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.:	**409-8325**
SRP Section:	**SRP 19**
Application Section:	**19.1**
Date of RAI Issued:	**02/22/2016**

Question No. 19-20

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. SRP Chapter 19, Revision 3 (Draft), "Design-Specific PRA (PRA for Non-Power Modes of Operation)" states that, "Given that shutdown risk may be highly outage-specific, the staff reviews the shutdown PRA insights to confirm that operational assumptions used to develop an average shutdown model (e.g., use of nozzle dams, outage schedule, containment status, procedural requirements) have been clearly documented in the FSAR." The APR 1400 DCD provides no discussion on the risk of boron dilution events. In an example from NUREG-1449, (which is discussed in the Shutdown Evaluation Report), a loss of offsite power (LOOP) has occurred and the charging pumps are returned online, powered by the emergency diesel generators (EDG). If the plant is in startup mode (i.e., deboration in progress), the charging pumps could continue to operate, causing a "slug" of unborated water to collect in the lower plenum of the reactor vessel (RV). If it is then assumed that offsite power is restored and the reactor coolant pumps (RCPs) are restarted, then a water slug of deborated water can be injected into the core. The staff has the following questions and requests for clarification:

a. In the APR 1400 design, the staff understands the charging pumps are not automatically loaded on the EDGs. The operator must manually reload the charging pumps onto the EDGs and restart the pumps for deboration to continue. The staff is requesting this clarification to be added to Section 19.1.6 of the design control document (DCD).

b. The staff is requesting the applicant to add in Section 19.1.6 of the DCD the procedure or guidance that prevents the operator from restarting the charging chemical and volume control system (CVCS) pumps and thus preventing reactor coolant system (RCS) deboration from continuing.

c. The staff is requesting a justification to be added in Section 19.1.6 of the DCD as to why boron dilution events were screened from the low-power shutdown (LPSD) PRA. If operator actions are important in screening the risk of boron dilution events from the

PRA, the staff is requesting that these operator actions be added to the risk insights Table 19.1-4 or provide instead a justification as to why this addition to the risk insights table is not necessary. In addition, please consider whether a COL item should be added to section 19.1.6 of the DCD.

Response

The responses are as follows:

a. This clarification will be included in DCD Section 19.1.3.1.f as shown in Attachment 1.

b. This clarification will be included as an additional COL item in DCD Section 19.1.9, and the associated COL Table 1.8-2 will be revised as shown in Attachment 2.

c. The basis for screening boron dilution events from the LPSD PRA is as follows. DCD Section 15.4.6 provides the results of the analysis for uncontrolled deboration events. It indicates that the minimum time available for operators to stop the deboration is 72.8 minutes under the most adverse conditions such as minimum RCS inventory, maximum charging flowrate, maintaining subcriticality only by boron concentration, etc. DCD Section 15.4.6 also states that when an uncontrolled deboration occurs, operators are alerted through a high neutron flux alarm on the startup flux channel, the reactor makeup water flow alarm (in Mode 6 only), sampling results, boronometer indications or the boric acid flow rate. The operators will secure the charging pump in order to halt further dilution. Next, the operator increases the RCS boron concentration by initiating the emergency boration procedure to restore shutdown margin. Given the conservative analysis assumptions, the diverse alarms available to the operators, the minimum time available for operator response before recriticality, and the simple, proceduralized actions to terminate the events, boron dilution events are screened out on the basis of engineering judgment.

   It will be added in the Risk Insights Table (Table 19.1-4) to provide this basis as shown in Attachment 3.

   No COL addition is required, as the shutdown emergency procedures [WE SHOULD LIST THE PROCEDURE REFERENCE. IF THERE IS NONE, THEN THAT WOULD WARRANT A COL ADDITION] already reflect these alarms for event identification, and these recovery actions.

Impact on DCD

The DCD will be revised as shown in Attachments 1, 2 and 3.

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 Environmental Report.

The CVCS provides four key functions for accident mitigation in the unlikely event of an accident. The first accident mitigation function is to support the auxiliary pressurizer spray function. This function is accomplished by the centrifugal charging pumps drawing suction from either the volume control tank (VCT) or boric acid storage tank (BAST) and discharging to the pressurizer spray nozzle via the auxiliary spray line. Successful delivery of the BAST contents to the charging pump suction is accomplished either via the boric acid makeup pumps (BAMPs) or via gravity drain.

The second accident mitigation function is the emergency boration that provides an independent means of supplying borated water to the RCS for reactivity control following an ATWS. This is done by delivering the contents of the BAST via the charging pumps to the RCS via the normal charging line.

The third accident mitigation function is to replenish the inventory in the IRWST. This is done by delivering the contents of the BAST to the IRWST using the BAMPs.

The fourth accident mitigation function is RCP seal cooling. RCP seal cooling is normally accomplished using the CVCS centrifugal charging pumps taking suction from the BAST via gravity drain and discharging to the individual RCP seal packages via the RCP seal injection filters.

The ACP is a positive displacement pump that is placed in parallel with the CVCS centrifugal charging pumps. The ACP is manually started and supplies injection water when RCP seal injection is not available through the two centrifugal charging pumps. The ACP takes suction from the VCT or the BAST and supplies seal injection water to the RCPs through the normal CVCS seal injection flow path. The ACP is considered as a diverse capability from the two centrifugal pumps.

g. Reactor Protection System (RPS)

The RPS is a part of the plant protection system (PPS). Nuclear steam supply system (NSSS) parameters and containment conditions are monitored by the PPS continuously. If monitored conditions approach specific safety limits, the PPS

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The ACP is powered by the safety-related buses which are normally supplied by Class 1E onsite or offsite power. Following a Loss of Offsite Power, the buses will be re-energized by the emergency diesel generators. However, the ACP will not automatically re-start. It needs to be manually re-started by the operators after bus voltage has been restored.

bolts (versus the 40 bolts used to secure the hatch during at-power operation); four bolts are sufficient to secure the hatch so that no visible gap can be seen between the seals and the sealing surface. See Subsection 19.1.6.2.

COL 19.1(15) The COL applicant is to develop a configuration control program requiring that, during Modes 4, 5, and 6, the watertight flood doors and fire doors be maintained closed in at least one quadrant. Furthermore, the COL applicant is to incorporate, as part of the aforementioned configuration control program, a provision that if the flood or fire doors to this designated quadrant must be opened for reasons other than normal ingress/egress, a flood or fire watch must be established for the affected doors.

COL 19.1(15) The COL applicant is to develop outage management procedures that limit planned maintenance that can potentially impair one or both SC trains during the shutdown modes.

COL 19.1(16) The COL applicant is to develop procedures and a configuration management strategy to address the period of time when one SC train is unexpectedly unavailable (including the termination of any testing or maintenance that can affect the remaining train and restoration of all equipment to its nominal availability).

19.1.10 References

1. ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Revision 1 RA-S-2002), American Society of Mechanical Engineers, April 2008.

2. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008," American Society of Mechanical Engineers, February 2009.

3. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Rev. 2, U.S. Nuclear Regulatory Commission, March 2009.

COL 19.1 (20) The COL applicant is to develop management procedures for charging pump operation, following recovery from a loss of offsite power (LOOP), to ensure that deboration is not resumed until after at least one Reactor Coolant Pump (RCP) has been restarted.

Table 1.8-2 (28 of 29)

Item No.	Description
COL 19.1(11)	The COL applicant is to develop the fire barrier management procedures that direct the appropriate use of a fire watch and use of the isolation devices with a quick-disconnect mechanism for hose and cables that bleach a fire barrier.
COL 19.1(12)	The COL applicant is to develop procedures and operator training for reliance (during fire response) on undamaged instrumentation (when the location of the fire is known).
COL 19.1(13)	The COL applicant is to develop procedures specifying that a fire watch be present when hot work is being performed.
COL 19.1(14)	The COL applicant is to establish procedures for closing the containment hatch (after being opened during during LPSD operations) to promptly re-establish the containment as a barrier to fission product release. This guidance must include steps that allow for sealing of the hatch with four bolts (versus the 40 bolts used to secure the hatch during at-power operation); four bolts are sufficient to secure the hatch so that no visible gap can be seen between the seals and the sealing surface.
COL 19.1(15)	The COL applicant is to develop a configuration control program requiring that, during Modes 4, 5, and 6, the watertight flood doors and fire doors be maintained closed in at least one quadrant. Furthermore, the COL applicant is to incorporate, as part of the aforementioned configuration control program, a provision that if the flood or fire doors to this designated quadrant must be opened for reasons other than normal ingress/egress, a flood or fire watch must be established for the affected doors.
	The COL applicant is to develop outage management procedures that limit planned maintenance that can potentially impair one or both SC trains during the shutdown modes.
COL 19.1(16)	The COL applicant is to develop procedures and a configuration management strategy to address the period of time when one SC train is unexpectedly unavailable (including the termination of any testing or maintenance that can affect the remaining train and restoration of all equipment to its nominal availability).
COL 19.2(1)	The COL applicant is to perform and submit site-specific equipment survivability assessment in accordance with 10 CFR 50.34(f) and 10 CFR 50.44.
COL 19.2(2)	The COL applicant is to develop and submit an accident management plan.
COL 19.1 (20)	The COL applicant is to develop management procedures for charging pump operation, following recovery from a loss of offsite power (LOOP), to ensure that deboration is not resumed until after at least one Reactor Coolant Pump (RCP) has been restarted.

Table 19.1-4 (25 of 25)

No.	Insight	Disposition
	Risk Insights from PRA Models	
58	The fire PRA assumes that the fire barrier management procedures used during LPSD will include directions to provide reasonable assurance that breached risk-significant fire barriers can be closed in sufficient time to prevent the spread of fire across the barrier. The procedural direction is to include the use of a fire watch whose duties are commensurate with the risk associated with the barrier. For example, for fire barriers that separate two fire compartments that both contain no equipment or cables necessary to prevent core damage or large early release during LPSD conditions, or have been demonstrated to have low risk significance, there will at least be a roving fire watch to check the barrier during rounds. For fire barriers separating fire compartments that contain equipment or cables necessary to prevent core damage or large early release during LPSD conditions, and have been demonstrated to be risk significant with respect to fire, a permanent fire watch will be established until the barrier is reclosed. In the latter case, the fire barrier management procedure is to direct that hoses or cables that pass through a fire barrier use isolation devices on both sides of a quick-disconnect mechanism that allow for reclosure of the barrier in a timely fashion to re-establish the barrier prior to fire spread across the barrier.	Subsection 19.1.6.3.1.2 COL 19.1(11)
XX	Boron dilution events at shutdown were screened from the analysis, due to the diverse means of event identification, the availability of procedural recovery actions and the time available for operator response.	Subsection 19.1.3.1 COL 19.1 (20)

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RAI No.:	**409-8325**
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Question No. 19-21

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. SRP Chapter 19, Revision 3 (Draft), “Design-Specific PRA (PRA for Non-Power Modes of Operation)” states that, “Given that shutdown risk may be highly outage-specific, the staff reviews the shutdown PRA insights to confirm that operational assumptions used to develop an average shutdown model (e.g., use of nozzle dams, outage schedule, containment status, procedural requirements) have been clearly documented in the FSAR.” DCD section 19.2.2.2, "Midloop Operation" states, "Alternate inventory additions and decay heat removal methods if SCS is lost during Mode 5 reduced water inventory operations, containment spray (CS) pumps or the safety injection (SI) pumps are used to provide makeup. If all above methods of decay heat removal and inventory replenishment are unavailable, a charging pump or a boric acid makeup pump is used to provide makeup for Modes 5 and 6. If no method of pumped inventory addition is available, a source for gravity feed inventory addition can be used via the SI tanks." In Section 19.2.2.2 of the DCD, the staff requests the following information to be addressed:

a. Please justify how the safety injection tanks (SITs) can keep the core covered assuming the RCS is vented via the pressurizer given possible pressurizer surgeline flooding. Surgeline flooding following an extended loss of decay heat removal (DHR) may negate the elevation head necessary for SIT flow. Based on the shutdown evaluation report, the staff understands “With the earliest nozzle dam installation occurring at 4 days after shutdown, the decay heat present would require approximately 481 L/min (127 gpm)”.

b. Please clarify whether a charging pump and a boric acid pump are needed to keep the core covered or if either a single charging pump or a single boric acid pump is sufficient to keep the core covered. Please include the flowrate capabilities of the pumps.

Response

a. This question is currently being addressed in the response to RAI 8432, Question 19.03-13.

b. A charging pump with a makeup capacity of 150 gpm is available to provide sufficient decay heat removal capability at this time after shutdown. The boric acid makeup pump is not required and will be deleted from the discussion in the DCD.

Impact on DCD

DCD Section 19.2.2.2 will be revised as shown in 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

There is no impact on any Technical, Topical, or Environmental Reports.

opened via the pressurizer manway prior to reduced-inventory operation. When the pressurizer manway is opened to the containment atmosphere, the surge line provides sufficient venting capacity to prevent RCS pressurization and preclude subsequent nozzle dam failure. The pressurizer surge line vent pathway has sufficient capacity to prevent core uncovery due to pressurization of the hot side resulting from boiling coolant.

d. Alternate inventory additions and decay heat removal methods

(flowrate capability of 150 gpm)

If SCS is lost during Mode 5 reduced water inventory operations, containment spray (CS) pumps or the safety injection (SI) pumps are used to provide makeup. If all above methods of decay heat removal and inventory replenishment are unavailable, a charging pump ~~or a boric acid makeup pump~~ is used to provide makeup for Modes 5 and 6. If no method of pumped inventory addition is available, a source for gravity feed inventory addition can be used via the SI tanks.

19.2.2.3 Station Blackout

The minimum makeup flow of 481 L/min (127 gpm) is required to keep the core covered for the loss of DHR during the mid-loop operation.

One alternate ac (AAC) source is provided to help mitigate the effects of an SBO. The AAC automatically starts and is manually aligned to provide power to a Class 1E 4.16 kV bus in case Class 1E emergency diesel generators (EDGs) fail to start and load during loss-of-offsite-power (LOOP) events. This standby unit is independent and diverse from the Class 1E EDGs. Successful startup of the AAC together with turbine-driven auxiliary feedwater pumps is sufficient to prevent core damage in station blackout events (SBOs).

19.2.2.4 Fire Protection

The systems and components required for safe shutdown are physically separated from functionally similar or redundant systems or components to maintain the ability to perform safe shutdown functions in the event of a fire. Fire protection features such as fire detection, automatic and manual fire suppression, and fixed fire barriers provide reasonable assurance that the plant does not enter an unrecoverable state as a result of a fire incident. Fire protection system is described in Subsection 9.5.1.

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Question No. 19-22

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. SRP Chapter 19, Revision 3 (Draft), "Design-Specific PRA (PRA for Non-Power Modes of Operation)" states that, "Given that shutdown risk may be highly outage-specific, the staff reviews the shutdown PRA insights to confirm that operational assumptions used to develop an average shutdown model (e.g., use of nozzle dams, outage schedule, containment status, procedural requirements) have been clearly documented in the FSAR." The APR1400 design has incore instrument nozzles installed from the bottom of the vessel. The staff is asking whether temporary seals are used during refueling and/or maintenance similar to operating pressurized water reactors (PWRs). The staff could not find information on the design pressure of any temporary seals and the leakage from the seals during a postulated reactor coolant system (RCS) re-pressurization. The staff is requesting that information regarding temporary seals used for the incore instrumentation be documented in Section 19.1.6 of the DCD.

Response

There are three (3) types of ICI system generally on the pressurized water reactors (PWRs) which adopt a Bottom Mounted In-Core Instrumentation (BM-ICI) system;

1) "L" type BM-ICI system using a movable ICI system as Hanul Power Plants 1&2 (Supplied by AREVA)

2) "U" type BM-ICI system using a movable ICI system as Kori Power Plants 3&4 (Supplied by Westinghouse)

3) "U" type BM-ICI system using a fixed ICI system as Shin Kori Power Plants 3&4 (Palo Verde Reactor Type)

In the above cases 1) and 2), the movable ICI system needs not only temporary seals during refueling but also maintenance during normal operation.

The APR1400 which adopts a fixed BM-ICI system, however, does not need any temporary seals during shutdown mode, even though vessel head is on and mid-loop evolutions are in progress (refer to Table 1.4-1 of shutdown evaluation report).

Impact on DCD

The DCD 19.1.6.1.1.5 will be revised as shown in 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

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

c. Failure to begin secondary cooling before RCS level drops below the top of the hot leg is assumed to result in failure of secondary cooling.

d. One SG is assumed to be rendered unavailable by planned outage activities when the plant enters POS 4A.

e. The success criteria and time available for operator actions and events occurring in POS 3B is assumed to be the same as for events that occur in POS 3A. Since RCS temperature is lower in POS 3B, the timing for events is expected to take longer and, therefore, this assumption is conservative.

f. If feed and bleed cooling is used in POS 3A, containment design pressure would be exceeded after 24 hours. Although containment ultimate pressure capability will not be exceeded within 24 hours, operator action to begin IRWST cooling is assumed to be required to provide reasonable assurance safe, stable conditions.

g. Success criteria for unrecoverable LOCA (JL) events are analyzed assuming that the maximum break is the 34.1 m^3/hr (150 gpm) flow rate of the CVCS letdown line that occurs at-power.

h. Success criteria for LTOP safety valve fails to reclose (RL) events are based on the relief capacity of one LTOP relief valve.

Tables for the success criteria for LPSD various initiating event categories and operating states are shown in Table 19.1-89 through Table 19.1-92.

19.1.6.1.1.6 Human Reliability Analysis

The human reliability analysis (HRA) for the LPSD PRA is performed using the same methods as the at-power PRA described in Subsection 19.1.4.1.1.7.

Operator actions that respond to events that occur in Technical Specification Mode 2 or Mode 3 are assumed to be the same as the responses to events that occur at-power. Although the time available for response to an event in Mode 2 or Mode 3 is expected to be longer, thereby resulting in a lower HEP, this conservatism is considered to be negligible to overall risk because the time spent in these modes is short.

i. Success criteria in POS 7 and 9 for refueling does not consider seal leaks of ICI system which is a fixed ICI system of "U" type BM-ICI system preventing seal leaks

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Question No. 19-23

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. SRP Chapter 19, Revision 3 (Draft), “Design-Specific PRA (PRA for Non-Power Modes of Operation)” states that, “Given that shutdown risk may be highly outage-specific, the staff reviews the shutdown PRA insights to confirm that operational assumptions used to develop an average shutdown model (e.g., use of nozzle dams, outage schedule, containment status, procedural requirements) have been clearly documented in the FSAR.”

The staff understands interfacing-systems loss-of-coolant accidents (ISLOCAs) were screened from the low-power shutdown (LPSD) PRA. The staff also understands that the chemical and volume control system (CVCS) letdown line is directly connected to the reactor coolant system (RCS) and is a primary interface through which an ISLOCA event can begin. Pressurization is postulated from the letdown nozzle, through the regenerative and letdown heat exchangers, through the letdown orifices, and out of containment through the containment isolation and letdown control valves to the low-pressure sections of the system. The letdown line has a high-pressure alarm that is located downstream of the letdown control valves and warns the operator when the pressure is approaching the low-pressure system design pressure. When a warning is issued, the control room operator isolates the letdown line to terminate any further pressure. The staff is requesting additional information in Section 19.1.6 of the design control document (DCD) justifying why ISLOCAs were screened from the PRA. Specifically, the staff is requesting additional information in Section 19.1.6 of the DCD explaining how the closure of this valve is modeled during any postulated RCS re-pressurization when letdown is operating.

Response

The ISLOCA initiating event was retained in the low power and transition modes, using the same frequency as the at-power PRA. There is a negligible ISLOCA vulnerability once the

reactor is depressurized. Prior to establishing a primary vent, a letdown line rupture or a diversion LOCA was examined for a potential containment bypass vulnerability.

These initiators were analyzed as follows.

A rupture of the CVCS letdown line is explicitly included as an initiating event (JL, an unrecoverable LOCA). Letdown isolation is modeled as per Table 5.8.1 of Section 5.8.3 ("Event Tree Nodes") in the LPSD Accident Sequence Analysis Notebook (APR1400-K-P-NR-013702, Rev. 0). Letdown isolation is specifically checked at the IL node of the applicable event trees.

However, the letdown rupture is not a containment bypass vulnerability. There are restricting orifices downstream of the letdown isolation valves, but still within containment, that limit letdown flow even with full RCS pressure during power operation. During shutdown operation, the primary system pressure is reduced and the potential rupture flow between the isolation valves. A rupture upstream of the orifices will occur within containment; a downstream rupture will result in a negligible break flow. Therefore the letdown isolation does not impact potential containment bypass.

A diversion LOCA at shutdown may be similar to an ISLOCA, whether the primary system has been depressurized or not. This initiator was explicitly modeled.

Impact on DCD

The DCD will be revised as shown in 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

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

The POS defined for the APR1400 are summarized in Table 19.1-81. The POS for the APR1400 are defined in a manner that is consistent with the draft LPSD PRA Standard.

19.1.6.1.1.3 Initiating Events

The identification of potential initiating events considers generic information sources, information from similar plants, and a systematic review of the APR1400 design to identify unique initiating events. A detailed failure modes and effects analysis (FMEA) was performed to identify potential initiating events. The potential initiating events are grouped into similar functional categories to reduce the complexity of the PRA. The initiating event frequency for each of these groups is then quantified.

Once the initiating events are identified with preliminary definitions, the final initiating event groups are developed with the final group definitions. Since these initiating events are similar to those of existing nuclear power plants, the frequency for each initiating event is based on generic estimates for the operating power plants. When generic estimates are not available or the APR1400 design indicates that a different frequency is more appropriate, engineering judgment is used to estimate the initiating event frequency.

Based upon this review of shutdown PRAs for PWRs, the following shutdown initiating events are selected:

S1 – Recoverable loss of shutdown cooling system

S2 – Unrecoverable loss of shutdown cooling system

SO – Overdrainage during reduced inventory operation

SL – Failure to maintain water level during reduced inventory operation

SL – Small break LOCA

SL1 – Small break LOCA during reduced inventory operation

SL2 – Small break LOCA above reduced inventory operation

SG – Steam generator tube rupture

An explicit Intersystem LOCA (ISLOCA) initiating event was not included in the shutdown states. The ISLOCA initiator is analyzed in the low power and transition states, and the vulnerability after primary system depressurization is negligible. In the shutdown states with the primary system intact, the diversion LOCA captures the potential vulnerability of an ISLOCA. A letdown line rupture will not result in a significant break flow downstream of the letdown orifices in containment. Nevertheless, letdown isolation is specifically checked in the letdown rupture analysis.

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Question No. 19-24

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. SRP Chapter 19, Revision 3 (Draft), “Design-Specific PRA (PRA for Non-Power Modes of Operation)” states that, “Given that shutdown risk may be highly outage-specific, the staff reviews the shutdown PRA insights to confirm that operational assumptions used to develop an average shutdown model (e.g., use of nozzle dams, outage schedule, containment status, procedural requirements) have been clearly documented in the FSAR.” In Section 19.2.5.1.1.2, "Accident Management - During Low- Power Shutdown Operations," the design control document (DCD) states, "If RCS water level decreases too far, it can reach a level that is insufficient for SC pump suction. If this occurs, SC pumps are isolated to prevent damage to the pumps. In this situation, the charging pumps can be used to increase RCS water level and allow resumed operation of the SCS." Based on staff review of DCD Chapter 9, each charging pump has a rated flow rate of 155 gpm. The staff is requesting additional information be included in DCD Section 19.2.5.1.1.2 whether one or two charging pumps are needed to keep the core covered early in the outage, addressing plant operation state (POS) 3 through POS 5.

Response

As described in the shutdown evaluation report (SDER), the minimum makeup flow of 481 L/min (127 gpm) is required to keep the core covered for the loss of DHR during the mid-loop operation. This makeup flow is calculated based on the boil-off condition with the decay heat of 4 days after shutdown. Even though each charging pump has a rated flow rate of 155 gpm during normal operation, one charging pump flowrate is limited to maximum 150 gpm during mid-loop operation to prevent boron dilution, which is greater than the minimum required makeup flow (127 gpm). Therefore, during POS 5, one charging pump is needed to keep the core covered.

The minimum makeup flow depends on the decay heat level. Therefore, during the POS 3 and POS 4, the required charging pump to keep the core covered can be two in accordance with the

decay heat level.

Impact on DCD

The DCD will be revised as shown in 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

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

In LOOP scenarios, the APR1400 design prevents core damage by operation of emergency diesel generators to provide power. If the emergency diesel generators are unavailable, an alternate AC source is available to provide backup power.

19.2.5.1.1.2 During Low-Power Shutdown Operations

During LPSD operations, core damage is prevented by operation of the SI, recovery of water level using the charging pumps, secondary side cooling, and SCS isolation if the water level is insufficient.

If inventory loss is identified during LPSD operations, operators can manually isolate the failed SCS train to terminate the loss of inventory.

If RCS water level decreases too far, it can reach a level that is insufficient for SC pump suction. If this occurs, SC pumps are isolated to prevent damage to the pumps. In this situation, the charging pumps can be used to increase RCS water level and allow resumed operation of the SCS.

In LPSD scenarios where the RCS is fully filled with water, secondary cooling can be used to induce natural circulation in the coolant loops to remove decay heat from the core.

The SIS is isolated during LPSD operations; however, at least two trains are kept in standby so that SI can be available to provide core cooling if necessary; when needed, the SIS is manually activated by the operators.

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19.2.5.1.2 Retain the Core within the Reactor Vessel

19.2.5.1.2.1 During Operations at Power

The onset of core damage is identified when core-exit temperature reaches 922.04 K (1,200 °F). The primary way to terminate the progress of core damage is inject water into the reactor vessel. This can be achieved by operation of the SI, SC, or CS pumps. Once core relocation to the lower plenum occurs, another option available to prevent accident progression is ex-vessel cooling.

A

As described in the shutdown evaluation report, the minimum makeup flow of 481 L/min (127 gpm) is required to keep the core covered for the loss of DHR during the mid-loop operation. This makeup flow is calculated based on the boil-off condition with the decay heat of 4 days after shutdown. Even though each charging pump has a rated flow rate of 155 gpm during normal operation, one charging pump flowrate is limited to maximum 150 gpm during mid-loop operation to prevent boron dilution, which is greater than the minimum required makeup flow (127 gpm). Therefore, during POS 5, one charging pump is needed to keep the core covered. The minimum makeup flow depends on the decay heat level. Therefore, during the POS 3 and POS 4, the required charging pump to keep the core covered can be two in accordance with the decay heat level.

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.:	**409-8325**
SRP Section:	**SRP 19**
Application Section:	**19.1**
Date of RAI Issued:	**02/22/2015**

Question No. 19-25

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. SRP Chapter 19, Revision 3 (Draft), “Design-Specific PRA (PRA for Non-Power Modes of Operation)” states that, “Given that shutdown risk may be highly outage-specific, the staff reviews the shutdown PRA insights to confirm that operational assumptions used to develop an average shutdown model (e.g., use of nozzle dams, outage schedule, containment status, procedural requirements) have been clearly documented in the FSAR.” In the KHNP PRA notebook APR1400-K-P-NR-013702, LPSD Accident Sequence Analysis, Section 4.6, General Assumptions, it defines core damage as peak cladding temperature (PCT) > 1300°F. Please include this definition of core damage in Chapter 19 of the DCD.

Response

The definition of core damage will be included in Chapter 19 of the DCD.

Impact on DCD

The definition of core damage which is provided as the response of this RAI will be added in DCD 19.1 as shown in 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

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

c. Failure to begin secondary cooling before RCS level drops below the top of the hot leg is assumed to result in failure of secondary cooling.

d. One SG is assumed to be rendered unavailable by planned outage activities when the plant enters POS 4A.

e. The success criteria and time available for operator actions and events occurring in POS 3B is assumed to be the same as for events that occur in POS 3A. Since RCS temperature is lower in POS 3B, the timing for events is expected to take longer and, therefore, this assumption is conservative.

f. If feed and bleed cooling is used in POS 3A, containment design pressure would be exceeded after 24 hours. Although containment ultimate pressure capability will not be exceeded within 24 hours, operator action to begin IRWST cooling is assumed to be required to provide reasonable assurance safe, stable conditions.

g. Success criteria for unrecoverable LOCA (JL) events are analyzed assuming that the maximum break is the 34.1 m^3/hr (150 gpm) flow rate of the CVCS letdown line that occurs at-power.

h. Success criteria for LTOP safety valve fails to reclose (RL) events are based on the relief capacity of one LTOP relief valve.

Tables for the success criteria for LPSD various initiating event categories and operating states are shown in Table 19.1-89 through Table 19.1-92.

19.1.6.1.1.6 Human Reliability Analysis

The human reliability analysis (HRA) for the LPSD PRA is performed using the same methods as the at-power PRA described in Subsection 19.1.4.1.1.7.

Operator actions that respond to events that occur in Technical Specification Mode 2 or Mode 3 are assumed to be the same as the responses to events that occur at-power. Although the time available for response to an event in Mode 2 or Mode 3 is expected to be longer, thereby resulting in a lower HEP, this conservatism is considered to be negligible to overall risk because the time spent in these modes is short.

i. For LPSD events, core damage is assumed to occur if peak clad temperature exceeds 1300°F. This temperature is consistent with that used in NUREG/CR-6144 (BNL-NUREG-52399) and IMC-0609 Appendix G, attachment 2 published by the NRC.

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.:	**409-8325**
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Question No. 19-27

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. Based on Table 19.1-96, LPSD Internal Events PRA Top 100 CDF cutsets – All POSs, the top two cutsets are initiated by overdraining of the RCS to reach midloop conditions. To mitigate these events, the operators need to initiate reactor coolant system (RCS) injection and recover the Shutdown Cooling System. To quantify the failure rate of these operator actions, the analyst should consider dependence for core damage frequency calculations. Dependence was quantified in the top two cutsets. However, the staff searched through the low-power shutdown (LPSD) human reliability analysis (HRA) notebook and could not find how dependence was calculated or what factors were considered in the dependence calculation (e.g. similar alarms and cues). The staff could find the dependence calculations for other LPSD initiators in the LPSD HRA notebooks. The staff is requesting KHNP to provide the staff additional information on how dependence was calculated and update the DCD as necessary for: (1) reactor coolant system (RCS) overdraining at reduced inventory operation and (2) failure to maintain water level during reduced inventory operation, so the staff can better understand the numerical results of the KHNP LPSD PRA.

Response

Human error is dominant in the LPSD cutsets. For example, five (5) of the top ten Level 1 cutsets include a failure to recover the shutdown cooling system (SCS), followed by a subsequent failure to initiate feed & bleed (F&B) operation. Another three (3) of the top ten cutsets include a failure to provide RCS makeup, followed by a failure to initiate F&B operation.

The LPSD HRA is documented in LPSD Human Reliability Analysis Notebook (APR1400-K-P-NR-013705-P, Rev.0). It quantifies the dependency between subsequent human actions with the methodology which was described in Appendix N in the Human Reliability Analysis Notebook of Full Power Level 1 (APR1400-K-P-NR-013105-P, Rev.0). The overdrain during reduced inventory operation (SO) and failure to maintain water level during reduced inventory

operation (SL) initiating events were grouped with the unrecoverable LOCA (JL) for the operator actions addressed by this RAI.

The cues for operator action to attempt SCS recovery or initiate RCS makeup are the SCS pump discharge flow alarm and an abnormal increase in the Reactor Coolant System (RCS) temperature. For example, the worksheets for the development of HR-RS-JL(SO,SL)P05 and HR-MI-JL(SO,SL)P05 in LPSD Human Reliability Analysis Notebook (APR1400-K-P-NR-013705-P, Rev.0). If the initial action fails, the operators must initiate F&B operation to prevent core damage. The cues for F&B operation include the same SCS pump alarm but also uniquely include an abnormal increase in the Core Exit Thermocouple (CET) indications. And for example, the worksheet for HR-FB-JL(SO,SL)P05-01. Since the CET cue for the second action is uniquely different from either cue for the first action, the second action is assigned a low dependence (LD). The combined failure probability is quantified with the Table 10-2 dependency equations in NUREG/CR-1278.

Impact on DCD

The DCD will be revised as shown in 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

There is no impact on Technical/Topical/Environmental Report.

Operator actions for responses to events that occur in TS Modes 4, 5, or 6 are summarized in the following groups:

a. Actions to restore SCS

b. Actions to provide reasonable assurance of secondary cooling

c. Actions to initiate feed and bleed cooling

d. Actions to isolate RCS leakage and restore inventory

e. Actions to align the AAC power source

The time available to perform each of these categories of actions varies with POS. As a result, the HEPs for each event also vary with POS. Some actions are not applicable to all initiators and timing can be affected by specific initiating events. For example, actions to isolate RCS leakage and restore inventory are not applicable to loss of SCS initiating events.

19.1.6.1.1.7 Systems Analysis

Insert A in next page.

The following summarizes differences in LPSD PRA system models versus at-power PRA modeling:

a. Safety Injection System

1) The SITs are isolated in the late POS 2 (TS Mode 3) and below. They are considered unavailable during all of POS 3 through POS 13.

2) Manual actuation of the SIPs is assumed to be required in Mode 4 (POS 3) and below. Automatic actuation is not credited.

b. Shutdown Cooling System

1) The system is modeled as aligned for shutdown with one train in operation and one train on standby.

2) No maintenance is performed on the SCS when operation of the system is required.

A

Some of the top cutsets include operator failures to perform multiple actions including, for example, a failure to restore the Shutdown Cooling System, followed by a failure to initiate feed & bleed operation. The Cause-Based Decision Tree Methodology (CBDTM) methodology, as implemented in the EPRI HRA Calculator (Reference 26), was used to evaluate the dependency of the subsequent operator action. The dependency itself was quantified with the NUREG/CR-1278 (Reference 21), Table 10-2 equations.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

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Question No. 19-28

10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. The low-power shutdown (LPSD) large release frequency (LRF) contribution from midloop operation is reduced because credit is taken for initiation of safety injection (SI) to arrest core damage in the vessel as a severe accident mitigation guidelines (SAMG) action. However, a key contributor to the LPSD core damage frequency (CDF) in the mid-loop plant operational state (POS) is due to operator failure to initiate SI before core damage. The staff noted that credit for the SAMG action of initiating SI is included in the Containment Event Tree top event, MELTSTOP. The staff searched through the LPSD human reliability analysis (HRA) notebook and could not find how dependence between the Level 1 and Level 2 LPSD PRA was calculated for these two actions or what factors were considered in the dependence calculation (e.g. similar alarms and cues). The staff is requesting KHNP to provide the staff additional information on how dependence was calculated between the operator action to initiate SI to prevent core damage and the SAMG action to initiate SI to arrest core damage in the vessel and to update the DCD, as necessary. The staff needs this information to better understand the numerical results of the KHNP LPSD PRA.

Response

New text will be added to the DCD to address the Level 2 dependence in the credit for SI initiation.

Impact on DCD

DCD Section 19.1.6.2.1.3 will be revised as shown in the Attachment.

The following paragraph will be added to DCD Section 19.1.6.2.1.3, in the subsection on Safety Injection:

“SI Initiation is first credited early in the accident sequence for the prevention of core damage, and again later, during SAMG actions, to arrest core damage in the vessel. The SAMGs are entered when the Core Exit Thermocouples indicate 1200 degrees F, which is unique and unrelated to the initial cue for Safety Injection before core damage occurs. Therefore the dependency evaluation yielded a Low Dependence between the first opportunity for SI initiation and the subsequent SAMG cue. The subsequent dependency calculation increased the SAMG SI initiation HEP accordingly.”

Impact on PRA

The DCD will be revised as shown in Attachment.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on Technical/Topical/Environmental Report.

some point after core damage but before vessel failure in which recovery of SI prevents vessel failure; there is uncertainty as to the exact criteria to define this point, so conservatism is applied in this instance.

For the LPSD analysis, successful initiation of SI to prevent vessel failure is assumed until the time of core damage. This allows some conservatism to account for uncertainties in the thermal-hydraulic analysis, which provides sufficient realism to credit the SAMG action. The time to reach 1,200 °F and the time to core damage were developed using the MAAP code for a mid-loop scenario with loss of SDC and no SI.

External Reactor Vessel Cooling (ERVC)

insert A in next page

When the core exit thermocouple temperature of 1,200 °F is reached, another SAMG action is initiation of the cavity flooding system (CFS). This is credited if there is a mechanical failure of SI, but if there is an operator failure to initiate SI by the SAMG action, then complete dependence is assumed, and no credit is given to cavity flooding by SAMG action. In addition, the ERVC system provides a means to flood the containment cavity in an attempt to prevent vessel failure by cooling the molten core from outside the vessel. Consistent with the at-power Level 2 PRA, ERVC is not credited in the baseline LPSD analysis.

d. DCF – No Dynamic Containment Failure

Given that the RV has failed (failure of the MELTSTOP top event), the LPSD CET then questions whether the containment fails dynamically at vessel breach. This event is similar to the at-power Level 2 CET event DCF, except that in the LPSD model for POS 4B-12A, the RCS pressure will always be “low” at vessel breach (due to the open pressurizer manway). The evaluation of “low” pressure vessel failure from the at-power Level 2 analysis presents a $1x10^{-3}$ probability of dynamic containment failure due to Alpha-mode containment failure. The same value is conservatively used for the LPSD Level 2.

e. ECF – No Early Containment Failure

A

SI Initiation is first credited early in the accident sequence for the prevention of core damage, and again later, during SAMG actions, to arrest core damage in the vessel. The SAMGs are entered when the Core Exit Thermocouples indicate 1200 degrees F, which is unique and unrelated to the initial cue for Safety Injection before core damage occurs. Therefore the dependency evaluation yielded a Low Dependence between the first opportunity for SI initiation and the subsequent SAMG cue. The subsequent dependency calculation increased the SAMG SI initiation HEP accordingly.