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.:	**108-7973**
SRP Section:	**15.00.03 – Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors**
Application Section:	**SRP 15.0**
Date of RAI Issue:	**07/23/2015**

Question No. 15.00.03-03

The initial reactor coolant system (RCS) liquid mass is not consistent in all DBA dose analyses as expected – explain the differences. The initial reactor coolant mass is listed as input to analyses in the following DCD tables, and the value is different in each: Table 11.1-1 (coolant concentration), Table 15.1.5-12 (main steam line break), Table 15.2.8-3 (feedwater line break), Table 15.3.3-3 (reactor coolant pump rotor seizure), Table 15.4.8-4 (control element assembly ejection), Table 15.6.2-4 (small line break), Table 15.6.3-5 (steam generator tube rupture), and Table 15.6.5-13 (LOCA).

Response

The initial RCS liquid mass is determined by the CESEC-III code using the thermal and hydraulic initial conditions of the RCS and main steam supply system. The inconsistencies of the initial RCS liquid masses used for each of the DBAs, except for the LOCA, are caused by the differences in initial conditions of each accident, which are determined to maximize the radiological consequences. Major initial condition parameters used to determine the initial RCS liquid mass are the core inlet temperature, core mass flow rate, pressurizer pressure, and pressurizer water volume, which are provided in Table 1. The RCS liquid mass given in Table 11.1-1 is not addressed in Table 1 since it is used to determine the reactor coolant fission product source term for the normal operation condition and is not relevant to the DBA dose analyses. For the LOCA, the same RCS liquid mass as the SGTR event was used for its radiological consequence, of which the impact was estimated to be negligible compared to the total dose (=0.0028%).

Table 1 Major Parameters Used to Determine the Initial RCS Liquid Mass

Relevant Table	Mass, kg	Major Parameters
Table 15.1.5-12 (MSLB)	274,392 (Full Power)	Core inlet temperature, ℃ : 296.11 Core mass flow rate, 10^6 kg/hr : 85.03 Pressurizer pressure, kg/cm^2 A : 163.5 Pressurizer liquid volume, m^3 : 39.91
	286,829 (Zero Power)	Core inlet temperature, ℃ : 295 Core mass flow rate, 10^6 kg/hr : 69.64 Pressurizer pressure, kg/cm^2 A : 163.5 Pressurizer liquid volume, m^3 : 39.91
Table 15.2.8-3 (FLB)	288,086	Core inlet temperature, ℃ : 296.11 Core mass flow rate, 10^6 kg/hr : 69.64 Pressurizer pressure, kg/cm^2 A : 159.6 Pressurizer liquid volume, m^3 : 39.91
Table 15.3.3-3 (RCP rotor seizure)	272,000	Core inlet temperature, ℃ : 287.8 Core mass flow rate, 10^6 kg/hr : 69.64 Pressurizer pressure, kg/cm^2 A : 163.5
Table 15.4.8-4 (CEA ejection)	267,620	Core inlet temperature, ℃ : 295 Core mass flow rate, 10^6 kg/hr : 69.64 Pressurizer pressure, kg/cm^2 A : 163.5
Table 15.6.2-4 (small line break)	292,431	Core inlet temperature, ℃ : 296.11 Core mass flow rate, 10^6 kg/hr : 69.64 Pressurizer pressure, kg/cm^2 A : 163.5 Pressurizer liquid volume, m^3 : 39.91
Table 15.6.3-5 (SGTR)	290,680	Core inlet temperature, ℃ : 295 Core mass flow rate, 10^6 kg/hr : 69.64 Pressurizer pressure, kg/cm^2 A : 163.5 Pressurizer liquid volume, m^3 : 39.91

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 Environmental Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**108-7973**
SRP Section:	**15.00.03 – Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors**
Application Section:	**SRP 15.0**
Date of RAI Issue:	**07/23/2015**

Question No. 15.00.03-04

The initial secondary liquid mass in the steam generators is not consistent in all DBA dose analyses as expected-explain the differences. The initial secondary coolant liquid mass is given as analysis input in the following DCD table, and the value is different in each: Table 11.1-5 (coolant concentration), Table 15.1.5-12 (main steam line break), Table 15.2.8-3 (feedwater line break), Table 15.3.3-3 (reactor coolant rotor seizure), Table 15.4.8-4 (control element assembly ejection), Table 15.6.2-4 (small line break), and Table 15.6.3-5 (steam generator tube rupture).

Response

The initial secondary liquid mass is determined by the CESEC-III code using the thermal and hydraulic initial conditions of the RCS and main steam supply system. The inconsistencies of the initial secondary liquid masses used for each of the DBAs are caused by the differences in initial conditions of each accident, which are determined to maximize the radiological consequences. Major initial condition parameters used to determine the initial secondary liquid mass in the SG are the steam generator pressure, steam generator temperature, and steam generator water level, which are provided in Table 1. The secondary liquid mass in the SG given in Table 11.1-5 is not addressed in Table 1 since it is used to determine the secondary source term for the normal operation condition and is not relevant to the DBA dose analyses.

Table 1 Major Parameters Used to Determine Initial Secondary Liquid Mass in the SG

Relevant Table	Mass, kg	Major Parameters
Table 15.1.5-12 (MSLB)	54,592 (Full Power)	Steam generator temperature, ℃: 292.1 Steam generator pressure, kg/cm^2 A: 78.24 Steam generator water level, %: 40.7WR
	187,658 (Zero Power)	Steam generator temperature, ℃: 295 Steam generator pressure, kg/cm^2 A: 81.62 Steam generator water level, %: 97.68WR
Table 15.2.8-3 (FLB)	89,721	Steam generator temperature, ℃: 291 Steam generator pressure, kg/cm^2 A: 77 Steam generator water level, %: 77WR
Table 15.3.3-3 (RCP rotor seizure)	84,200	Steam generator temperature, ℃: 282.8 Steam generator pressure, kg/cm^2 A: 68.3 Steam generator water level, %: 70.5WR
Table 15.4.8-4 (CEA ejection)	104,326	Steam generator temperature, ℃: 289.9 Steam generator pressure, kg/cm^2 A: 75.9 Steam generator water level, %: 70.5WR
Table 15.6.2-4 (small line break)	89,086	Steam generator temperature, ℃: 290.78 Steam generator pressure, kg/cm^2 A: 76.81 Steam generator water level, %: 76.87WR
Table 15.6.3-5 (SGTR)	117,688	Steam generator temperature, ℃: 289.9 Steam generator pressure, kg/cm^2 A: 75.9 Steam generator water level, %: 95NR

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 Environmental Report.

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.:	**108-7973**
SRP Section:	**15.00.03 – Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors**
Application Section:	**SRP 15.0**
Date of RAI Issue:	**07/23/2015**

Question No. 15.00.03-06

Clarify the assumptions for MSLB releases through the affected SG, both break flow and steaming, and releases through the unaffected SG.

a. DCD Table 15.1.5-12 states that the release from the affected SG is terminated at 30 minutes, which is coincident with the operator initiating cooldown. However, the table also gives mass releases through 8 hours for the affected SG.

b. DCD Table 15.1.5-12 states that the release through the unaffected SG (both primary-to-secondary leakage and steaming) is terminated at 8 hours. However, the table only gives mass releases from the unaffected SG through 0.5 hours.

Response

Releases from both the unaffected and affected SGs are terminated at 30 minute and 8 hour, respectively, which is consistent with the time-dependent release in Table 15.1.512 (2 of 3). The relevant parameter descriptions in Table 15.1.5-12 (2 of 3) were editorial errors, and will be revised.

Impact on DCD

DCD Table 15.1.5-12 (2 of 3) will be revised as indicated in Attachment.

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 Environmental Report.

Table 15.1.5-12 (2 of 3)

Parameter	Value
Secondary System Activity Transport Model	
Primary-to-secondary Leakage Rate through SGs	2.27 L/min (0.6 gpm) for two SGs
Integrated P-T-S Leakage 0 ~ 2 hr 2 ~ 8 hr	 272 kg (601 lbm) 818 kg (1,803 lbm)
Total Mass Release from Affected SG For SLBFPDLOOP 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr For SLBZPLOOPD 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr	 196,862 kg (434,000 lbm) 241,315 kg (532,000 lbm) 657,720 kg (1,450,000 lbm) 158,760 kg (350,000 lbm) 226,346 kg (499,000 lbm) 639,576 kg (1,410,000 lbm)
Total Mass Release from Unaffected SG For SLBFPDLOOP 0 ~ 0.5 hr 0.5 ~ 8 hr For SLBZPLOOPD 0 ~ 0.5 hr 0.5 ~ 8 hr	 40,824 kg (90,000 lbm) 0.0 kg (0.0 lbm) 39,010 kg (86,000 lbm) 0.0 kg (0.0 lbm)
Termination of Release from ~~Affected~~ Unaffected SG	30 min
~~Unaffected~~ Affected SG P-T-S Leak Duration, and Termination of Release from ~~Unaffected~~ Affected SG	8 hr
SG Liquid Iodine Partition Coefficient	100
Letdown System Flow Rate	18,100 kg/hr (39,842 lbm/hr)
RCS Fluid Released to IRWST	5,443 kg (12,000 lbm) For SLBFPDLOOP 2,948 kg (6,500 lbm) For SLBZPLOOPD

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Docket No. 52-046

RAI No.:	**108-7973**
SRP Section:	**15.00.03 – Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors**
Application Section:	**Chapter 15 including 15A**
Date of RAI Issue:	**07/23/2015**

Question No. 15.00.03-14

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.

With respect to the reactor coolant pump rotor seizure accident dose analysis, on DCD page 15.3-12, it states that during the first 30 minutes both SGs are expected to have the tubes uncovered. It also states that during this period, the primary-to-secondary leakage flashing fraction averages 15 percent. RG 1.183 guidance is that all of the primary-to-secondary leakage to that SG is assumed to flash to vapor during periods of dryout. Provide the basis for this difference from the guidance.

Response

Both SGs are expected to have the tubes uncovered but the dryout of both SGs does not occur during the reactor coolant pump rotor seizure accident. Based on this analysis result, the flashing fraction of the primary-to-secondary leakage is calculated using the below equation by considering the thermodynamic conditions in the reactor and the secondary coolant during the reactor coolant pump rotor seizure accident.

$$\text{Flashing Fraction} = (H_{RCS} - H_{SG,f}) / H_{SG,fg}$$

Where, H_{RCS} = leaked reactor coolant enthalpy (hot leg), Btu/lbm
$H_{SG,f}$ = saturated liquid enthalpy at steam generator pressure, Btu/lbm
$H_{SG,fg}$ = vaporization enthalpy at steam generator pressure, Btu/lbm

The average calculated flashing fraction during the first 30 minutes is less than 15%.

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 Environmental Report.

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.:	**108-7973**
SRP Section:	**15.00.03 – Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors**
Application Section:	**Chapter 15 including 15A**
Date of RAI Issue:	**07/23/2015**

Question No. 15.00.03-16

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.

With respect to the control element assembly (CEA) ejection accident dose analysis, on DCD page 15.4-33, it states that during the first 30 minutes both SGs are expected to have the tubes uncovered. It also states that during this period, the primary-to-secondary leakage flashing fraction averages 15 percent. RG 1.183 guidance is that all of the primary-to-secondary leakage to that SG is assumed to flash to vapor during periods of dryout. Provide the basis for this difference from the guidance.

Response

Both SGs are expected to have the tubes uncovered but the dryout of both SGs does not occur during the control element assembly (CEA) ejection accident. Based on this analysis result, the flashing fraction of the primary-to-secondary leakage is calculated using the below equation by considering the thermodynamic conditions in the reactor and the secondary coolant during the CEA ejection accident.

$$\text{Flashing Fraction} = (H_{RCS} - H_{SG,f}) / H_{SG,fg}$$

Where, H_{RCS} = leaked reactor coolant enthalpy (hot leg), Btu/lbm
$H_{SG,f}$ = saturated liquid enthalpy at steam generator pressure, Btu/lbm
$H_{SG,fg}$ = vaporization enthalpy at steam generator pressure, Btu/lbm

The average calculated flashing fraction during the first 30 minutes is less than 15%.

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 Environmental Report.