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. 15.06.03-2

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.

Response

1. The analysis in DCD represents the bounding case in terms of radiological consequences during 30 minutes after event initiation. After 30 minutes, the operator initiates its orderly actions to mitigate the event based on the relevant emergency operating guidelines even though the action can be performed from the main control room (MCR) within 30 minutes. The analysis in DCD conservatively assumed that the break flow rate at 30 minutes is maintained until the time at which the primary and secondary pressures are same, or the break flow is terminated (additional times are considered based on the cooldown rate for the conservatism). In addition, during this transient, the flashed portion of reactor coolant is conservatively assumed as completely discharged to the environment.
Based on the above, the analysis presents bounding case in terms of radiological consequences, therefore, it is not practical with respect to the characteristic of safety analysis to extend the CESEC-III analysis until the break flow through the ruptured steam generator tube is terminated.

2. The sequence of events presented in Table 15.6.3-1 and Table 15.6.3-3 of the DCD provide the relevant actuations of engineered safety features (ESFs) along with their setpoints. However, each Table does not provide any credited operator actions, because the operator action is not credited until 30 minutes after event initiation event. Therefore, there is no need to update DCD.

3. There is no need to update the analysis of radiological consequences, because the analysis presents a bounding case in terms of radiological consequences.

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. 15.06.03-4

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.

Response

For the steam generator overfill event, the initial SG level may vary, however, the SG level has no effect on SG overfill, because the reactor trip by the high steam generator level trip (HSLT) signal would cause a concurrent turbine trip following the reactor trip and the SG mass inventory would be decreased by the MSSV opening and its opening cycles as shown in Figures 15.6.3-10 and 15.6.3-26.

For the radiological consequences, the earlier reactor trip causes the most release of contaminated steam to the environment through MSSV. Therefore, maximum initial steam generator level is assumed to trip the reactor at the beginning of the event with a HSLT signal. Based on the above, it could be said that the limiting radiological consequences case presents the limiting case with respect to steam generator overfill.

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.