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.:	**379-8476**
Review Section:	**07-19 – Branch Technical Position - Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems**
Application Section:	**7.8**
Date of RAI Issued:	**01/28/2016**

Question No. 07-1

The applicant provided a quantitative diversity and defense-in-depth (D3) analysis regarding the steam line break (SLB) outside containment event in Section 5.4.2.2 of APR1400-Z-A-NR-14019-P, Rev. 0, "CCF [common cause failure] Coping Analysis." In the calculation, the applicant determined that a small amount of fuel may experience DNB and is therefore be considered to fail. This maximum amount of fuel failure is used as the source term for the radiological analysis.

In accordance with Branch Technical Position (BTP) 7-19, "Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems," Section B, 1.4, Point 2, which is quoted from the SRM on SECY-93-087 (ML003708056), the applicant shall demonstrate adequate diversity in the design of the digital I&C safety systems to ensure that a SLB outside containment concurrent with a CCF does not challenge the plant's safety more than a SLB outside containment as analyzed in Chapter 15 of the DCD. Based on the applicant's current analysis, the staff is unable to confirm that the applicant's source term for the radiological analysis (i.e. the assumed amount of failed fuel for this event) is adequate; therefore, the staff cannot conclude the applicant is in compliance with BTP 7-19 Point 2.

The staff requests the applicant to provide a response demonstrating how the amount of fuel failing due to exceeding specified acceptable fuel design limits (SAFDLs) was determined for the SLB event.

Response

Section B, 1.4, Point 2 of BTP 7-19, states the applicant shall demonstrate adequate diversity within the design for each of the events that are evaluated in the accident analysis section of the safety analysis report. The common cause failure (CCF) coping analysis technical report (TeR) demonstrates the APR1400 design is capable of meeting the acceptance criteria required by Section B, 3.1 of BTP 7-19, that the plant response calculated using best-estimate (realistic assumptions) analyses does not result in radiation release exceeding 10 percent of the 10 CFR 100 guidance value or violation of the integrity of the primary coolant pressure boundary.

TS

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 the Technical/Topical/Environmental Reports.

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Question No. 07-2

The applicant provided a quantitative D3 analysis regarding the reactor coolant pump (RCP) rotor seizure event in Section 5.4.2.4 of APR1400-Z-A-NR-14019-P, Rev. 0, "CCF Coping Analysis." In this section of the technical report, the applicant stated that the minimum departure from nuclear boiling ratio (MDNBR) occured at 178.6 seconds into the transient. However, Figure 5-25, " of the technical report shows the MDNBR occurs right after the event initiation (i.e. <5 seconds into the event).

In accordance with BTP 7-19 Point 2, which is quoted from the SRM on SECY-93-087, the applicant shall demonstrate adequate diversity in the design of the digital I&C safety systems to ensure that an RCP rotor seizure concurrent with a CCF does not challenge the plant's safety more than an RCP rotor seizure as analyzed in Chapter 15 of the DCD. The staff is unable to confirm that the applicant's current analysis is adequate; therefore, the staff cannot conclude the applicant is in compliance with BTP 7-19 Point 2.

The staff requests the applicant to provide a response addressing this inconsistency.

Response

TS

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

Section 5.4.2.4.2.c. of APR1400-Z-A-NR-14019, Rev. 0, “CCF Coping Analysis” will be revised, as indicated in the attachment associated with this response.

reactor coolant pumps. Due to the loss of offsite power, power supply to CEDM is shut off. Then reactor trip breakers open at 5.55 seconds and a CEA is inserted. High pressurizer pressure trip (HPPT) signal by the DPS occurs at 5.65 seconds. The core maintains its full power before CEA drop; however, the DNBR is reduced by the reduction in reactor coolant flow. The DNBR degradation is terminated by the insertion of CEA and then increased suddenly. The minimum DNBR is [ ]TS and occurs at 7.23 seconds. Subsequent to this minimum value, the DNBR continuously increases and, after 30 minutes, the operator is assumed to take manual control of the plant under appropriate recovery procedures. Alarms and indications support the operators to trip the reactor.

5.4.2.3.3. Conclusions

The minimum DNBR was shown to remain well above the specified acceptable fuel design limit ensuring that no fuel failures occur. Also, the plant was shown to remain in a stable condition for at least 30 minutes ensuring that the operator has sufficient time to take manual control of the plant in order to execute a controlled cooldown.

5.4.2.4. RCP Rotor Seizure/Shaft Break

5.4.2.4.1. Events Overview

A single reactor coolant pump rotor seizure (rocked rotor) can be caused by seizure of the upper or lower thrust-journal bearings. A single reactor coolant pump shaft break (sheared shaft) could be caused by mechanical failure of the pump shaft. Since the sheared shaft event related flow coastdown is determined to be less adverse than that related to the rotor seizure event, it is judged that the sheared shaft event is less challenging to the event acceptance limits. Therefore, only single reactor coolant pump rotor seizure event is analyzed.

5.4.2.4.2. Analysis of Effects and Consequences

a. Mathematical Model

The NSSS thermal hydraulic response to a reactor coolant pump locked rotor with a postulated CCF in the PPS/ESF-CCS was simulated using the CESEC-III computer program. The minimum DNBR was calculated using the CETOP computer program which uses the KCE-1 CHF correlation.

b. Initial Conditions and Assumptions

Table 5-4 presents the input parameters and initial conditions used to analyze the NSSS and core thermal hydraulic responses to the reactor coolant pump locked rotor with a postulated CCF in the PPS/ESF-CCS.

c. Results

The time-dependent behavior of the important NSSS parameters following a reactor coolant pump locked rotor with a postulated CCF in the PPS/ESF-CCS is provided in Figures 5-20 through 5-26. If the one rotor locked out of 4 reactor coolant pumps, it will bring a rapid decrease in reactor coolant flow. The flow reduction terminates within a few seconds and stabilizes at a flow of approximately [ ]TS % of the rated flow. Low reactor coolant flow trip signal is assumed not to be generated due to the CCF in the PPS/ESF-CCS. The flow reduction results in degradation of DNBR in the early stage of the transient. Core power decreases due to the fuel temperature feedback effect within several seconds after event initiation and then restores to the initial core power due to the moderator temperature feedback effect. When the fuel and moderator temperature feedback effects lead to another balanced reactivity condition, a quasi-steady state is reached. The minimum DNBR of [ ]TS occurs at ~~178.6~~ 0.6 seconds. Subsequence to this minimum, the DNBR ~~slowly increases~~ rapidly increases and then slowly decreases until an essentially constant value is reached. After 30 minutes into the event, the operator is assumed to trip the reactor at which time the DNBR will again increase. The operator will