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.:	**340-8395**
SRP Section:	**15.04.08 – Spectrum of Rod Ejection Accidents**
Application Section:	**15.04.08**
Date of RAI Issue:	**12/17/2015**

Question No. 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

Response

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"

Fuel damage probability distribution for use with the KCE-1 CHF correlation is as follows:

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

The damage probability distribution was obtained by combining system parameter uncertainties based on the Monte-Carlo method. The uncertainties of the system parameters and Code/CHF correlation were calculated based on the drawing tolerance and generic values of PLUS7 fuel and CHF test results as follows:

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Impact on DCD

There is no impact on DCD.

Impact on PRA

There is no impact on PRA.

Impact on Technical Specifications

There is no impact on Technical Specifications.

Impact on Technical/Topical/Environmental Reports

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

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Question No. 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.

Response

As specified in CENPD-190-A, a longer CEA ejection time affects the net energy rise (enthalpy rise) for the full power case; for the low power case, the effect is negligible. The fuel enthalpy rise is an important parameter for the evaluation of the pellet cladding mechanical interaction (PCMI) and the maximum fuel enthalpy. The following description explains the CEA ejection time change effect on the fuel enthalpy rise.

a) Maximum fuel enthalpy evaluation

The change of net energy increase is not bigger than [ ]TS while the CEA ejection time has changed from 0.05 second to 0.15 second. In other words, the difference of maximum fuel radial average enthalpy between 0.05 second and 0.15 second of CEA ejection time is smaller than [ ]TS (Figure 1). So, even though the longer CEA ejection time is considered, due to the small increase of net energy, there is still enough margin with regards to the fuel enthalpy limit (Figure 1). Therefore the CEA ejection time has an insignificant effect on the maximum fuel enthalpy during the CEAE event.

b) PCMI evaluation

A predominant parameter for the PCMI evaluation is the enthalpy rise within a prompt power pulse width. However, if there is no prompt power pulse, the PCMI is not expected to occur irrespective of the enthalpy rise value. In case of full power, there is no prompt power pulse due to the small ejected CEA worth, so the PCMI is not a concern at full power. Therefore, even though the small change of total net energy increase due to the longer ejection time exists, it has no effect on the full power PCMI evaluation result.

In conclusion, the 0.05 second CEA ejection time can be suitably used in the CEAE event.

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Impact on DCD

There is no impact on DCD.

Impact on PRA

There is no impact on PRA.

Impact on Technical Specifications

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Impact on Technical/Topical/Environmental Reports

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Question No. 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.

Response

The principal process variables that determine the thermal margin to the departure from nuclear boiling (DNB) in the core are monitored by the core operating limit supervisory system (COLSS). COLSS computes a power operating limit (POL), which assists the operator in maintaining the thermal margin in the core. COLSS monitors the same amount of thermal margin between 95% power and hot full power (HFP); therefore the initial DNBR margin is the same at the POL conditions. But the power dependent insertion limit (PDIL) at the 95% power level is deeper than for the HFP condition. The ejected rod worth, integrated radial peaking increase, and axial power shift during a CEAE event at the 95% power level are more conservative than those at HFP conditions. Also most positive moderator temperature coefficient (MTC) at 95% power is more positive than at HFP. Therefore the DNB analysis for the CEAE event is performed at 95% power.

Impact on DCD

There is no impact on DCD.

Impact on PRA

There is no impact on PRA.

Impact on Technical Specifications

There is no impact on Technical Specifications.

Impact on Technical/Topical/Environmental Reports

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

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Question No. 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.

Response

The 3-D power peaking factor variation during the CEAE event is composed of the variation of the radial and axial power peaking factors respectively. So the variation of each power peaking factor must be considered in the transient calculation. In terms of the average channel power during the CEAE event, it is conservative to use the pre-ejected average shape for minimizing the Doppler feedback in the point kinetics model, so that the core power is maximized. Therefore, the DNBR and fuel enthalpy case use the pre-ejected axial shape in the average channel.

For the DNBR calculation case, the variation of the radial power peaking factor is considered in the 2-D pin census file. The 2-D pin census file includes the number of the fuel pin undergoing the variation of pre- and post-ejected radial power peaking factor during the CEAE event. And the axial power peaking factor variation (axial power shape shift effect) is considered by using the post-ejected axial power shape for the hot channel.

For the enthalpy calculation case, the pre-ejected axial power shape is used for the hot channel and the 3-D power peaking factor variation is assumed as the radial power peaking factor variation. Since the net energy rise in the hot channel is directly proportional to the value of the post-ejected fuel pin radial peaking factor, this assumption is conservative. However, there is enough margin in the fuel enthalpy limit, so the conservative assumption is used in the fuel enthalpy calculation.

Impact on DCD

There is no impact on DCD.

Impact on PRA

There is no impact on PRA.

Impact on Technical Specifications

There is no impact on Technical Specifications.

Impact on Technical/Topical/Environmental Reports

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