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.:	**409-8325**
SRP Section:	**SRP 19**
Application Section:	**19.1**
Date of RAI Issue:	**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 – (Rev. 1)

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 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

insert A in next page.

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A

~~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.~~

The charging pumps are powered by the safety-related buses which are normally supplied by Class 1E onsite or offsite power. Following a loss of offsite power (LOOP), the buses will be re-energized by the emergency diesel generators. However, the charging pumps 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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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 – (Rev. 1)

There are generally three (3) types of the ICI systems on a pressurized water reactor (PWR) which adopt the 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 cases 1) and 2) above, the movable ICI system needs temporary seals during refueling and 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

DCD section 19.1.6.1.1.5 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.

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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~~

i. The temporary seal is not required during refueling, since the BM-ICI system uses a fixed ICI system.

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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 – (Rev. 1)

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.

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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.~~

Replace with "A"

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A

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. The letdown rupture is not a containment bypass vulnerability because 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 results in a break 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.