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.:	**407-8447**
SRP Section:	**19.03 – Beyond Design Basis External Event (APR1400)**
Application Section:	**19.03**
Date of RAI Issue:	**02/17/2016**

Question No. 19.03-25

DCD Tier 2, Section 19.3.2.3 and the technical report (TR), APR1400-E-P-NR-14005-P, "Evaluations and Design Enhancements to Incorporate Lessons Learned from Fukushima Dai-Ichi Nuclear Accident," describe the details of the proposed mitigation strategies. The applicant performed analyses to demonstrate the capability of the proposed mitigation strategies for core and SFP cooling, and containment function. The proposed acceptance criteria are as follows:

· Core Cooling – Appendix A, Section A.3 of the TR states that the acceptance criteria for core cooling are (1) core cooling being maintained, (2) no fuel failures.

· Spent Fuel Pool Cooling – Appendix B, Section B.1 of the TR states that the acceptance criterion for SFP cooling is that fuel in the SFP remains water covered.

· Containment Function – DCD Tier 2, Section 19.3.2.3.3 states that the containment pressure is controlled within the ultimate pressure capacity (UPC) limit.

NEI 12-06, Section 3.2.1.1 states that, for core cooling in a PWR, the requirement is to keep the fuel in the reactor covered. In Table 5-9 Item 3.2.1.1 of the TR, the applicant indicates that APR1400 complies with NEI guidance. However, the staff found that the applicant’s criteria for core cooling are inconsistent with the NEI guidance regarding keeping the fuel covered.

The applicant is requested to:

a) Explain the inconsistency and justify the deviation in regards to the criteria for maintaining core cooling.

b) Confirm the acceptance criterion for the containment function.

The NRC staff believes that the above acceptance criteria for the mitigation strategies are part of the licensing bases for APR1400 design and should be documented in the DCD instead of being captured in a technical report.

The applicant is requested to include the acceptance criteria for mitigation strategies in the DCD.

Response

a) Figures A-7, A-17 and A-27 of Technical Report APR1400-E-P-NR-14005-P, Rev.0 show the collapsed downcomer and core level for full-power operation and shutdown operation with SGs not available. As shown in those figures, for a short time periods the ‘collapsed’ level is below the active core top. But it doesn’t necessarily mean the core uncovery in this two phase flow condition. Figure A-26 shows that enough liquid fractions at the core top is maintained during the shutdown operation with SGs not available and Figure A-28 shows that the core cooling is sufficiently maintained even if the collapsed core level is below the active core top for a short time periods. In case of the full-power operation, deviation between the active core top and the collapsed level is less than the deviation in case of the shutdown operation with SGs not available. It means that there is liquid in the region of core top.

   In addition, “no fuel failures” in technical report means to maintain the coolable geometry and the fuel cladding temperature below 1204℃ (2200℉). The criterion “keep fuel in the reactor covered” in NEI 12-06 is also thought to be set up to make sure the core cooling and coolable geometry.

   Therefore the criterion, “no fuel failures”, in the technical report is equivalent to the criterion, “keep fuel in the reactor covered”, in NEI 12-06, Rev.0 in regards to the criteria for maintaining core cooling.

b) The requirements with regards to the containment integrity described in the SECY 11-0093 is that the licensee provides reasonable protection from beyond design basis external events (BDBEEs) and add additional equipment necessary to mitigate events that are similar to those of the Fukushima Dai-Ichi. Based on the requirement, the UPC value (158 psia) is chosen as the acceptance criteria upper limit to ensure the containment integrity and the emergency containment spray backup system (ECSBS) is used to maintain the containment pressure lower than the UPC limit against BDBEEs.

The acceptance criteria for mitigation strategies will be documented in Section 19.3.2.3.1 of DCD Tier 2.

The DCD will be revised to include the acceptance criteria regarding the containment function to the BDBEEs.

Impact on DCD

DCD Tier 2, section 19.3.2.3.1 and 19.3.2.3.3 will be revised as indicated on the attachment markup.

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.

The guidance for developing, implementing, and maintaining mitigation strategies from JLD-ISG-2012-01 (Reference 6) and the methodology to establish baseline coping capability from Nuclear Energy Institute (NEI) 12-06 (Reference 7) were considered in developing the APR1400 FLEX strategy. Each FLEX strategy follows a three-phase approach as required in the Order EA-12-049.

The three phases are:

a. Phase 1 – Initial response phase using installed equipment

b. Phase 2 – Transition phase using portable equipment and consumables

c. Phase 3 – Indefinite sustainment of these functions using offsite resources

19.3.2.3.1 Core Cooling

The following acceptance criteria are applied to core cooling during the ELAP concurrent with LUHS.
a. Core cooling is maintained
b. No fuel failures

The APR1400 FLEX strategy can be divided into two sets of operational strategies, as follows:

a. FLEX strategy for Modes 1 through 4 (full-power operation, startup, hot standby, hot shutdown) and Mode 5 operation (cold shutdown) with steam generators (SGs) available

b. FLEX strategy for Modes 5 and 6 operations with SGs not available

Supporting analysis is performed to demonstrate the APR1400 baseline coping capability based on both of the FLEX strategies. In the support analysis, the full-power operation case is selected as a representative one for the operational strategy for the Modes 1 through 5 with SGs available. Mid-loop operation case is selected as a representative one for the operational strategy for Mode 5 and 6 with SGs not available.

The initiating event is assumed to be a loss of offsite power (LOOP) with concurrent loss of all ac power and LUHS during the full-power operation or mid-loop operation. Based on the analysis performed, the APR1400 will consider the three-phase approach as shown in Table 19.3-1 to address FLEX strategies for the various plant operations, namely, full-power operation, low-power, and shutdown operations, with and without SGs available.

BDBEE are designed to be isolated by either inside containment or outside containment isolation valves, as follows:

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

b. Air-operated valve (fail closed)

c. Check valve inside containment (automatic isolation)

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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 DBA and beyond DBA. No special means are necessary for the APR1400 to maintain containment function during full-power operation, after a BDBEE with simultaneous loss of all ac power and LUHS. The emergency containment spray backup system (ECSBS) is used to maintain containment pressure and temperature during loss of RHR (Mode 5).

During the BDBEE, no major pipe break is postulated inside the containment, but RCP seal leakage is assumed with the leakage rate of 94.64 L/min (25 gpm) per RCP, a total of 378.5 m^3/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 (74.7 psia) in 63 days from beginning of the event. The design temperature of 143 °C (290 °F) is reached in 71 days following the event. The technical report (Reference 5) provides the containment pressure and temperature analyses response for the full-power case with the assumed RCP seal leakage, and confirms that, during the course of the event for all phases, containment integrity is maintained.

Loss of RHR during mid-loop operation in Mode 5 is additionally assumed for the evaluation of containment capability. 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. 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/cm2 (184 psia)~~11.11 kg/cm^2 (158 psi). After the initial operation, the ECSBS is assumed to be intermittently operated for 2 hours whenever the containment pressure reaches the UPC value. GOTHIC analyses are performed to confirm that the R_129-8085(5S)

BDBEE are designed to be isolated by either inside containment or outside containment isolation valves, as follows:

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

b. Air-operated valve (fail closed)

A

c. Check valve inside containment (automatic isolation)

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 DBA and beyond DBA.

The requirements regarding containment integrity described in Recommendation 4.2 in the SECY 11-0093 (Reference 11) is that the licensee provides reasonable protection from beyond design basis external events (BDBEEs) and add additional equipment necessary to mitigate events that are similar to those of the Fukushima Dai-Ichi.

The UPC value (158 psi) is chosen as the acceptance criteria upper limit to ensure the containment integrity. The emergency containment spray backup system (ECSBS) is used to maintain the containment pressure lower than the UPC limit against BDBEEs.

The containment analyses using the GOTHIC(Version 8.0) computer program are performed to estimate the containment pressure and temperature responses to the BDBEEs.

As a BDBEE at full-power operation, the RCP seal leakage is assumed with a leakage rate of 94.64 L/min (25 gpm) per RCP, a total of 378.5 m^3/min (100 gpm) for four RCPs. The containment pressure reaches the design pressure of 4.22 kg/cm^2 (60 psi) in 63 days from beginning of the event. The design temperature of 143 °C (290 °F) is reached in 71 days following the event. Consequently, it is demonstrated that the containment system has enough time to cope with the BDBEE with simultaneous loss of all ac power and LUHS at full-power operation even without special means such as the ECSBS operation.

The technical report (Reference 5) provides the containment pressure and temperature analyses response for the full-power case with the assumed RCP seal leakage, and confirms that, during the course of the event for all phases, containment integrity is maintained.

Loss of RHR during mid-loop operation in Mode 5 is additionally assumed for the evaluation of containment capability. 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. 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/cm2 (184 psia)~~11.11 kg/cm^2 (158 psi). After the initial operation, the ECSBS is assumed to be intermittently operated for 2 hours whenever the containment pressure reaches the UPC value. GOTHIC analyses are performed to confirm that the R_129-8085(5S)

2. Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," U.S. Nuclear Regulatory Commission, March 12, 2012.

3. Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," U.S. Nuclear Regulatory Commission, March 12, 2012.

4. "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," March 12, 2012.

5. APR1400-E-P-NR-14005-P, "Evaluations and Design Enhancements to Incorporate Lessons Learned from the Fukushima Dai-Ichi Nuclear Accident," Rev. 0, KHNP, December 2014.

6. JLD-ISG-2012-01 "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," Rev. 0, U. S. Nuclear Regulatory Commission, August 29, 2012.

7. NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Rev. 0, Nuclear Energy Institute, August 2012.

8. NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, 'To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,'" Rev. 1, Nuclear Energy Institute, August 2012.

9. JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," Rev. 0, U.S. Nuclear Regulatory Commission, August, 2012.

10 NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communication Capabilities," Rev. 0, Nuclear Energy Institute, May 2012.

Insert

11 SECY-11-0093, "Recommendations for Enhancing Reactor Safety in the 21st Century," U.S. Nuclear Regulatory Commission, July 2011.

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Application Section:	**19.03**
Date of RAI Issue:	**02/17/2016**

Question No. 19.03-26

Connections:

10 CFR 52.47(a)(2) requires that a standard design certification applicant provide a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished.

In SECY 12-0025, the staff provided the Commission with proposed orders requiring mitigation strategies for beyond-design-basis external events to be issued to all power reactor licensees and holders of construction permits. In the paper, the staff indicated that for New Reactors that are currently under active staff review, the staff plans to ensure that the Commission-approved Fukushima recommended actions are addressed prior to licensing. On March 12, 2012, the NRC issued Orders EA-12-049 requiring operating nuclear plants to develop and implement strategies that will allow them to cope without ac power for an indefinite amount of time. The strategies must ensure that the reactor core and spent fuel pool are adequately cooled, and containment function is maintained.

NEI 12-06, Section 3.2.2 states that the portable fluid connections for core and SFP cooling functions are expected to have a primary and an alternate connection. Both the primary and alternate connection points do not need to be available for all applicable hazards, but the location of the connection points should provide reasonable assurance of at least one connection being available.

The staff reviewed the information in APR1400-E-P-NR-14005-P, "Evaluations and Design Enhancements to Incorporate Lessons Learned from Fukushima Dai-Ichi Nuclear Accident," Section 5.1.2.4.1.2, comparing it with the NEI guidance. It is not clear whether the connections being used in the proposed mitigation strategies for SFP cooling are consistent with the guidance that the location of the connection points should provide reasonable assurance of at least one connection being available for all applicable external hazards.

The applicant is requested to clarify how the APR1400 design for connections to the FLEX equipment is consistent with the NEI guidance.

Response

The APR1400 mitigation strategy for SFP cooling is schematically illustrated in Figures 6-2 and 6-3, and the approximate locations for makeup water connections are indicated in Figure 6-4. The APR1400 design provides two locations for SFP makeup water connections: a set of two primary connections (one for makeup and one for spray) is mounted on the outside wall of the Auxiliary Building (AB), adjacent to the south side of the Emergency Diesel Generator (EDG) building; and the alternate set of identical connections (one for makeup and one for spray) is mounted on the outside wall of AB, adjacent to the north side of the EDG building. Please refer to Figure 1 below for illustration.

Since the south and north connections are physically separated by the EDG building which is a seismic Category I reinforced concrete structure and also designed to withstand the effects of internal and external hazards. This configuration provides reasonable assurance of at least one connection being available; and is consistent with the guidance in Section 3.2.2 of NEI 12-06.

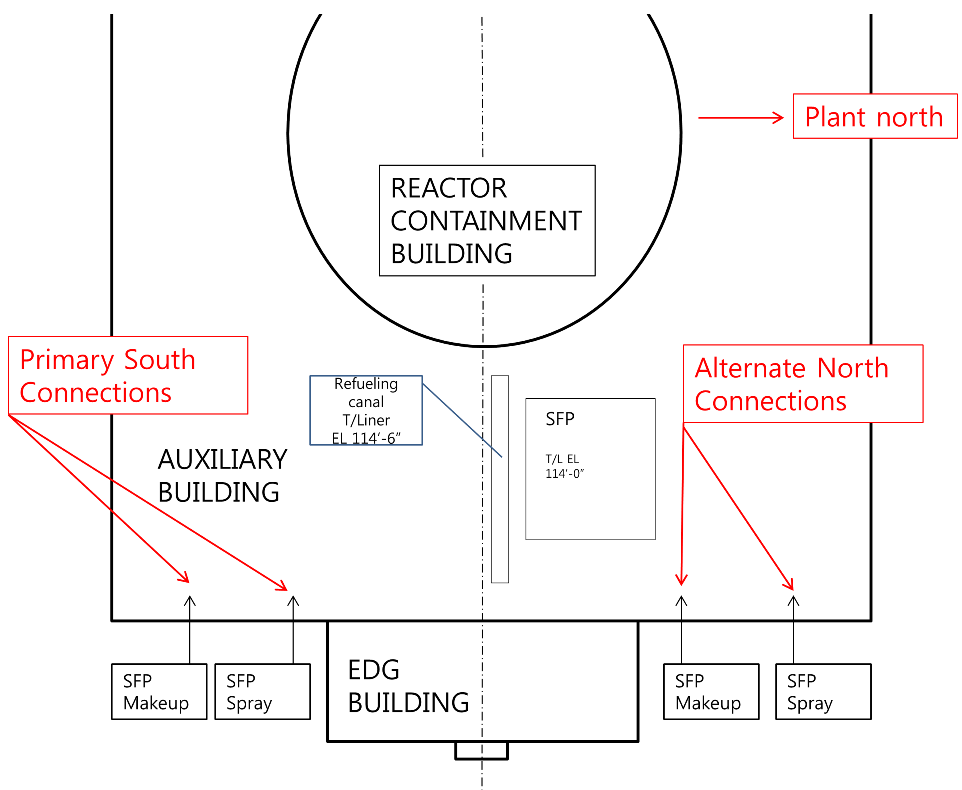

Figure 1 Locations for SFP FLEX pump connection

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

There is no impact on any Technical, Topical, or Environment Report.

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Question No. 19.03-28

Makeup Water:

10 CFR 52.47(a)(2) requires that a standard design certification applicant provide a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished.

In SECY 12-0025, the staff provided the Commission with proposed orders requiring mitigation strategies for beyond-design-basis external events to be issued to all power reactor licensees and holders of construction permits. In the paper, the staff indicated that for New Reactors that are currently under active staff review, the staff plans to ensure that the Commission-approved Fukushima recommended actions are addressed prior to licensing. On March 12, 2012, the NRC issued Orders EA-12-049 requiring operating nuclear plants to develop and implement strategies that will allow them to cope without ac power for an indefinite amount of time. The strategies must ensure that the reactor core and spent fuel pool are adequately cooled, and containment function is maintained.

Table B-3 of APR1400-E-P-NR-14005-P provides information on the required makeup volume and available water source. The NRC staff reviewed the table and found some error. In the table, Column 2 is for Mode 1 to 6 (with no full core offload) and Column 3 is for Mode 6 (with no full core offload). Total coping time for Modes 5 and 6 is 6.4 days in Column 2, and the total coping time for Mode 6 is 15.1 days in Column 3.

The applicant is requested to explain the differences between these two columns (6.4 days vs. 15.1 days) for Mode 6 both with no full core offload or correct the error, if any.

Response

The last column on Table B-3, top cell has an editorial error: (with no full core offload) should be replaced with (with full core offload). Table B-3 will be corrected as indicated in the Attachment.

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, Appendix B, Table B-3 will be corrected as indicated in the Attachment.

Table B-3

Required Makeup Volume and Water Source

Delete "no"

Parameter / MODE		Mode 1~6 (with no full core offload)	Mode 6 (with ~~no~~ full core offload)
Water source Total volume, m^3 (gal)		RWT 9,993.49 (2,640,000)	RWT 9,993.49 (2,640,000)
Available volume for SFP makeup, m^3 (gal)		Mode 1 to 4[Note 1]: 2,793 (737,851); Mode 5 and 6[Note 2]: 1,124 (296,800)	9,993.49 (2,640,000)
Makeup during 72 hours	Time for makeup (72 hours minus time to 3.05 m [10 ft] above fuel top), (hours)	8.32	46.97
	Required makeup volume, m^3 (gal)	101.46 (26,827)	1,390 (367,214)
Makeup during 12 days (288 hours)	Time for makeup (12 days minus time to 3.05 m [10 ft] above fuel top), (hours)	224.32	262.97
	Required makeup volume, m^3 (gal)	2,739(723,538)	7,783(2,056,041)
Total copying Time		Modes 1 to 4: 12.2 days Modes 5 and 6: 6.4 days	Mode 6: 15.1 days

(Note 1): RWT can be used as the water source for NCC operation through TDAFWP and SFP makeup.

(Note 2): RWT can be used as the water source for RCS makeup through primary low-head FLEX pump, SFP makeup, and ECSBS operation.

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Question No. 19.03-29

The applicant states in DCD Tier 2, Section 19.3.2.3.4, that the design approach meets the NEI 12-06 guidance of the N+1 approach for the FLEX equipment. It is not clear to the staff how many primary low-head FLEX pumps, which are used for core cooling in low modes Phase 2 operation (see APR1400-E-P-NR-14005-P Table 5-6), are in the design to satisfy the N+1 guidance.

The applicant is requested to clarify how many primary low-head pumps are proposed in the design.

Response

There are two primary low-head FLEX pumps designed for APR1400. One primary low-head FLEX pump is required for core cooling in phase 2 of “FLEX strategy for shutdown operation with SGs not available”. An additional pump is kept in reserve to satisfy the N+1 guidance.

Section 19.3.2.3.1.2 of DCD Tier 2 and Section 5.1.2.3.3.2 of Technical Report APR1400-E-P-NR-14005-P, Rev.0 will be revised to clarify the number of FLEX pumps in accordance with the N+1 guidance by adding the following statement:

“Two primary low-head FLEX pumps are provided to meet the N+1 requirement.”

Impact on DCD

DCD Tier 2, section 19.3.2.3.1.2 will be revised as indicated on the attachment.

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 APR 1400-E-P-NR-14005, Rev.0, section 5.1.2.3.3.2 will be revised as indicated on the attachment.

Based on the analysis performed, the APR1400 will consider the following event sequence to address FLEX strategy for the shutdown operations with SGs not available:

a. Phase 1: 0 to 3 hours

b. Phase 2: 3 to 72 hours

c. Phase 3: indefinite time period following the phase 2

During Phase 1, decay heat is removed as latent heat that developed during the water boil-off in the core. At the same time, the water source for gravity feed from the SITs is utilized to prevent core uncovery. Since the operator can easily identify the initiation of loss of residual heat removal (RHR), the operator can promptly initiate the necessary recovery action for keeping the core covered: manually opening the valves needed for gravity feed. Then, the operator prepares for the next phase. A primary low-head FLEX pump is connected to the SIS injection line. A mobile GTG is connected to Train A or Train B 480 V Class 1E ac power system. All of the operator actions will be finished within 3 hours following the event.

During Phase 2, the RCS inventory makeup is carried out by external injection using the primary side low-head FLEX pump, with a rated flow of 2,839.06 L/min (750 gpm), which is sufficient capacity for removing decay heat.

Two primary low-head FLEX pumps are provided to meet the N+1 requirement.

In Phase 3, the 4.16 kV mobile GTG, fuel, and cooling water are available for long-term coping for the event. The 4.16 kV mobile GTG will be used to restore Train A or Train B of the 4.16 kV Class 1E power system. If the SCS is operable when the 4.16 kV Class 1E power is restored, the plant will be cooled down or maintained by resuming the SCS operation. If not, the RCS inventory is maintained by the primary FLEX pump, as in Phase 2. In this case, the primary makeup water source and fuel oil for the mobile GTGs will be refilled from offsite resources.

19.3.2.3.2 Spent Fuel Pool Cooling

Based on the supporting analyses described in Reference 5, the following is the bulk SFP heatup time and boil-off rate for the worst-case full core offload:

5.1.2.3.3 FLEX Strategy for Shutdown Operation with SGs Not Available

The APR1400 shutdown operations with SGs not available include the mode 5 reduced inventory operation and the mode 6 refueling operation. If the ELAP concurrent with LUHS occurs during the reduced inventory operation or refueling, decay heat can be removed from the core by the RCS feed-and-bleed operation.

In developing the APR1400 baseline coping capability during shutdown operations with SGs not available, the mid-loop operation case is selected as a representative one, because this operation mode has the lowest RCS inventory and requires the earliest operator action for the feed-and-bleed operation.

Based on the analysis performed, the APR1400 design includes consideration of the following event sequence to address FLEX strategy for shutdown operations with SGs not available:

Phase 1: 0 to 3 hours

Phase 2: 3 to 72 hours

Phase 3: Indefinite time period following Phase 2

The timeline of the APR1400 FLEX strategy for the shutdown operations with SGs not available is shown in Figure 5-2 and the detailed sequence of events is tabulated in Table 5-3. The following are the operational strategies for each phase.

5.1.2.3.3.1 Phase 1: Coping with Installed Plant Equipment (0 to 3 hours)

During Phase 1, decay heat is removed by the latent heat resulting from water boiloff in the core. At the same time, the SITs are used as a water source for gravity feed to the RCS. Since the operator can easily identify the initiation of loss of residual heat removal, the necessary recovery action of manually opening the valves needed for gravity feed from SITs can promptly begin and the core remains covered. Then, the operator connects a primary low-head FLEX pump to the SIS injection line for preparation of the feed-and-bleed operation in Phase 2. The operator actions are finished by 3 hours following the event. The operator has a 1-hour margin for preparation of Phase 2, because the analysis result shows that the Phase 1 gravity feed and boiling operation can last for 4 hours.

5.1.2.3.3.2 Phase 2: Coping with Installed Plant Equipment and Onsite Portable Resources (3 to 72 hours)

In Phase 2, the plant can be maintained at cold shutdown by the RCS feed-and-bleed operation using the FLEX pump. The RCS inventory makeup is carried out by external injection using the primary side low-head FLEX pump with rated flow of 2,839.06 L/min (750 gpm), which is sufficient capacity for removing decay heat. Decay heat is removed by boiloff from the core, while the steam generated from the core is released through the pressurizer manway. The low-head FLEX pump takes suction from the raw water tank (RWT), and the rate of injection flow is controlled to maintain the RCS water level between the core top and the hot leg center line. In this feed-and-bleed operation, the RCS is maintained at the initial boron concentration, because the rate of unborated water injection is well balanced with the rate of steam discharge. In the meantime, a mobile GTG is connected to Train A or Train B 480 V Class 1E ac power system within 8 hours to supply power to Class 1E battery.

Two primary low-head FLEX pumps are provided to meet the N+1 requirement.

The Phase 2 feed-and-bleed operation using onsite water source is assumed to last for 72 hours in the timeline of the mid-loop operation FLEX strategy, but the capacity of the RWT is sufficient to extend the period of Phase 2 up to 6.4 days even if the water source is shared with SFP cooling (see Table B-3 in