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.:	**339-8415**
SRP Section:	**15.01.05 - Steam System Piping Failures Inside and Outside of Containment (PWR)**
Application Section:	**15.01.05**
Date of RAI Issue:	**12/17/2015**

Question No. 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-ANR-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.

Response

1. The flow mixing tests at 0.248 scale, representing a two steam generator NSSS system with 133 fuel assemblies were conducted with a modelled closed core. In the experiment,

SO_2 gas was injected into one of the four reactor vessel inlet air streams. The distribution of SO_2 inside 133 core flow passages and the two reactor vessel outlets was measured by sampling the SO_2 concentration in the air at these locations. The ratio of the SO_2 concentration in each flow passage to the concentration in the inlet was used to determine the flow distribution fraction between an inlet and any core flow passage or core outlet. From these two fraction values, the mixing parameter at the core inlet and at the core outlet were calculated.

2. The mixing parameter values used in the steam-line break (SLB) analysis have been determined experimentally as mentioned above. The conservatism for the SLB analysis is obtained by using the affected cold leg temperature, which represents the lowest core temperature, as the reference temperature considering the reactivity insertion by the moderator temperature feedback effect during core cooling down by the SLB.

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, or Environment Report.

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Question No. 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.

Response

1. In CESEC-III, the safety injection flow vs. pressure is based on the minimum pump flow provided in DCD Table 6.3.2-4. A safety injection flow corresponding to a certain pressure is calculated by interpolation within CESEC-III.

2. The inverse boron worth (IBW, ppm/%$\Delta\rho$) as a function of core average coolant temperature is used to account for the addition of negative reactivity by boron injection during cooldown, and it is calculated for N-1 rod (the most reactive rod stuck out) configuration at EOC (End of Cycle) condition (0 ppm) because the consequences of steam line break event are more severe at EOC. In order not to overestimate the credit of boron injection, the maximum IBW is used, and it is obtained by applying the appropriate uncertainties to the best estimated IBWs. Therefore, the boron reactivity in the CESEC-III analysis for the steam line break event is determined based on the relevant boron concentration with the maximum IBW during the event.

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, or Environment Report.

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Question No. 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.

Response

The rated speed of the RCP is 1190 rpm. The measurement channel error for the RCP shaft speed is ±2 rpm, which is 0.17 percent at 1190 rpm. Therefore, a value of 94.83 percent for the CPC RCP Shaft Speed Setpoint is used in the Chapter 15 safety analyses.

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, or Environment Report.

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Question No. 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.

Response

The nominal setpoint of the VOPT is 109.6 % and its channel uncertainty is +6.1 % / -6.9 %. Therefore, for the VOPT, the maximum and minimum analysis setpoints are determined as 116.5 % and 103.5%, respectively.

In addition, because the minimum analysis setpoint of 103.5 % is used in the RTP analysis of the SLB event, the relevant value will be added to Table 15.0-2.

Impact on DCD

DCD Chapter 15.0 (Table 15.0-2) will be revised as indicated on the attached markup.

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, or Environment Report.

Table 15.0-2

<u>Reactor Protection System Trips Used in the Safety Analysis</u>

Event	RPS	Analysis Setpoint (1)	Sensor Response Time	Reactor Trip Delay Time (2)
Events not Mentioned Below	High Logarithmic Power Level	0.05 %	0 ms	550 ms
	Variable Overpower	116.5 %	0 ms	550ms
	CPC Variable Overpower	115 %	0 ms	650 ms
	High Pressurizer Pressure	169.7 kg/cm^2A (2,414 psia)	300 ms	550 ms
	Low Pressurizer Pressure	122.0 kg/cm^2A (1,735 psia)	600 ms	550 ms
	Low SG Pressure	57.1 kg/cm^2A (812 psia)	600 ms	550 ms
	Low SG Water Level	40.7 % wide range (3)	650 ms	600 ms
	High SG Water Level	95 % narrow range (4)	600 ms	550 ms
	Low Reactor Coolant Flow	80 % (5)	0 ms	1200 ms (7)
	CPC Low RCP Shaft Speed	94.83 %	0 ms	450 ms
	CPC Coincident Low Pressure/DNBR	140.6 kg/cm^2A (2,000 psia) /1.45 (6)	300 ms	650 ms
Feedwater and Steam Line Breaks	High Pressurizer Pressure	173.17 kg/ cm^2A (2,463 psia)	300 ms	550 ms
	Low Pressurizer Pressure	109.3 kg/cm^2A (1,555 psia)	600 ms	550 ms
	Low SG Pressure	52.7 kg/cm^2A (750 psia)	600 ms	550 ms
	Low SG Water Level	28.4 % wide range (3)	650 ms	600 ms
	High SG Water Level	95 % narrow range (4)	600 ms	550 ms
	Low Reactor Coolant Flow	60 % (5)	0 ms	850 ms (7)
	CPC Low RCP Shaft Speed	94.83 %	0 ms	450 ms
	CPC Variable Overpower	121 % (8)	0 ms	650 ms
	High Containment Pressure	0.28 kg/cm^2G (4 psig)	600 ms	550 ms

(1) Some Chapter 15 analyses assumed more conservative setpoints for specific events

Variable Overpower 103.5 % 0 ms 550 ms

(3) Percent of distance between the wide-range instrument taps; the setpoint is valid at full power only (i.e., 100 − 102 % power).

(4) Percent of distance between the narrow-range instrument taps

(5) Percent of hot leg flow

(6) Trip credited for 15.6.3 events

(7) The total response time is the sum of sensor response time and reactor trip delay time. For a shaft break event, a reactor trip is required 1.2 seconds after the flow in the hot leg reaches its analysis setpoint. For a steam line break (SLB) with a LOOP up to 30 minutes into the event, a reactor trip is required 0.85 second after the core flow reaches its analysis setpoint.

(8) For SLB outside the containment, an additional 6 percent is considered conservative.