KHNPDCDRAIsPEm Resource

From:	Ciocco, Jeff
Sent:	Thursday, October 22, 2015 9:33 AM
To:	apr1400rai@khnp.co.kr; KHNPDCDRAIsPEm Resource; Harry (Hyun Seung) Chang; Andy Jiyong Oh; Young H. In (yhin@KHNP.co.kr); James Ross
Cc:	Pohida, Marie; Mrowca, Lynn; Steckel, James; Lee, Samuel
Subject:	APR1400 Design Certification Application RAI 268-8308 (19 - Probabilistic Risk Assessment and Severe Accident Evaluation)
Attachments:	APR1400 DC RAI 268 SPRA 8308.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 19-12. 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
Protecting People and the Environment

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Options
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REQUEST FOR ADDITIONAL INFORMATION 268-8308

Issue Date: 10/22/2015
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: 19

QUESTIONS

19-11

10 CFR 52.47(27) requires that a standard design certification applicant provide a description of the design specific PRA and the results. Design Control Document Section 19.1.6., appears to be missing event trees for all plant operational states other than plant operational state 5, which represents midloop conditions. However, the plant response to a loss of decay heat removal (DHR) is significantly different if the reactor coolant system (RCS) is intact versus an open RCS. To provide an example of how the different plant operational states were modeled in the probabalistic risk assessment (PRA), the staff is requesting that the applicant add the event trees for all plant operational states for the initiating event, loss of the operating train of the shutdown cooling (SDC) system. The staff considers these low power shutdown (LPSD) event trees part of the PRA results. These event trees will allow the reader to understand the varying plant response to a shutdown initiating event given the different plant operational states.

19-12

10 CFR 52.47(27) requires that a standard design certification applicant provide a description of the design specific probabilistic risk assessment (PRA) and the results. Standard Review Plan (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, Section 2.8.3.2.1, Level, states, "Four unique sets of instruments are provided for the measurement of level during RCS drain down and reduced inventory operations. These instruments make up the refueling water level indication system (RWLIS)." The first set of instruments is a pair of wide-range, pressure differential (dP)-based level sensors. These sensors are provided to measure the level between the pressurizer (PZR) and the bottom of the hot leg during drain- down operations. Another pair of dP-based level sensors is used to determine reactor coolant system (RCS) water level once it is within the reactor vessel (RV). These narrow-range level sensors function to measure level between the direct vessel injection (DVI) nozzle and the bottom of the hot leg. The Ultrasonic level measurement system measures from twenty percent to one hundred

percent of the hotleg level. During a loss of shutdown cooling at reduced inventory conditions, it is not clear whether the operator reviews all three level indicators. If the RCS is vented via the pressurizer manway, and decay heat removal (DHR) is lost, RCS heatup and re-pressurization could result in hot leg inventory being swept into the pressurizer. For the wide-range level indication that is tapped into the pressurizer, in this condition, level indication could read erroneously high. However, the other two sets of narrow range indication should still be accurate. The staff requests additional information in the design control document (DCD) on how this condition has been accounted for in the post-initiator, human error probabilities (HEPs).

19-13

10 CFR 52.47(27) requires that a standard design certification applicant provide a description of the design specific probabilistic risk assessment (PRA) and the results. Standard Review Plan (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 is requesting additional information in the design control document (DCD) regarding how the reactor coolant system (RCS) is drained to midloop conditions sufficient to install steam generator (SG) nozzle dams. From evaluating the applicant's response to request for additional information (RAI) question 19-4, in plant operational state (POS) 04A, the staff understands the RCS is drained from normal operating levels in the pressurizer with the Reactor Coolant System Gas Vent system open, and the low temperature overpressure protection (LTOP) valves are in automatic protection mode. RCS inventory is drained through the chemical and volume control system (CVCS) letdown line. The staff has the following specific requests for additional information:

1. In POS 4a, the staff is requesting clarification in the DCD whether a cover gas is used to speed draining or prevent the RCS from drawing a vacuum.

2. In POS 4B, the staff is requesting clarification in the DCD whether the pressurizer manway is opened once midloop conditions are reached. Midloop conditions are defined in Generic Letter 88-17 as when the RCS water level is below the top of the flow area of the hotlegs at the junction with the reactor vessel.

3. The staff is also requesting the applicant to document in the DCD whether vacuum refill of the RCS is performed from midloop conditions.

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