01.05-4: 60 days
15.01.05-5: 45 days
15.01.05-6: 60 days

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

Thank you,

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REQUEST FOR ADDITIONAL INFORMATION 339-8415

Issue Date: 12/17/2015
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 15.01.05 - Steam System Piping Failures Inside and Outside of Containment (PWR)
Application Section:

QUESTIONS

15.01.05-1

General Design Criteria (GDC) 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated accidents cannot sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. Additionally, the standard review plan (SRP), NUREG-0800, requires the evaluation model for postulated accidents be suitably conservative.

Sections 2.5.1 and 3.1.2.1 of APR1400-Z-A-NR-14006, “Non-LOCA Safety Analysis Methodology,” state that CESEC-III contains a detailed thermal-hydraulic model that explicitly simulates the mixing in the reactor vessel from asymmetric transients. Figure 3.1-3 of Technical Report APR1400-Z-A-NR-14006-P shows the mixing parameters can have a significant impact on the reactivity insertion during the steam line break event. Technical Report APR1400-Z-A-NR-14006-P states that the mixing parameters are experimentally determined. NRC staff is questioning if the experimentally determined values remain applicable to the steam line break scenario. NRC staff requests the following:

1. Explain the process for determining the experimentally obtained values for the mixing parameters.

2. Explain how the values used in the steam-line break analysis provide a suitably conservative estimate of mixing in the reactor vessel.

15.01.05-2

General Design Criteria (GDC) 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated accidents cannot sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. Additionally, the standard review plan (SRP), NUREG-0800, requires the evaluation model for postulated accidents be suitably conservative.

The safety injection system is credited during the steam line break event to inject borated water which provides negative reactivity to ensure that the core remains subcritical. However, significant information regarding the modeling of the safety injection system in CESEC-III is not contained within the design control document (DCD) or supporting technical reports. NRC staff requests the following information:

1. Describe how the safety injection flow vs pressure is modeled in CESEC-III. If the modeling is not consistent with DCD Table 6.3.2-4, then additional justification is required to demonstrate how the modeling is suitably conservative.

2. Describe how the boron reactivity vs concentration is determined for use in the CESEC-III analysis of the steam line break event. Explain how this is suitably conservative.

15.01.05-3

General Design Criteria (GDC) 13 requires that instrumentation is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety.

Table 15.0-2 of the design control document (DCD) provides a value of 94.83 percent for the Core Protection Calculator (CPC) Low Reactor Coolant Pump (RCP) Shaft Speed Setpoint, which is used in the Chapter 15 safety analyses. The nominal trip setpoint, provided in DCD Table 7.2-4, is 95 percent. Therefore, the analysis assumes the RCP shaft speed measurement is accurate to within 0.17 percent. The small uncertainty associated with the Low RCP Shaft Speed trip has caused NRC staff to question if the modeling of this trip provides sufficient margin to account for uncertainty. NRC staff requests KHNP provide justification for the small uncertainty used in the modeling of the CPC Low RCP Shaft Speed setpoint.

15.01.05-4

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 52.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in SRP 15.0.3. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation under NUREG-0800, Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors.” NRC staff needs to ensure that a suitably conservative estimate is determined for the number of failed fuel rods.

Modeling of the loss of offsite power (LOOP) for the steam line break (SLB) event is treated differently between the return to power (RTP) and departure from nucleate boiling ratio (DNBR) analysis. In particular, the RTP analysis assumes a LOOP coincident with the break and the DNBR analysis assumes a LOOP coincident with a reactor and turbine trip. Since a LOOP coincident with the break would result in a lower reactor coolant system (RCS) flow at the time of minimum DNBR, NRC staff is questioning whether the DNBR case presented in the design control document (DCD) is bounding. NRC staff requests that KHNP conduct a SLB DNBR analysis assuming a LOOP coincident with the break to verify that the limiting DNBR case presented in the DCD Section 15.1.5 is bounding.

15.01.05-5

General Design Criteria (GDC) 13 requires that instrumentation is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety.

The limiting case for the return to power (RTP) analysis of the steam line break (SLB) event utilizes a variable overpower trip (VOPT) setpoint of 103.5 percent. However, DCD Table 7.2-4 provides a nominal VOPT trip setpoint of 109.6 percent and DCD Table 15.0-2 provides a safety analysis VOPT setpoint of 116.5 percent. NRC staff is requesting that KHNP:

1. Explain the basis for the 103.5 percent VOPT setpoint in the RTP analysis of the SLB event.

2. Explain how the 103.5 percent VOPT setpoint adequately accounts for instrument uncertainty.

15.01.05-6

General Design Criteria (GDC) 31 requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. GDC 31 is highlighted in the acceptance criteria of Section 15.1.5 of the Standard Review Plan (SRP), NUREG-0800.

Section 15.1.5 of the design control document (DCD) does not address SRP Section 15.1.5 acceptance criteria associated with GDC 31. Additionally, the figures contained in DCD Section 15.1.5 do not extend beyond 10 minutes. Therefore, NRC staff cannot verify that the RCS temperature remains above the temperature limit corresponding to safety injection shutoff head on the cooldown pressure-temperature limit curve for 30 minutes (the time at which operator action is credited). NRC staff requests Section 15.1.5 of the DCD be updated as required to demonstrate that the pressure-temperature limits are not violated during the limiting steam line break event.

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