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. 19.03-14

NEI 12-06 Section 3.2.1.5, “Reactor Coolant Inventory Loss,” identifies that normal system leakage is a source of expected reactor coolant inventory loss. Technical Report APR1400-EP-NR-14005-P, Table 5-9, “Conformance with NEI 12-06, Rev 0,” indicates conformance with NEI 12-06 Section 3.2.1.5. However, the Technical Report does not identify if normal reactor coolant inventory loss (Technical Specifications typically permit up to 11 gpm) is considered as contributing to the mass and energy input into the containment. Therefore, the staff requests that the applicant assess all potential sources into the containment to include normal system leakage and evaluate the impact on containment capabilities.

Response – (Rev. 1)

The normal identified leakage consists of pressurizer pilot-operated safety relief valves, reactor coolant pump seals, valves, reactor vessel head flange leakage, leakage through steam generator tubes or tubesheet and leakage to auxiliary systems. These are totally allowed up to 10 gpm but have no impact on containment capabilities because all the identified leakages are collected to RDT or VCT. The unidentified leakage is allowed up to 1 gpm in Technical Specification. Those leakages are, however, accounted for by assuming large RCP seal leakage (i.e., 100 gpm from 4 RCPs) in the support analysis for APR1400 FLEX strategy.

According to Technical Report APR1400-A-M-NR-14002-P, revision 1, the RCP seal leakage is under 0.1 gpm per RCP, which is much lower than the assumption, 25 gpm per RCP. Therefore, the total amount of the actual RCP seal leakage plus other leakage, 11 gpm permitted by Technical Specification is also much lower than the assumed seal leakage.

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, Rev.0, table 5-9 will be revised as indicated on the attachment.

Table 5-9 Conformance with NEI 12-06, Rev. 0 (7 of 20)

NEI 12-06, Rev. 0			APR1400
Section		**Summary**	
3.2.1.5	Reactor Coolant Inventory Loss	Sources of expected PWR reactor coolant inventory loss include: (1) normal system leakage (2) losses from letdown unless automatically isolated or until isolation is procedurally directed (3) losses due to reactor coolant pump seal leakage (rate is dependent on the RCP seal design) the controlled bleed-off Other normal system leakages, i.e., identified leakage of 10 gpm and unidentified leakage of 1 gpm, are allowed in APR1400 technical specification.	The APR1400 FLEX strategy complies with the guidance. During normal operation, there is ~~no system leakage except normal leakage~~ of 12.11 L/min (3.2 gpm) through each RCP, which is compensated by charging flow. RCP seal leakage might progress from ~~the normal leakage~~ of 12.11 L/min (3.2 gpm) per RCP to around 75.71 L/min (20 gpm) per RCP at 158.19 kg/cm^{2}A (2,250 psia) after 30 minutes. Normal letdown flow is 302.83 L/min (80 gpm), but letdown isolation valve is designed to close at setpoint of PZR low pressure. The letdown isolation valve could be also closed by operator action within 30 minutes following the event. In the support analysis for the APR1400 FLEX strategy, the seal leakage from each RCP is assumed to be 94.64 L/min (25 gpm) from the beginning of the event. Therefore, the assumption of seal leakage is conservatively determined to include all of system leakages considered above.
3.2.1.6	SFP Conditions	Initial conditions: (1) All boundaries of the SFP are intact, including the liner, gates, transfer canals, etc. (2) Although sloshing may occur during a seismic event, the initial loss of SFP inventory does not preclude access to the refueling deck around the pool. (3) SFP cooling system is intact, including attached piping. (4) SFP heat load assumes the maximum design basis heat load for the site.	The APR1400 FLEX strategy complies with the guidance.
3.2.1.7	Event Response Actions	Event response actions follow the command and control of the existing procedures and guidance based on the underlying symptoms that result from the event.	The APR1400 FLEX strategy complies with the guidance.

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Question No. 19.03-18

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.

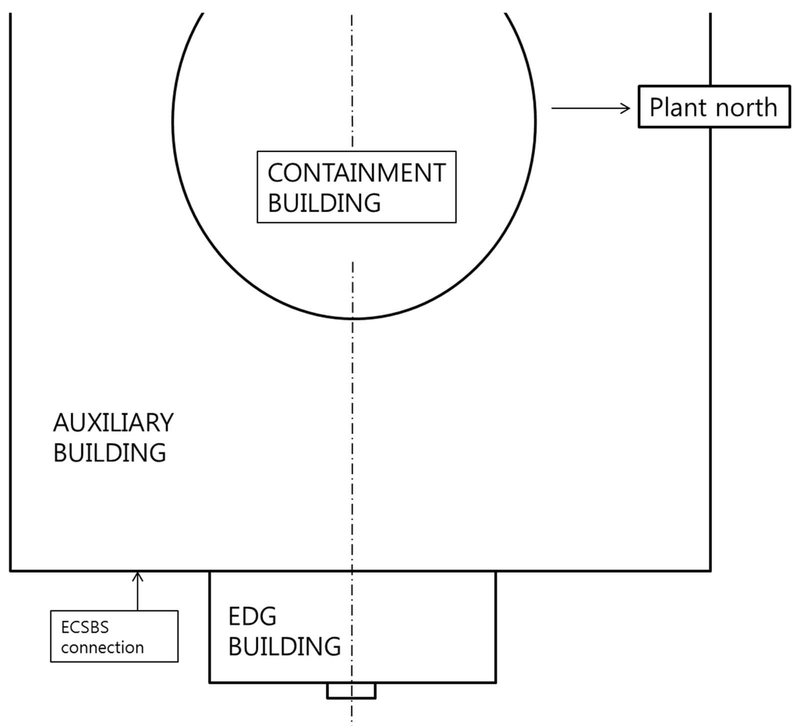
Technical Report APR1400-E-P-NR-14005-P does not contain simplified drawings to show how the FLEX strategy, using the emergency containment spray backup system (ECSBS), is used to maintain containment capabilities. The staff requests that the applicant provide a simplified drawing(s) that identifies the flow path to deliver water to containment. For example the drawing should depict plant piping, valves, pumps, water sources, power needs (as applicable), and any associated connections (FLEX pump suction, FLEX pump discharge, and fuel supply) and support systems. The staff also requests that the applicant provide the quality classification of installed structures, systems, and components used to maintain containment capabilities. Additionally, the location for any connections should be identified.

Response – (Rev. 1)

The ECSBS can use an external water source from the RWT using the ECSBS FLEX pump, which is connected to the ECSBS via a fire siamese connection located outside of the Auxiliary Building. Please refer to Figure 1 below for illustration. The ECSBS line runs inside the containment building from the fire siamese connection to the ECSBS nozzles.

The ECSBS FLEX pump is self-powered pump (diesel-driven). And the pump is connected to the RWT via a flexible hose on the suction side and is connected to the fire siamese connection via a flexible hose on the discharge side. Fuel is supplied from the diesel fuel oil day tank "A/B" as illustrated in Figure 6-6 of the Technical Report APR1400-E-P-NR-14005-P. The quality classifications of installed structures, systems, and components used to maintain containment capabilities (Containment Building, Auxiliary Building, and components of Containment Spray System) are described in DCD Tier 2 Table 3.2-1 (Page 1, 12, and 13 of 86) and Figure 6.2.2-1. A new schematic drawing (Figure 6-7) will be added to the next page of Figure 6-6 and Subsection 5.1.2.5.3 will be revised as indicated in the Attachment.

Figure 1 –Locations for ECSBS connection



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 5.1.2.5.3 will be revised and Figure 6-7 will be added as indicated in the Attachment.

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.

A simplified drawing that identifies the flow path to deliver water to the ECSBS is schematically shown in Figure 6-7.

5.1.2.6 Support Systems

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

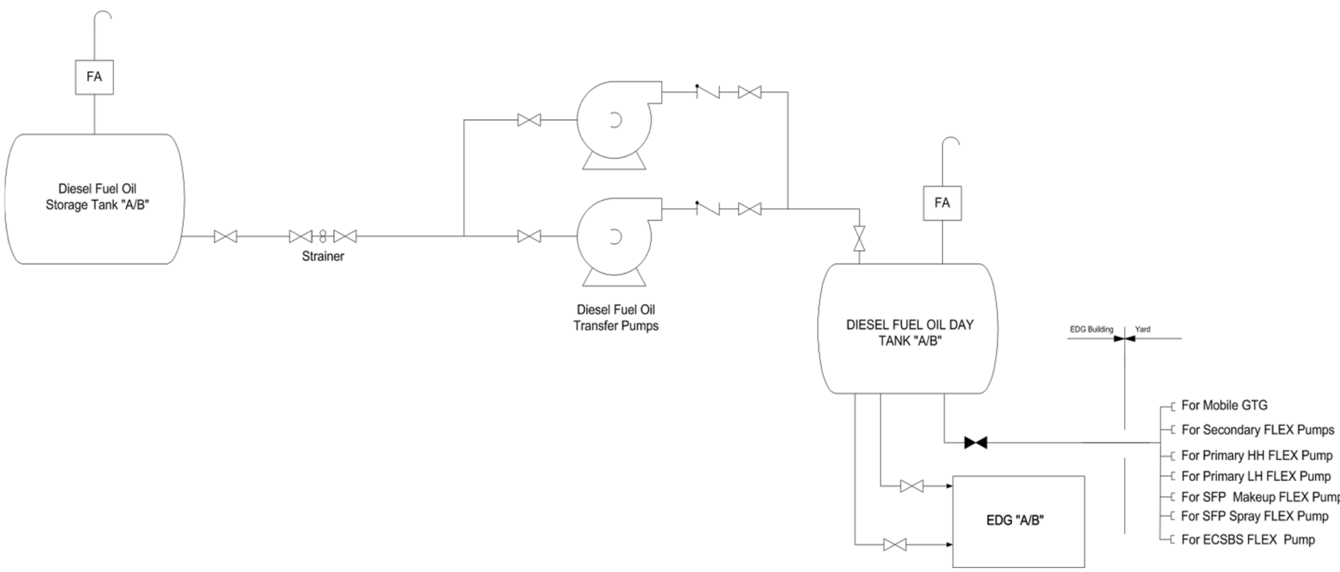


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Figure 6-6 Fuel Oil Supply System to FLEX Pumps

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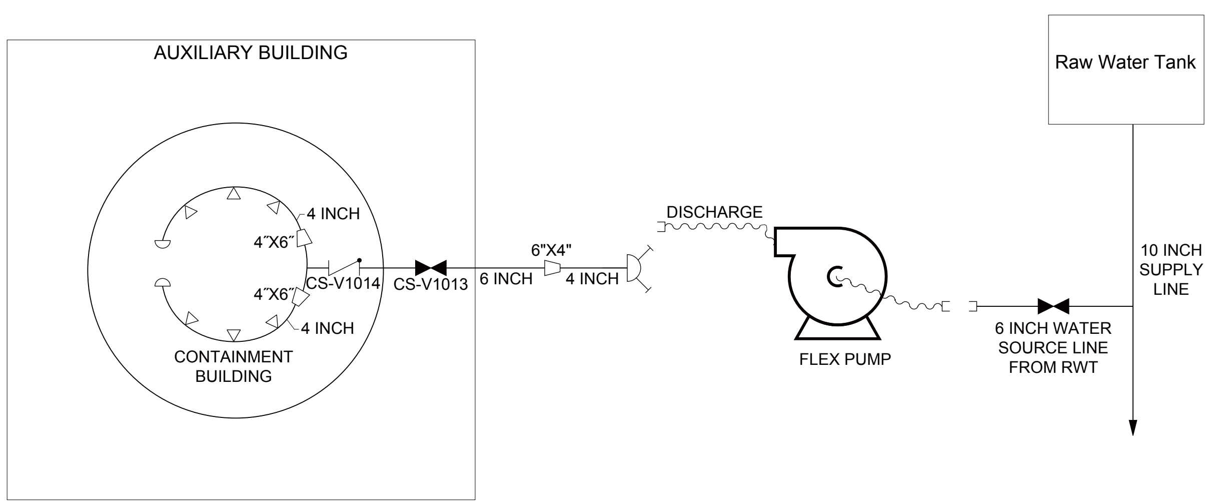

Figure 6-7 Flow Path for FLEX Connection to Deliver Water to Containment for ECSBS