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.:	**370-8450**
SRP Section:	**15.06.03 - Radiological Consequences of Steam Generator Tube Failure (PWR)**
Application Section:	**15.06.03**
Date of RAI Issue:	**01/19/2016**

Question No. 15.06.03-1

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, General Design Criteria (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 the analysis showing no failure of fuel is suitably conservative.

The thermal margin analysis, i.e. Minimum Departure from Nucleate Boiling (MDNBR) presented in Section 15.6.3 of the Design Control Document (DCD), is missing the following information:

1. Figures that show dynamic behavior of important Nuclear Steam Supply System (NSSS) parameters for the CESEC-III analysis that corresponds to the limiting MDNBR analysis.

2. A table that presents the chronological list of events that occur during the steam generator tube rupture event that corresponds to the limiting MDNBR analysis. This table should include, at a minimum, the time and associated setpoint of any reactor protection system (RPS) or engineered safety features (ESFs) actuation, and any significant operator actions.

Due to the missing information, NRC staff is unable to state the APR1400 meets GDC 13 for the instrumentation credited in the SGTR event, or that the analysis presented in the DCD

represents a suitably conservative analysis. NRC staff requests the DCD be updated with the information described in items 1 and 2 above, and item 3 below.

3. Update DCD Table 15.0-2 to include the core protection calculator (CPC) hot leg temperature trip used in the safety analysis along with the associated sensor response time and reactor trip delay time.

Response

1. Figures with dynamic behavior of important Nuclear Steam Supply System (NSSS) parameters for the CESEC-III analysis that corresponds to the limiting MDNBR analysis are shown in Figure 1 through Figure 4.

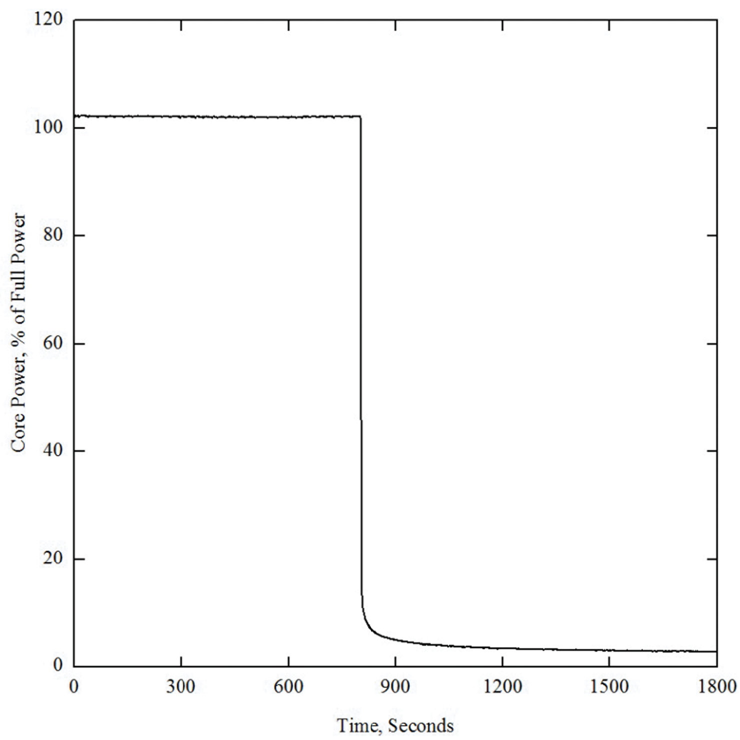

Figure 1 SGTR with Concurrent LOOP corresponds to MDNBR:
Core Power vs. Time

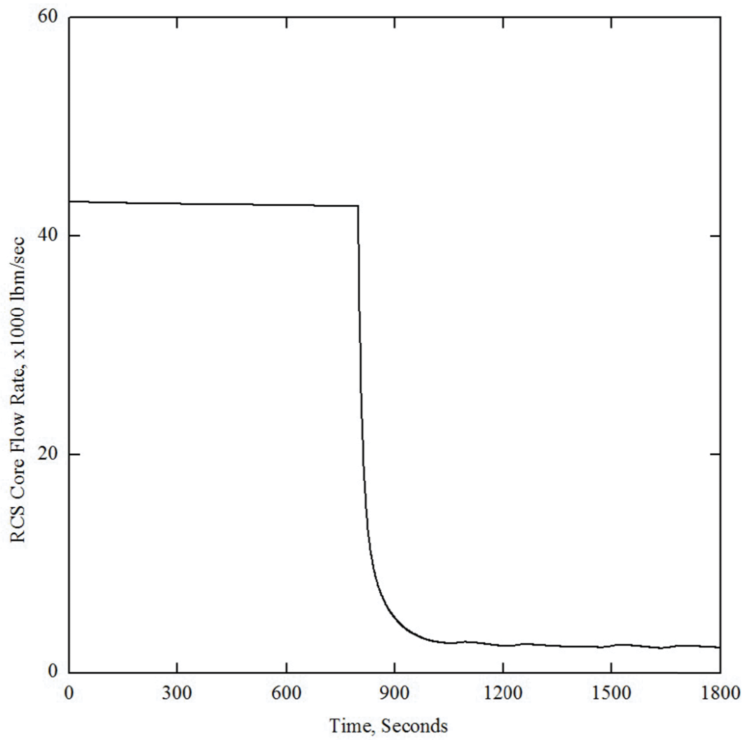

Figure 2 SGTR with Concurrent LOOP corresponds to MDNBR:
RCS Core Flow Rate vs. Time

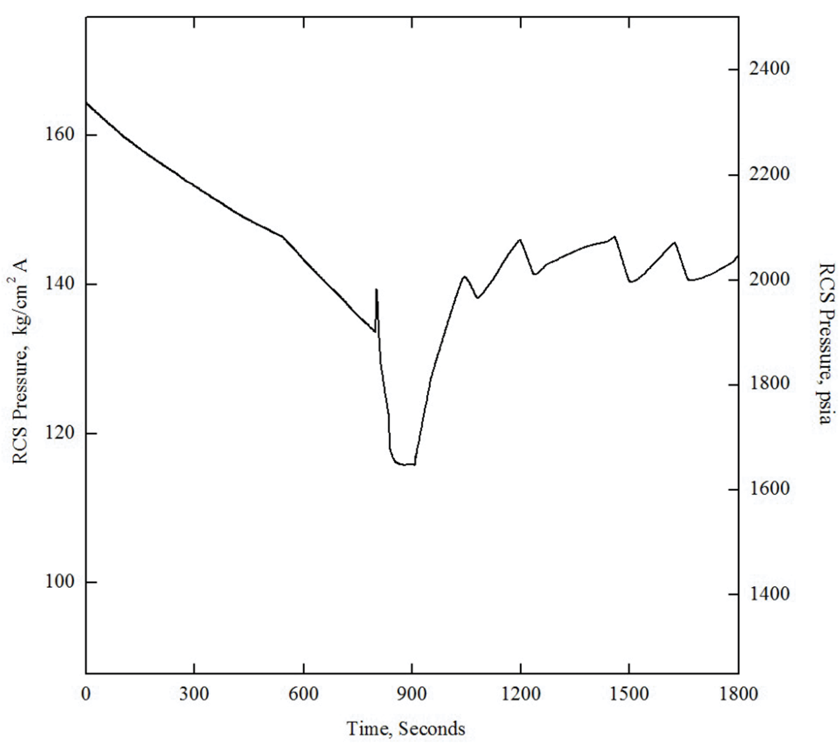

Figure 3 SGTR with Concurrent LOOP corresponds to MDNBR:
RCS Pressure vs. Time

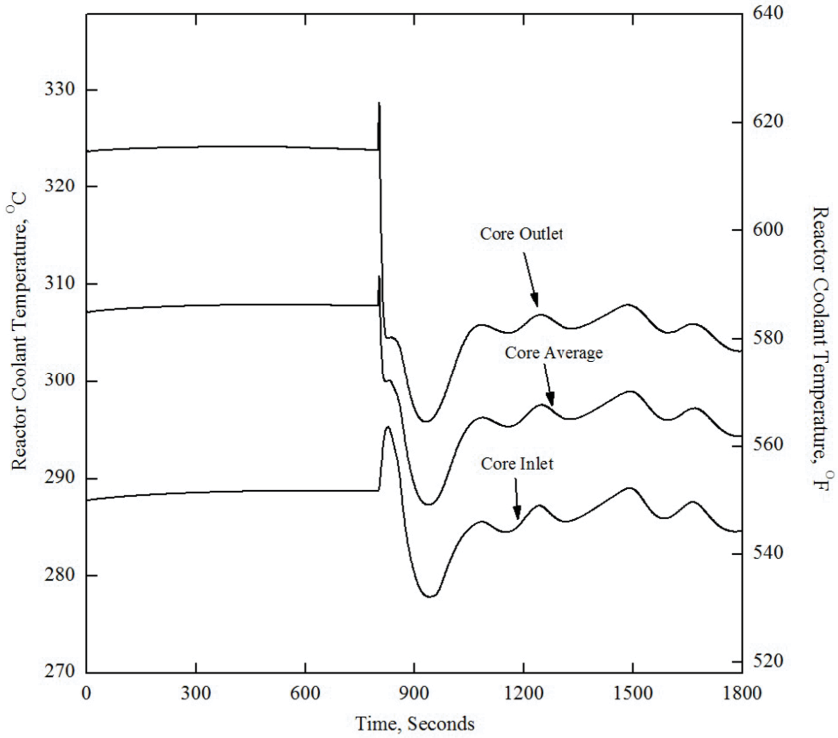

Figure 4 SGTR with Concurrent LOOP corresponds to MDNBR: Reactor Coolant Temperature vs. Time

2. Table 1 below presents the chronological list of events that occur during the SGTR event that corresponds to the limiting MDNBR analysis. The occurrence of fuel failure was evaluated as shown in DCD Tier 2, Figures 15.6.3-16 and 15.6.3-32. As found in DCD Figures 15.6.3-16 and 15.6.3-32, the MDNBRs occur shortly after the reactor trip and sharply rise, therefore, to present important TH behaviors after this point is not practical for MDNBR analysis. In addition, the minimum DNBR case and the dynamic behavior of important parameters for the limiting radiological consequences are already addressed as shown in DCD Subsection 15.6.3.2.3 and 15.6.3.2.5, respectively.

Table 1 Sequence of Events for a Steam Generator Tube Rupture with a Loss of Offsite Power Corresponds to MDNBR

Time (Sec)	Event	Setpoint or Value
0.0	Tube Rupture Occurs	-
799.85	Trip Breakers Open due to Hot Leg Saturation Temperature Signal	-
799.85	Turbine Trip: Turbine Stop Valves Start to Close	-
799.85	Loss of Offsite Power	-
806.10	Main Steam Safety Valves Open, kg/cm^2A (psia)	80.27 (1,141.74)
812.65	Maximum Steam Generator Pressure, kg/cm^2A (psia)	82.88 (1,178.81)
811.8	Safety Injection Actuation Signal On, kg/cm^2A (psia)	132.53 (1,885)
851.8	Safety Injection Flow Begins	-
1,800	Operator Cools the NSSS Using Plant Emergency Procedure After Isolation of Affected Steam Generator or Confirmation of Isolation	-
28,800	Shutdown Cooling Entry Conditions are Assumed to be Reached, RCS Pressure, kg/cm^2A (psia)/ Temperature, °C (°F)	31.64/176.7 (450/350)

3. The relevant value will be added to Table 15.0-2.

Impact on DCD

DCD Chapter 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.

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

CPC Hot Leg Saturation	T_{sat} - 7.2°C (T_{sat} - 13°F)	8000 ms	650 ms

Reactor Protection System Trips Used in the Safety Analysis

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^{2}A (2,414 psia)	300 ms	550 ms
	Low Pressurizer Pressure	122.0 kg/cm^{2}A (1,735 psia)	600 ms	550 ms
	Low SG Pressure	57.1 kg/cm^{2}A (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^{2}A (2,000 psia) /1.45 (6)	300 ms	650 ms
Feedwater and Steam Line Breaks	High Pressurizer Pressure	173.17 kg/ cm^{2}A (2,463 psia)	300 ms	550 ms
	Low Pressurizer Pressure	109.3 kg/cm^{2}A (1,555 psia)	600 ms	550 ms
	Low SG Pressure	52.7 kg/cm^{2}A (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^{2}G (4 psig)	600 ms	550 ms

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

(2) Reactor protection system response time testing is described in Section 7.2.

(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.

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Question No. 15.06.03-3

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, General Design Criteria (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 the analysis showing no failure of fuel is suitably conservative.

During an audit of the calculations supporting Chapter 15 of the APR1400 Design Control Document (DCD), NRC staff observed that departure from nuclear boiling (DNB) analysis for the steam generator tube rupture (SGTR) event resulted in a violation of the safety limit for a few cases. In particular, two cases that are initialized to preserve a required overpower margin (ROPM) of 18% resulted in a minimum DNB ratio less than 1.29. During a public phone call on October 27, 2015 (ADAMS Accession No. ML15289A410) KHNP stated that the ROPM set aside by the Core Operating Limit Supervisory System (COLSS) is greater than 18%. Because the limiting DNB analysis for SGTR presented in the DCD assumes 20% ROPM and shows no fuel failure, but additional analyses that assume 18% ROPM show violation of the safety limit, NRC staff is questioning if the case presented in the DCD represents a bounding case in terms of minimum DNB ratio and dose consequences. NRC staff requests KHNP either explain why the DNB analysis presented in DCD Section 15.6.3 represents the limiting case, or update the DCD with the limiting case.

Response

An SGTR with a required overpower margin (ROPM) of 18% resulted in a minimum DNB ratio less than 1.29, and the radiological consequences would exceed the relevant acceptance criteria with the expected fuel failure. However, an SGTR with a ROPM of 20% resulted in a minimum DNB ratio above 1.29, and the radiological consequences would meet the relevant acceptance criteria. The ROPM is considered as a penalty for the core operating limit supervisory system (COLSS), and a ROPM of 20% is within the acceptable range in the COLSS even though this penalty would be a burden to the operator. Therefore, the DNB analysis presented in DCD Section 15.6.3 represents the limiting case with respect to the radiological consequences, because a ROPM of 20% is the minimum value required to maintain MDNBR above the specified acceptable fuel design limit (SAFDL).

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

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Impact on Technical/Topical/Environmental Reports

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