KHNPDCDRAIsPEm Resource

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Sent: Monday, February 22, 2016 6:56 AM
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Cc: Pohida, Marie; Mrowca, Lynn; Steckel, James; Lee, Samuel
Subject: APR1400 Design Certification Application RAI 409-8325 (19 - Probabilistic Risk Assessment and Severe Accident Evaluation)
Attachments: APR1400 DC RAI 409 SPRA 8325.pdf

KHNP,

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form. Your licensing review schedule assumes technically correct and complete responses within 30 days of receipt of RAIs. However, KHNP requests, and we grant, 60 days to respond to RAI question 19-21. We may adjust the schedule accordingly.

Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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United States Nuclear Regulatory Commission
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REQUEST FOR ADDITIONAL INFORMATION 409-8325

Issue Date: 02/22/2016
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19

QUESTIONS

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 (EDGs). 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.

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.

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.

19-23

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 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.

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.

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) > 1300F. Please include this definition of core damage in Chapter 19 of the DCD.

19-26

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." During the evaluation of containment performance during low power and shutdown conditions, the staff noticed different containment ultimate pressure capacities were referenced in Section 19.3 of the design control document (DCD) (184 psig), Section 19.2 of the DCD (123 psig), and 19.1 of the DCD (163 psig). The staff is requesting the applicant to resolve these inconsistencies in the DCD or justify in the DCD why different containment ultimate pressure capacities were used.

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.

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.