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.:	**371-8456**
Review Section:	**07.07 – Control Systems**
Application Section:	**07.07**
Date of RAI Issued:	**01/19/2016**

Question No. 07.07-10

10 CFR 50.34 (b)(2)(i) requires, in part, that the technical contents of the applicantion include information pretaining to nuclear reactor control systems.

The second-to-last paragraph on page 53 of APR1400-Z-J-NR-14012, Control System CCF Analysis, appears to incorrectly state that Turbine Load Index (TLI) causes the excessive feedwater due to valve transfer from DV (downcomer valve) to EV (economizer valve). This is inconsistent with previous paragraph on page 53 and also inconsistent with the information provided in Table 5.1-1, "Shared Signals," of APR1400-Z-J-NR-14012. The applicant is requested to address this apparent inconsistency.

Response

The description of the second-to-last paragraph on page 53 of APR1400-Z-J-NR-14012, "Control System CCF Analysis," is incorrect and will be revised as follows:

1. Section 5.2.4.8 of the technical report (TeR) for control system CCF:

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2. Table 5.2-9 (Sh. 2 of 2) of the technical report (TeR) for control system CCF:

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Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

The second-to-last paragraph on page 53 and Table 5.2-9 of APR1400-Z-J-NR-14012, "Control System CCF Analysis," will be revised, as indicated in the attachment associated with this response.

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Table 5.2-9 Multiple Failures of Single Control group (RRS/RPCS) (Sh. 2 of 2)

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Question No. 07.07-14

10 CFR 50 Appendix A, General Design Criterion 10, “Reactor Design”, requires that the reactor be designed with sufficient 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. NUREG-0800 SRP Section 7.7 states, in part, that the effects of failures of control systems should not cause plant conditions more severe than those described in the analysis of design basis accidents and anticipated operational occurrences in Chapter 15 of the safety analysis report.

Section 5.2.4.13, "Feedwater HP Heater Bypass Line," of APR1400-Z-J-NR-14012-P, Revision 0, states multiple failures of the feedwater high pressure heater bypass line control group is bounded by the results of the DCD Section 15.1.2, Increase in Feedwater Flow event, because the maximum 25% increase in feedwater flow resulting from the postulated control system failures is less than that analyzed in DCD Section 15.1.2. However, the limiting increase in feedwater flow event evaluated in DCD Section 15.1.2 appears to be an inadvertent actuation of auxiliary feedwater, adding approximately 10% of the rated main feedwater flow.

Provide a more detailed description of the transient event resulting from the feedwater high pressure heater bypass line control group failures, including comparison to the DCD Section 15.1.2 Increase in Feedwater Flow event, in particular the assumed amount of feedwater increase, and confirm that the DNBR Specified Acceptable Fuel Design Limit (SAFDL) is not violated.

Response

As described in DCD Subsection 15.1.2.1, an increase in main feedwater flow could be caused by the further opening of a feedwater (FW) control valve and/or an increase in the feedwater pump speed by the high flow demand signal from the FW control system, or the inadvertent actuation of an auxiliary feedwater pump. In the event of a further opening of a feedwater control valve and/or an increase in the feedwater pump speed, the maximum increase of the feedwater flow, at full power, is less than 70 percent of the nominal flow for the main feedwater system, resulting in a total feedwater flow of 170 percent of the nominal flow. In the event of an inadvertent actuation of an auxiliary feedwater pump, the maximum increase in feedwater flow is less than 10 percent of the nominal flow for the main feedwater system, resulting in a total feedwater flow of less than 110 percent of the nominal flow. In the evaluation of the increased feedwater flow event presented DCD Subsection 15.1.2, an increase in feedwater flow of 170 percent of the nominal flow is assumed, because this assumption causes conservative results with respect to fuel integrity.

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Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

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

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Date of RAI Issued:	**01/19/2016**

Question No. 07.07-15

10 CFR 50 Appendix A, General Design Criterion 10, “Reactor Design”, requires that the reactor be designed with sufficient 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; and General Design Criterion 15, “Reactor Coolant System Design”, requires that the reactor coolant system be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. NUREG-0800 SRP Section 7.7 states, in part, that the effects of failures of control systems should not cause plant conditions more severe than those described in the analysis of design basis accidents and anticipated operational occurrences in Chapter 15 of the safety analysis report.

APR1400-Z-J-NR-14012-P, Revision 0, Section 5.3, "Failure Type 3: Multiple Failures of more than One Control Group," provides an explicit analysis of the limiting DNBR and primary system pressure transient events. The application of best-estimate methods is described in Section 5.3.2, "Assumptions Used in the Evaluation," and the selection of plant initial conditions is provided in Section 5.3.3, "Initial Conditions." The reactor characteristics, however, are not addressed in the report.

Provide a description of and the basis for selection of the parameters which significantly affect the Event 1, minimum DNBR and Event 2, maximum RCS pressure results. Provide such information as the time in life moderator temperature coefficient, axial power shape, radial power distributions, peaking factors, CEA worths, RCS core average flow rate and scram reactivity characteristics. Compare the parameter values used for the Events 1 and 2 calculations with similar parameter values used in the Chapter 15 accident analyses.

Response

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Figure 1. Moderator Density vs. Moderator Reactivity Feedback

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Figure 2. Fuel Temperature vs. Doppler Reactivity Feedback

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

The description of the section 5.3.2. of APR1400-Z-J-NR-14012, Revision 0 will be revised, as indicated in the attachment to this associated with this response.

5.3. Failure Type 3: Multiple Failures of more than One Control Group

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Question No. 07.07-16

10 CFR 50 Appendix A, General Design Criteria 10, “Reactor Design”, requires that the reactor be designed with sufficient 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. NUREG-0800 SRP Section 7.7 states, in part, that the effects of failures of control systems should not cause plant conditions more severe than those described in the analysis of design basis accidents and anticipated operational occurrences in Chapter 15 of the safety analysis report.

Section 5.3.5.1, "Fuel cladding integrity (Event 1)," of APR1400-Z-J-NR-14012-P, Revision 0, describes the limiting DNBR event resulting from multiple failures of more than one control group to be initiated as a result of an inadvertent opening of turbine bypass valves, increase in feedwater flow, and decrease in feedwater heating. The resulting cooldown of the primary system causes an increase in core power level due to moderator temperature reactivity feedback. Table 5.3-1, "Assumptions for Event 1," of APR1400-Z-J-NR-14012-P Revision 0, however, also identifies inadvertent withdrawal of CEAs as a contributor to this postulated event. A concurrent withdrawal of CEAs would result in a higher rate of reactivity insertion, and a potentially greater overshoot in core power and therefore greater challenge to the DNBR SAFDL.

Provide clarification of whether the limiting DNBR event resulting from multiple failures of more than one control group as presented in Section 5.3.5.1 of APR1400-Z-J-NR- 14012-P, Revision 0, includes a concurrent CEA withdrawal and supporting justification if a CEA withdrawal is not included.

Response

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Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

The description of Section 5.3.5.1. of APR1400-Z-J-NR-14012, Revision 0 will be revised, as indicated in the attachment to this associated with this response.

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