KHNPDCDRAIsPEm Resource

From: Ciocco, Jeff
Sent: Thursday, December 17, 2015 11:42 AM
To: apr1400rai@khnp.co.kr; KHNPDCDRAIsPEm Resource; Harry (Hyun Seung) Chang; Andy Jiyong Oh; Christopher Tyree
Cc: Drzewiecki, Timothy; McKirgan, John; Steckel, James; Lee, Samuel
Subject: APR1400 Design Certification Application RAI 340-8395 (15.04.08 - Spectrum of Rod Ejection Accidents (PWR))
Attachments: APR1400 DC RAI 340 SRSB 8395.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, 45 days to respond to RAI question 15.04.08-5. We may adjust the schedule accordingly.

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

Thank you,

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United States Nuclear Regulatory Commission
Protecting People and the Environment

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REQUEST FOR ADDITIONAL INFORMATION 340-8395

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: 15.04.08 - Spectrum of Rod Ejection Accidents (PWR)
Application Section:

QUESTIONS

15.04.08-1

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 Standard Review Plan (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.

The statistical convolution method is presented in CENPD-183, "C-E Methods for Loss of Flow Analysis ," where it is reviewed for use with the CE-1 critical heat flux correlation. The approval letter for CENPD-183 states the condition that if a different critical heat flux correlation (CHF) is used, the applicant is required to submit a fuel damage probability distribution for the staff's approval. NRC staff requests that KHNP submit:

1. Fuel damage probability distribution for use with the KCE-1 CHF correlation described in APR1400-F-C-TR-12002-P, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design"

2. The data that is used to develop the damage probability distribution

15.04.08-2

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

A control element assembly (CEA) ejection time of 0.05 seconds is determined by assuming a 2500 psid pressure differential across the pressure boundary and no viscous or drag forces on the ejected CEA. This approach produces a conservatively low estimate for the CEA ejection time. However, parametric studies conducted in CENPD-170-A, "C-E Method for Control Element Assembly Ejection Analysis," demonstrate that for full power initial conditions a longer ejection time results in a larger net energy rise. NRC staff requests KHNP explain how the use of a 0.05 second ejection time is suitably conservative for all cases of the CEA ejection event.

15.04.08-3

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

During an audit of the calculations supporting Chapter 15 of the APR1400 Design Control Document, NRC staff observed that departure from nuclear boiling (DNB) analysis for the control element assembly ejection (CEAE) event is not performed at hot full power conditions, but is performed at 95% power. NRC staff requests justification for not conducting the DNB analysis for the CEAE event at hot full power conditions.

15.04.08-4

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

During an audit of the calculations supporting Chapter 15 of the APR1400 Design Control Document, NRC staff observed that departure from nuclear boiling (DNB) analysis for the control element assembly ejection (CEAE) event uses the post-ejected axial power shape in the hot channel and the pre-ejected axial power shape in the average channel. However, the fuel enthalpy analysis for the CEAE event uses the pre-ejected axial power shape in both the hot and average channels. The use of different shapes for these similar analyses caused NRC staff to question whether the chosen axial power shapes are conservative. NRC staff requests KHNP explain how the treatment of the axial power shape is suitably conservative for each analysis.

15.04.08-5

General Design Criteria (GDC) 28 requires reactivity control systems be designed with appropriate limits on potential reactivity increases so the effects of a rod ejection accident can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

Analysis of the control element assembly ejection (CEAE) event credits the variable overpower trip (VOPT) to initiate a reactor trip. The modeling of the VOPT uses values for [*CEILING*], [*STEP*], *and* [*EXCORE PENALTY*]. The values used for [*CEILING*] and [*STEP*] in the safety analysis appear to include an additional margin of 6.9% and 1.5% over the values provided in Table 7.2-4 of the APR1400 Design Control Document. Additionally, an 11% value is provided for the [*EXCORE PENALTY*] to account for potential deficiencies in the excore detector response resulting from a CEAE event. A description of how these values are obtained is missing from the application, which has caused staff to question the basis for the chosen values. NRC staff requests:

1. Explain how the values for [*CEILING*] and [*STEP*] used in the safety analysis sufficiently account for uncertainty in the VOPT setpoint.

2. Explain how an 11% value for the [*EXCORE PENALTY*] is suitably conservative.

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