Request for Additional Information 232-7864

Issue Date: 09/30/2015
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:

QUESTIONS

19-5

10 CFR 52.47(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 Shutdown Evaluation report (reference 22 in Section 5.4.7 of the DCD), Section 2.5.2 "Temperature Considerations", states "In the analyses, the temperature of 71 degrees Celsius (160 degrees Fahrenheit) is used as the allowable upper limit, and the containment temperature should be maintained to be less than this value during the required time to close all containment openings including hatch or personal air locks at an accident." The staff acknowledges that this statement concerns the Mode 5 LOCA analysis with water level in the pressurizer which is not reduced inventory operation. Since the safety injection pumps are assumed to be manually actuated at 32 minutes after the reactor trip by a pipe break, the staff understands no fuel damage is postulated for the Mode 5 LOCA with full inventory. However, the staff is requesting KHNP to justify reliable operator action for hatch closure in the Shutdown Evaluation Report, given the presence of (1) steam, (2) high humidity, (3) low visibility due to fog, and (4) high temperatures. The staff is requesting KHNP to clarify in Chapter 19 of the DCD whether the 160 degree Fahrenheit upper limit was used, when the hatch is permitted to be opened, to develop the likelihood of the operator failing to re-close the equipment hatch above reduced inventory conditions following a loss of decay heat removal.

19-6

10 CFR 52.47(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, "The staff reviews the applicant's implementation of the applicable expeditious actions outlined in NRC Generic Letter (GL) 88-17 (Ref. 14)." The staff needs to ensure that the applicant is implementing the steam generator (SG) nozzle dam guidance described in GL 88-17 Enclosure Section 2.7, Nozzles Dams. This section addresses the proper sequence of nozzle dam installation and removal to prevent a sudden loss of RCS inventory given a loss of decay heat removal (DHR). This section also discusses the need for nozzle dam design pressures not to be exceeded given a postulated loss of DHR. Loss of a nozzle dam could lead to rapid reactor vessel voiding. These configurations are not modeled in LPSD PRAs. The GL recommends **removing** a pressurizer manway (**if analyses show this to provide a sufficient vent path**) or otherwise create a suitable opening while nozzle dams and the reactor vessel head are in place. In summary, SRP Chapter 19 also states, "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." Regarding SG nozzle dam installation, SG nozzle dam removal, and cold leg penetrations to prevent rapid core voiding, the staff has the following questions:

Request for Additional Information 232-7864

a. The risk significant operational assumptions regarding SG nozzle dam integrity, SG nozzle dam installation, and SG nozzle dam removal documented in DCD Section 19.2.2.2.c., should be included in the Risk Insights Table or a justification should be provided documenting why inclusion in the risk insight table is not necessary.

b. The following statement in DCD Section 19.2.2.2.c , “The installation procedure requires that the pressurizer manway be opened so that a hot side vent pathway exists prior to blocking both RCS hot legs with nozzle dams” seems to be inconsistent with the statement in DCD Section 19.2.2.2, "In order to provide reasonable assurance that nozzle dam design pressures are not exceeded during reduced inventory operations with boiling conditions in the reactor vessel, the APR 1400 design requires that a mid-loop vent pathway is opened via the pressurizer manway prior to reduced inventory operation." Please resolve the inconsistency in the DCD and the Risk Insights table.

c. In GL 88-17, Section 2.1.1, Pressurization, it states, “cold leg openings can allow water to be ejected from the vessel following a loss of DHR until sufficient water is lost that steam is relieved by clearing of the crossover pipes.” Information Notice 88-36 also discusses the potential for a rapid loss of RCS inventory through cold leg penetrations. IN 88-36 states, “The possibility of ejecting coolant by this mechanism can be eliminated by ensuring that a steam generator hot leg plenum manway and its associated hot leg pipe are kept open to provide an adequate vent path whenever any cold leg openings are made. This can be accomplished by ensuring that a hot leg manway is the first manway to be opened, and a hot leg nozzle dam is the last dam to be installed.” Please update the nozzle dam installation and removal procedures in the Risk Insights Table and DCD Section 19.2.2.2.c. to be consistent with the guidance from IN 88-36 or please justify why this change is not necessary.

d. Regarding overpressure protection of the SG nozzle dams as described in GL 88-17, the Shutdown Evaluation Report states, “An acceptable, conservative RCS equilibrium pressure that is below the assumed SG nozzle dam abnormal design pressure limit has been calculated when mid-loop operation is assumed to start at 4 days after shutdown. Therefore, the recommended earliest time after shutdown (from full power) for operating at mid-loop level is 4 days. It fixes the time to boil [

]

e. Concerning the differential pressure capability of the SG nozzle dams, DCD Section 19.2.2.2 states, “In the APR1400 design, the ability of the RCS to withstand abnormal pressurization during reduced-inventory operations with the nozzle dams installed is limited by the design pressure of the nozzle dams. Based on overpressure tests performed on nozzle dams, the design pressure is estimated to be 3.52 kg/cm2 (50 psia). The design pressure is sufficient to withstand an abnormal pressurization transient.” The staff understands these statements in this paragraph to mean that midloop operation initiated earlier than 4 days could result in an RCS re-pressuization beyond the differential pressure capability of the SG nozzle dams of 50 psid, if the DHR function is lost. Loss of a nozzle dam due to RCS pressurization could result in rapid RCS inventory loss and is risk significant. The staff has two questions:

> i) In section 19.2.2.2 of the DCD, the staff is requesting the applicant to discuss the results of an analysis that evaluates RCS peak pressure given an RCS re-pressurization following a loss of decay heat removal, 96 hours post shutdown. This discussion is needed to document that the steam generator nozzle dams will remain intact given an abnormal pressurization transient.
>
> ii) According to SRP 14.3, the differential pressure capability of the nozzle dams is risk significant and should be considered for ITAAC.

Request for Additional Information 232-7864

19-7
10 CFR 52.47(a)(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. To review the LPSD large release frequencies (LRFs), the staff reviewed the assumptions in DCD Section 19.1.6.2.2.5, "Key Assumptions." One of the assumptions states, "B. Failure of hydrogen control from PARs and/or igniters is assumed to yield a conditional probability of containment rupture due to hydrogen detonation of 0.1, plus another conditional probability of containment rupture due to hydrogen burn of 0.1 or 0.01. These probabilities are believed to be conservative, but additional calculations are needed for confirmation." The staff is requesting that the applicant provide in the DCD the results of the additional calculations documenting the conditional containment failure probability due to hydrogen which impact the total LRF. The staff needs to compare total LRF against the Commission goals for new reactors as directed in the SRP for Chapter 19.

19-8
10 CFR 52.47(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 noted that LPSD risk from POS 7 and POS 9 was screened from the average shutdown model. The staff understands the cavity is filled to the level necessary for core alterations in POS 7 and POS 9. The staff also acknowledges the flow limitations in the letdown line. However, the possibility of installed reactor internals could shorten the time to core boiling given limited communication between the RCS inventory around the core and inventory in the refueling cavity. In addition, losses of RCS inventory could be caused by operators (valve mis-alignments). To confirm appropriate screening of POS 7 and POS 9 from the LPSD PRA, the staff needs the following information documented in the DCD: (a) an evaluation documenting the time to core damage given an extended loss of the decay heat removal function with installed reactor internals, (b) an evaluation documenting the time to core damage given an extended loss of the decay heat removal function with the reactor internals removed, (c) an evaluation that considers all possible drain paths from the refueling cavity including drain rates, (d) the availability of instrumentation and alarms to detect and mitigate each potential drain path, (e) the likelihood of the operator failing to terminate each potential leak path, and (f) the availability of pumps and a source of water to restore RCS inventory for each leak path.

19-9
10 CFR 52.47(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 DCD Section 19.1.6, the staff understands POS 12B was screened from the average shutdown model based on thermal-hydraulic analysis. The analysis assumes the time to core damage is greater than 24 hours after a loss of

shutdown cooling. However, losses of inventory occurring in POS 12B may result in core damage before 24 hours. The staff understands that for the LOCA cases, the applicant has performed analyses that conclude that core uncovery does not occur until 23.7 hours, and core damage does not occur within the simulation time of 25 hours. Thus, the staff requests that the applicant (1) document in the DCD the results of POS 12B thermohydraulic analyses for LOCA and non-LOCA cases and (2) document in the DCD core cooling mitigation strategies for LOCA and non-LOCA cases to ensure that a safe and stable state is reached.

19-10

10 CFR 52.47(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. SRP Chapter 19, Revision 3, states "The staff will determine that the applicant has performed a PRA-based SMA to determine the seismic capacity of the plant and for each sequence that may lead to core damage or large release." Thus, the staff requests KHNP to update the DCD Section 19.1.6 to provide a discussion on how the seismic margins approach was applied to low power and shutdown conditions. Specifically, the staff is requesting the applicant to update Section 19.1.6 of the DCD to include the seismic cutsets and the sequence HCLPF capacities. The staff also requests KHNP to update Section 19.1.6 of the DCD to include the dominant mixed cutsets containing seismic failures and random failures or operator actions.