## KHNPDCDRAIsPEm Resource

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**Sent:** Tuesday, January 19, 2016 1:31 PM
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**Cc:** Drzewiecki, Timothy; McKirgan, John; Karas, Rebecca; Steckel, James; Lee, Samuel
**Subject:** APR1400 Design Certification Application RAI 370-8450 (15.06.03 - Radiological Consequences of Steam Generator Tube Failure (PWR) 07/1981)
**Attachments:** APR1400 DC RAI 370 SRSB 8450.pdf

KHNP,

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form. Your licensing review schedule assumes technically correct and complete responses within 30 days of receipt of RAIs. However, KHNP requests, and we grant, 60 days, 45 days, 60 days, and 45 days, respectively, to respond to these RAI questions. We may adjust the schedule accordingly.

Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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**Hearing Identifier:** KHNP_APR1400_DCD_RAI_Public
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United States Nuclear Regulatory Commission

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# REQUEST FOR ADDITIONAL INFORMATION 370-8450

Issue Date: 01/19/2016
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 15.06.03 - Radiological Consequences of Steam Generator Tube Failure (PWR) 07/1981
Application Section:

QUESTIONS

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.

15.06.03-2

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. NRC staff needs to ensure that a suitably conservative estimate is determined for the radiological release associated with the steam generator tube rupture event (SGTR).

Analysis of the radiological consequences for the SGTR event, presented in design control document (DCD) Section 15.6.3 credits operator actions, starting 30 minutes into the event, to (1) identify the affected steam generator, (2) confirm isolation, or isolate the affected steam generator, (3) establish cooldown of the reactor coolant system (RCS) via operation of the auxiliary feedwater system and the atmospheric dump valves of the unaffected steam generator, and (4) cool the RCS sufficiently to terminate break flow to the affected steam generator. Based on the description in the DCD, and the information contained in DCD Table 15.6.3-5, the analysis assumes that break flow is immediately terminated at 30 minutes with the initiation of operator action. However, as shown in Figures 15.6.3-19, 15.6.3-22, 15.6.3-23, and 15.6.3-29 of the DCD, flow through the break and the main steam safety valves of the affected steam generator remains significant 30 minutes into the event. Because the RCS pressure remains high and the break flow is significant at 30 minutes, NRC staff is unable to state that the analysis presents a bounding case in terms of radiological consequences. NRC staff requests the following:

1. Extend the CESEC-III analysis until break flow through the ruptured steam generator tube is terminated.

2. Update the text, figures, and tables in Section 15.6.3 of the DCD as appropriate. Be sure to include all actuations of engineered safety features (ESFs) along with their setpoints and credited operator actions in the sequence of events presented in Table 15.6.3-1 and Table 15.6.3-3 of the DCD.

3. Update the analysis of radiological consequence analysis as required.

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.

15.06.03-4

General Design Criteria (GDC) 14 requires that the reactor coolant boundary be designed, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The possibility of overfilling the steam generators during the steam generator tube rupture (SGTR) event could cause water to be introduced into the steam lines, which are serving as part of the pressure boundary during the SGTR event, producing loads beyond the design basis. Steam generator overfill is discussed in Section 15.6.3 of the Design Control Document (DCD) as part of the analysis that presents the limiting radiological consequences evaluation. NRC staff is questioning whether the limiting case in terms of radiological consequences is also limiting in terms of steam generator overfill. NRC staff requests KHNP either explain why the limiting radiological consequences case presents the limiting case for steam generator overfill, or update the DCD with a limiting steam generator overfill analysis.