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. 15.01.05-4

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.

Modeling of the loss of offsite power (LOOP) for the steam line break (SLB) event is treated differently between the return to power (RTP) and departure from nucleate boiling ratio (DNBR) analysis. In particular, the RTP analysis assumes a LOOP coincident with the break and the DNBR analysis assumes a LOOP coincident with a reactor and turbine trip. Since a LOOP coincident with the break would result in a lower reactor coolant system (RCS) flow at the time of minimum DNBR, NRC staff is questioning whether the DNBR case presented in the design control document (DCD) is bounding. NRC staff requests that KHNP conduct a SLB DNBR analysis assuming a LOOP coincident with the break to verify that the limiting DNBR case presented in the DCD Section 15.1.5 is bounding.

Response

A LOOP coincident with the break causes an early reactor trip on the core protection calculator (CPC) low reactor coolant pump (RCP) shaft speed trip. SLB DNBR analysis assuming a

LOOP coincident with a reactor and turbine trip generates more adverse results than assuming a LOOP coincident with the break (Figure 1). Therefore, the limiting DNBR case presented in the DCD Section 15.1.5 is bounding.

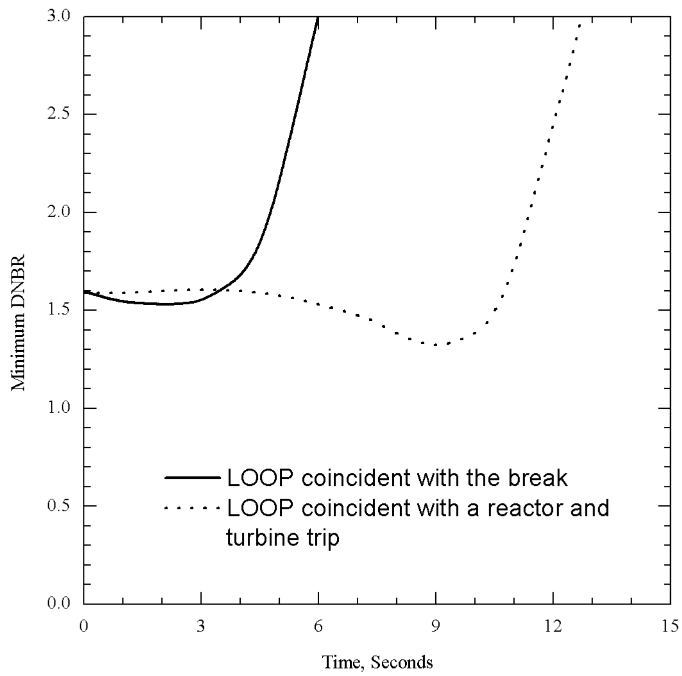

Figure Minimum DNBR vs Time on LOOP

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. 15.01.05-6

General Design Criteria (GDC) 31 requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. GDC 31 is highlighted in the acceptance criteria of Section 15.1.5 of the Standard Review Plan (SRP), NUREG-0800.

Section 15.1.5 of the design control document (DCD) does not address SRP Section 15.1.5 acceptance criteria associated with GDC 31. Additionally, the figures contained in DCD Section 15.1.5 do not extend beyond 10 minutes. Therefore, NRC staff cannot verify that the RCS temperature remains above the temperature limit corresponding to safety injection shutoff head on the cooldown pressure-temperature limit curve for 30 minutes (the time at which operator action is credited). NRC staff requests Section 15.1.5 of the DCD be updated as required to demonstrate that the pressure-temperature limits are not violated during the limiting steam line break event.

Response

During the SLB accident, since a fast decrease of the RCS pressure at the initial stage causes the safety injection, the peak core reactivity occurs within 10 minutes as shown in Figures 15.1.5-1.7, 15.1.5-2.7, 15.1.5-3.7, 15.1.5-4.7 and 15.1.5-5.6. Therefore, the figures contained in DCD Section 15.1.5 need not extend beyond 10 minutes. For P-T limits, as shown in Figures 1 through 10, the RCS temperature remains above the temperature limit corresponding to safety injection shutoff head on the cooldown pressure-temperature limit curve for 30 minutes.

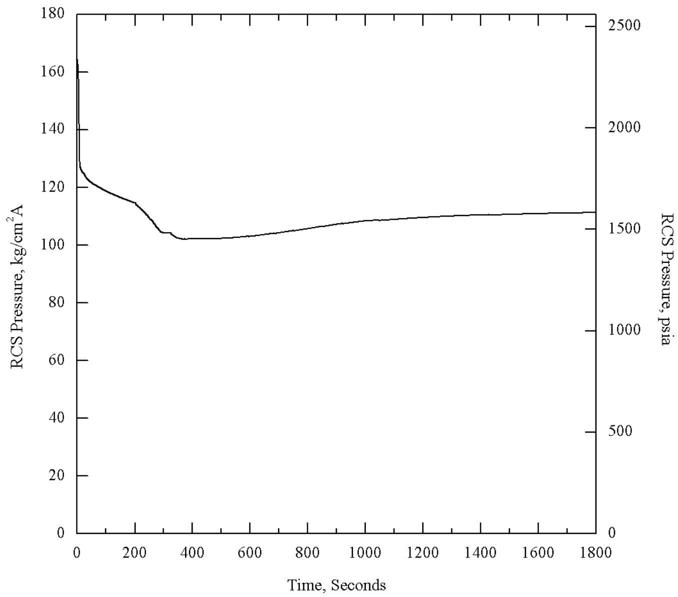

Figure 1. Full Power Large Steam Line Break with Concurrent LOOP:
RCS Pressure vs. Time

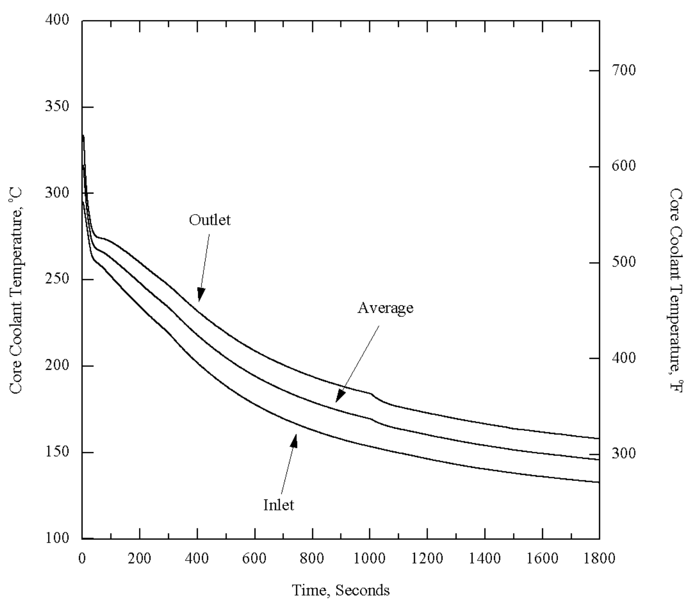

Figure 2. Full Power Large Steam Line Break with Concurrent LOOP:
Core Coolant Temperature vs. Time

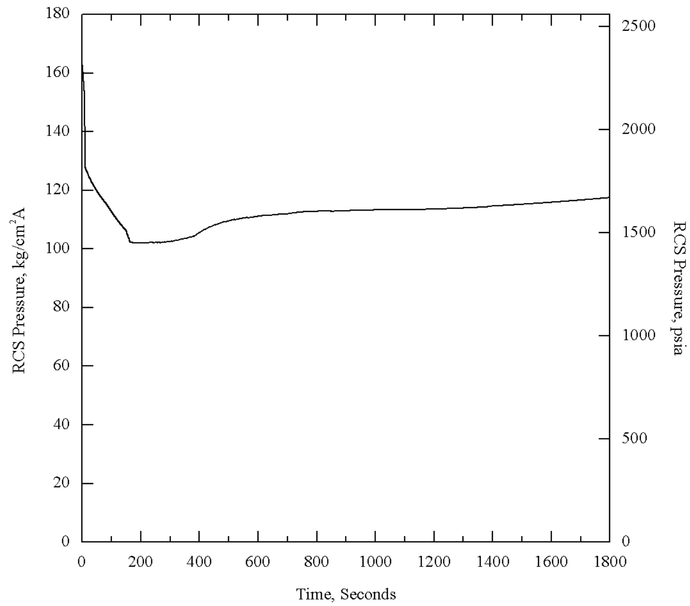

Figure 3. Full Power Large Steam Line Break with Offsite Power Available: RCS Pressure vs. Time

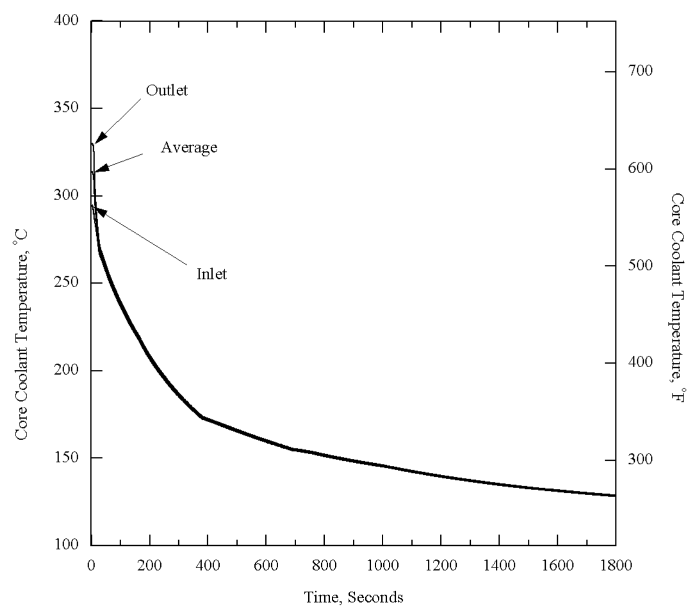

Figure 4. Full Power Large Steam Line Break with Offsite Power Available: Core Coolant Temperature vs. Time

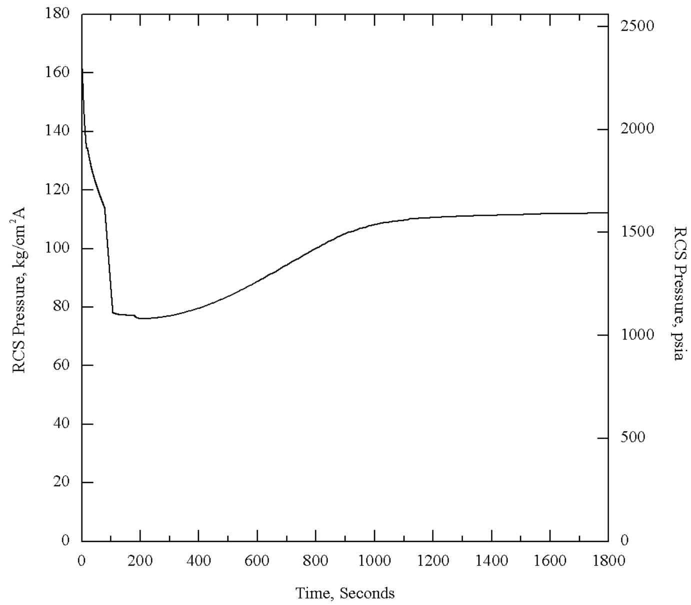

Figure 5. Zero Power Large Steam Line Break with Concurrent LOOP:
RCS Pressure vs. Time

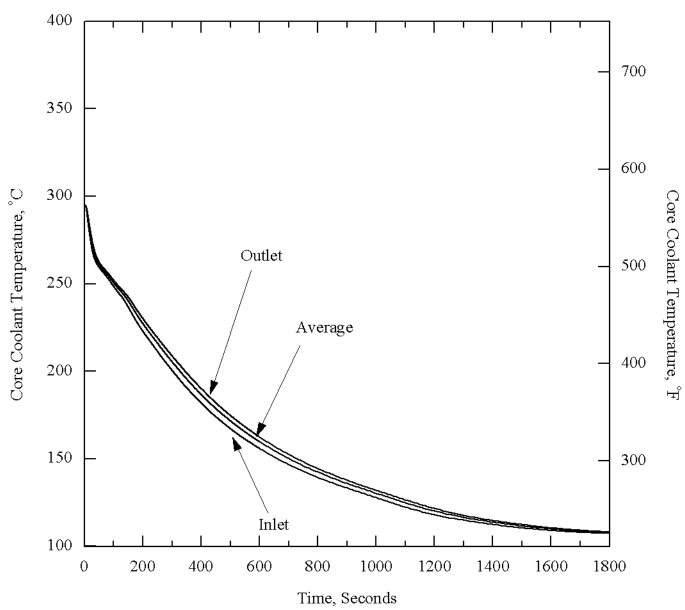

Figure 6. Zero Power Large Steam Line Break with Concurrent LOOP:
Core Coolant Temperature vs. Time

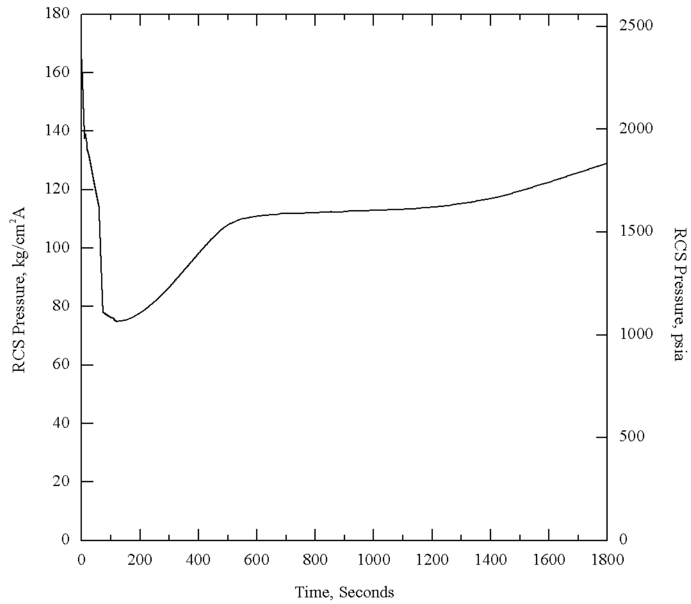

Figure 7. Zero Power Large Steam Line Break with Offsite Power Available: RCS Pressure vs. Time

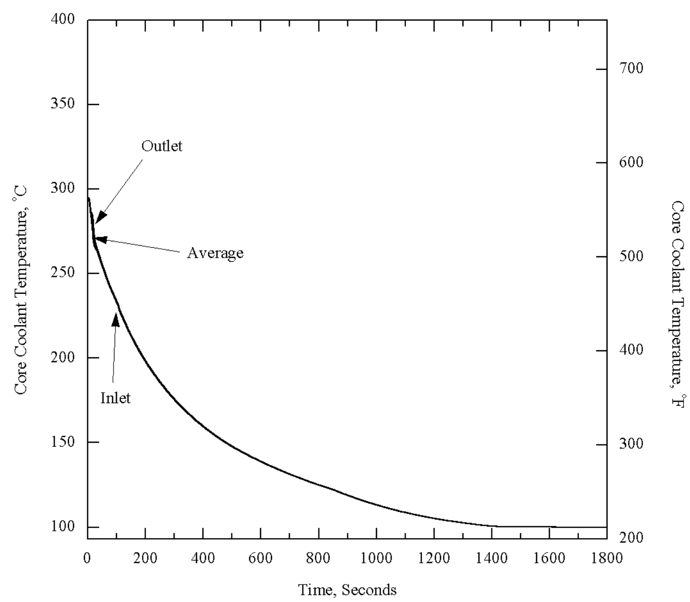

Figure 8. Zero Power Large Steam Line Break with Offsite Power Available: Core Coolant Temperature vs. Time

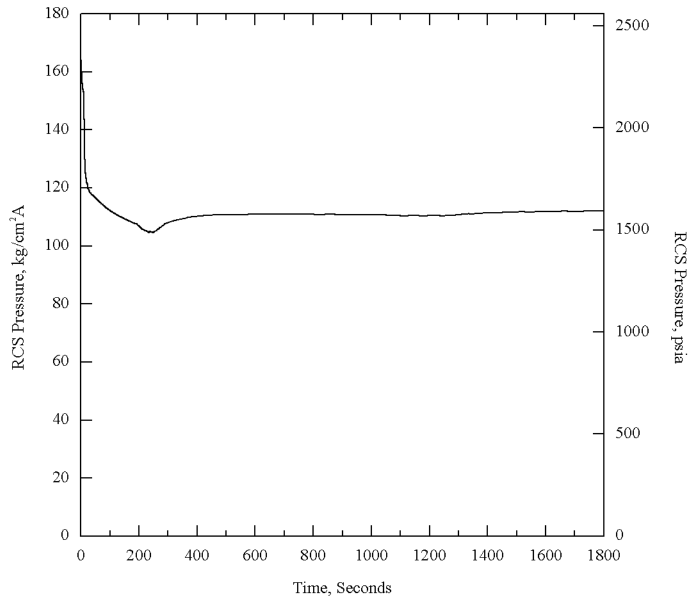

Figure 9. Full-Power Steam Line Break with LOOP: RCS Pressure vs. Time

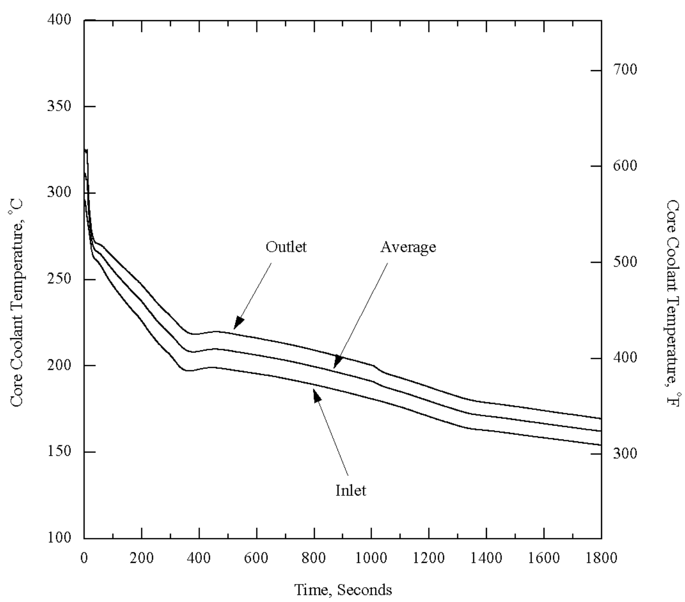

Figure 10. Full-Power Large Steam Line Break with LOOP:
Reactor Coolant Temperatures vs. Time

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

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