KHNPDCDRAIsPEm Resource

From: Ciocco, Jeff
Sent: Thursday, December 03, 2015 10:38 AM
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Cc: Gilmer, James; McKirgan, John; Vera, John; Olson, Bruce; Lee, Samuel
Subject: APR1400 Design Certification Application RAI 326-8408 (04.04 - Thermal and Hydraulic Design)
Attachments: APR1400 DC RAI 326 SRSB 8408.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 questions 04.04-5 and 04.04-6. We may adjust the schedule accordingly.

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

Thank you,

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REQUEST FOR ADDITIONAL INFORMATION 326-8408

Issue Date: 12/03/2015
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 04.04 - Thermal and Hydraulic Design
Application Section: 4.4.2.4

QUESTIONS

04.04-1

10 CFR 50, Appendix A, General Design Criterion 10 requires the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Section 4.4.2.4 (Void Fraction Distribution) of the APR1400 DCD states that the “core average void fraction and the maximum void fraction are calculated using the Maurer method." The staff has not previously documented a review of this method, and cannot assess margin with respect to specified acceptable fuel design limits. Provide justification that the Maurer method results in conservative calculation of thermal hydraulic parameters used in the core design. This should include comparison to modern (state-of-the-art) void fraction data and to currently-accepted analytical methods for void fraction determination, such as the drift-flux method. Additionally, please identify which computer codes used for core design employ this method.

04.04-2

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 10 (Reactor Design) requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Section 4.4.4.2.3 (Core Pressure Drop Correlations) of the APR1400 DCD states that “To calculate pressure drop either for heating without boiling or for subcooled boiling, the friction factor for isothermal flow is modified through the use of the multipliers given by Pyle”. The staff has previously reviewed and approved use of the curve fit to the Martinelli-Nelson two-phase friction factor data, but has not documented an evaluation of the Pyle multipliers to account for the effects of subcooled voids on the acceleration and elevation components of the pressure drop. Provide justification that the use of Pyle multipliers on the two-phase friction factors results in conservative calculation of thermal hydraulic parameters used in core design. Additionally, please identify which computer programs utilized in core design employ this method, including appropriate justification for its use with each program.

04.04-3

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 10 (Reactor Design) requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Section 4.4 of the APR1400 DCD describes the use of the TORC and CETOP codes for core design and thermal design margin determination. Both of these codes are steady state subchannel analysis codes, and the previous staff approval was limited to steady state applications. Additionally, their application was limited to single phase flow or homogeneous two-phase flow (such as bubbly flow regime). Provide justification that these analysis methods conservatively predict the transient and two-phase effects of all anticipated operational occurrences. If an alternate approach is used to analyze anticipated operational occurrences where transient or two-phase effects are expected to be significant, provide a discussion of the alternate approach.

REQUEST FOR ADDITIONAL INFORMATION 326-8408

04.04-4

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 10 (Reactor Design) requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The approved version of the TORC code required a simplified representation of the core due to limitations on number of channels. Further simplification was necessary to represent the hot channel. More recent analytical methods do not have this limitation. Have the TORC and CETOP codes been modified to eliminate the limitation on number of channels? If so, provide a description of the modifications. If not, explain how this limitation has been addressed in the analyses. Additionally, demonstrate that the use of the simplified model is conservative for this specific design application (e.g., comparisons to more recent methods, such as RELAP5 - which staff notes is being used to analyze large break loss-of-coolant accidents and anticipated transients without scram).

04.04-5

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 10 (Reactor Design) requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Section 4.4.2.6.2 of the APR1400 DCD, “Reactor Vessel and Core Pressure Drops” refers to pressure losses calculated using classical fluid mechanistic relationship and information from the System 80+ reactor flow tests. Provide further discussion on the code or codes used to develop design pressure drops for reactor vessel components, the nodalization model, assumptions and boundary conditions, and the results of any sensitivity analyses. Also, provide a comparison of calculated results to test measurements.

04.04-6

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 10 (Reactor Design) requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Section 4.4.2.6.3 of the APR1400 DCD, “Hydraulic Loads on Internal Components” refers to determination from analytical methods and from the results of reactor flow model and component test programs. Provide further discussion on the code or codes used to develop vessel internal component design hydraulic loads, the nodalization model, assumptions and boundary conditions (including the treatment of crud buildup), and the results of any sensitivity analyses. Also, provide a reference to previous test measurements, such as fuel bundle uplift measurements. If the method combines loads resulting from seismic events and LOCA, discuss the treatment of uncertainties in the analyses.

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