REVISED 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.:	**401-8402**
SRP Section:	**19.03 – Beyond Design Basis External Event (APR1400)**
Application Section:	**19.3**
Date of RAI Issue:	**02/08/2016**

Question No. 19.03-15

The staff conducted an Audit of Calculation 1-310-N380-008 Revision 0, "Containment Integrity Analysis Following RCP Seal Failure and Loss of RHR." This calculation discusses a single heat source, corresponding to a mass source, into containment from reactor coolant leakage flow (from reactor coolant pump seals). The staff requests that the applicant describe how sensible heat transfer from the reactor coolant system was evaluated. As part of the response, provide the value selected for sensible heat and any assumptions.

Response – (Rev.1)

As summarized in Table 5-1 of Technical Report APR1400-E-P-NR-14005-P, the plant is maintained at hot standby condition after the reactor trip. After 8 hours from event initiation, MSADVs can be supplied with power from the 480 V mobile GTG and the operator starts to cooldown the RCS using MSADV. Approximately 4 hours after RCS cooldown, the SCS entry condition is reached and the plant remains at the same SCS entry condition. Containment pressure and temperature analyses for Full-Power Operation were performed using mass and enthalpy data released from the RCS which includes the RCP seal leakage and normal system leakage. From the event initiation to 12 hours, mass release from the RCS is assumed to be 100 gpm and the enthalpy is conservatively determined under the assumption that the RCS remains at hot standby condition. After 12 hours, mass release from the RCS is assumed to critical flow at the SCS entry condition and the enthalpy is conservatively determined under the assumption that the RCS is remained at the same SCS entry condition.

The sensible heat transfer, which means the heat loss from the RCS to the containment, was additionally considered to analyze the containment pressure and temperature. The RCS heat loss of Shin-Kori Nuclear Power Plant Unit 3 which is the first of a kind plant for APR1400 reactor model was measured to be about 8.4 MWt at hot zero power condition.

During the BDBEE, however, the RCS heat loss decreases continuously as the containment temperature increases due to the RCP seal leakage and the sensible heat transfer. The RCS heat loss is a function of the differences between the RCS temperature and the containment temperature. When the temperature differences become zero (i.e., the containment temperature becomes the SCS entry condition), the RCS heat loss also becomes zero and the RCP seal leakage will be the only heat source.

The containment pressure and temperature analyses were re-performed considering the proportional relationship between the sensible heat transfer rate and the difference of RCS/containment temperatures. The results of re-analyses are submitted in the response to RAI 401-8402 question no. 19.03-16.

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. 19.03-19

NRC Commission paper SECY-12-0025 (February 17, 2012), "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," stated that the NRC staff expected new reactor design certification or license applications (e.g., construction permit, operating license, and combined license) not yet then-submitted to address the Commission-approved Fukushima actions in their applications, prior to submittal, to the fullest extent practicable. In SECY-12-0025, the NRC staff outlined a three-phase approach regarding mitigation strategies to respond to beyond-design basis external events (BDBEEs). The initial phase involved the use of installed equipment and resources to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling without alternating current power. The transition phase involved providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from offsite. The final phase involved obtaining sufficient offsite resources to sustain those functions indefinitely.

The NRC staff provided guidance for satisfying the Commission directives regarding BDBEE mitigation strategies in Japan Lesson-Learned Project Directorate (JLD)-ISG-2012-01, Revision 0, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," (ADAMS Accession No. ML12229A174). JLD-ISG-2012-01 endorsed with clarification the methodologies described in the industry guidance document Nuclear Energy Institute (NEI) 12–06, Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," (ADAMS Accession No. ML12242A378). The guidance in JLD-ISG-2012-01 describes one acceptable approach for satisfying the Commission directives regarding BDBEE mitigation strategies.

Technical Report APR1400-E-P-NR-14005-P does not describe the connection points (for FLEX equipment) necessary to maintain the containment capabilities. NEI 12-06 Section 3.2.1.3, "Initial Conditions," indicates that permanent plant equipment that is contained in structures with designs that are robust with respect to seismic events, floods, and high winds, and associated missiles, are available. The report should provide connection design information that justifies that a connection is robust and remains available to address a beyond-design-basis external event. The staff requests that the applicant provide information in the Technical Report that describes the connection design and connection quality classification used to maintain the containment capabilities and provide a basis for assuming that the connections will be available.

Response – (Rev. 1)

The connection line to the FLEX pump runs inside the containment building through the auxiliary building from the Siamese connection to the ECSBS nozzles, which is designed to Seismic Category I standpipe. The connection line is divided into two Quality Groups. The line from the Siamese connection to the ECSBS isolation valve (V1013) is designed with Quality Group D and the line from V1013 to ECSBS nozzles is designed with Quality Group B. Based on above discussion, Technical Report APR1400-E-P-NR-14005-P/NP, Subsection 5.1.2.5.2.3 will be revised as indicated in the Attachment to describe the connection design and connection quality classification.

As discussed in the revised response of RAI 407-8447 Question 19.03-26, the protection for the connection of FLEX pumps (including the connection for ECSBS FLEX pump) from applicable external hazards is the responsibility of the COL applicant. The identification of the connection for FLEX pumps and the COL item are described in the revised response to RAI 407-8447 Question 19.03-26.

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

Technical Report APR1400-E-P-NR-14005-P/NP, Subsection 5.1.2.5.2.3 will be revised as indicated in the Attachment.

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a. Normally closed motor-operated valve (MOV) (fail as-is)

b. Air-operated valve (AOV) (fail closed)

c. Check valve inside containment (automatic isolation)

5.1.2.5.2 Containment Capability during Full-Power Operation

The containment design incorporates a prestressed concrete containment with a steel liner to house the nuclear steam supply system. The containment and associated systems are designed to safely withstand environmental conditions that may be expected to occur during the life of the plant, including both short-term and long-term effects following a design basis accident (DBA) and beyond DBA.

During a BDBEE, no major pipe break is postulated inside the containment, but RCP seal leakage is assumed to be at a leak rate of 94.64 L/min (25 gpm) per RCP, a total of 378.5 L/min (100 gpm) for four RCPs. The containment pressure and temperature analyses are performed using the GOTHIC (Version 8.0) computer program. The containment pressure reaches the design pressure of 5.25 kg/cm^2 A (74.7 psia) in about 63 days from the beginning of the event. The design temperature of 143 °C (290 °F) is not exceeded until 71 days following the event. Figure 5-3 provides the containment pressure and temperature responses with the assumed RCP seal leakage. Therefore, containment integrity is maintained following full-power events through all phases.

5.1.2.5.3 Containment Capability during Mode 5 Operation

Loss of residual heat removal (RHR) during mid-loop operation in mode 5 is additionally assumed for the evaluation of containment capability. In the RCS mid-loop operation, SG nozzle dams are installed on the steam generator plena and the pressurizer manway remains opened. In this event, steam is assumed to be released from the RCS to the containment through the pressurizer manway due to the boiling of reactor coolant following the loss of RHR.

Due to the mass and energy released from the RCS, containment pressure increases consistently from the beginning of the event, but it can be maintained below UPC by operating the ECSBS intermittently after reaching UPC at around 83 hours. The ECSBS is assumed to start spraying water into the containment atmosphere via a FLEX pump when the containment pressure reaches the UPC value of 12.9 kg/cm^2 A (184 psia). After the initial operation, the ECSBS is assumed to be intermittently operated for 2 hours whenever the containment pressure reaches the UPC value. The FLEX pump provides the flow rate of 2,839 L/min (750 gpm) and the differential pressure of at least 2.8 kg/cm^2 (40 psi) at the ECSBS nozzle. The external water source for ECSBS operation is the RWT.

GOTHIC analyses are performed for evaluation of the containment pressure and temperature responses following loss of RHR in mode 5. Figure 5-4 shows that the containment pressure reaches the UPC value in about 3.5 days without ECSBS operation, but with the intermittent operation of ECSBS, containment pressure can be maintained within the UPC limit. Figure 5-5 shows that the containment temperature is maintained well below 185 °C (365 °F), which is less than the upper limit temperature of 196 °C (385 °F) for ensuring the operability of RCS sensors.

~~insert A in next page~~

5.1.2.6 Support Systems

insert B in next page

5.1.2.6.1 Electrical Systems

This subsection describes the electrical strategies to support the FLEX items described above for NTTF 4.1 and 4.2.

As stated earlier, the BDBEE causes the unit to lose all ac power. The initial condition is assumed to be

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A

~~The connection line to the FLEX pump runs inside the containment building from the Siamese connection to the ECSBS nozzles, which is designed to Seismic Category I standpipe for robustness and remains available in a BDBEE. The connection line is divided into two Quality Groups. The line from the Siamese connection to the ECSBS isolation valve (V1013) is designed with Quality Group D and the line from V1013 to ECSBS nozzles is designed with Quality Group B.~~

B

5.1.2.5.2.3 ECSBS operation for Containment Integrity

Connection line

The connection line to the FLEX pump runs inside the containment building through the auxiliary building from the Siamese connection to the ECSBS nozzles, which is designed to Seismic Category I standpipe. The connection line is divided into two Quality Groups. The line from the Siamese connection to the ECSBS isolation valve (V1013) is designed with Quality Group D and the line from V1013 to ECSBS nozzles is designed with Quality Group B.