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.:	**312-8343**
SRP Section:	**SRP 19**
Application Section:	**19.1**
Date of RAI Issued:	**11/16/2015**

Question 19-16

SRP Chapter 19.0, Revision 3 (Draft), Section II, "Acceptance Criteria," Item 14 on Page 19.0-16 states, "The staff will determine that FSAR Chapter 19 includes PRA qualitative results, including the identification of key PRA assumptions, the identification of PRA-based insights, and discussion of the results and insights from importance, sensitivity, and uncertainty analyses." The staff reviewed APR1400 DCD Chapter 19 and found insufficient information regarding reactor coolant pump (RCP) seal assumptions, leakage, detailed modeling, and failure information.

Therefore, in order for the staff to reach a reasonable assurance finding on the conformance to the SRP regarding the PRA results and insights, please provide the evaluation of RCP seal leakage and how it is modeled in the APR14000 PRA. Also include in the response the following information and revise the DCD as appropriate:

1. Seal LOCA failure probability and the reason for choosing engineering judgement as its basis
2. Failure modes and consequences
3. Leakage rates during normal operation and accidents
4. Timing of seal failure and conditional probability given a loss of power or seal cooling

Response – (Rev.2)

The evaluation of RCP seal leakage and the seal LOCA failure probability are documented in APR1400-A-M-NR-16001-P, Revision 0, "Model for RCP Seal Failure Given Loss of Seal Cooling for APR1400 KSB HDD-254 Type F RCP Seals," The RCP Seal LOCA is not modeled explicitly at the event tree level, but is modeled as a point estimate in the applicable event tree branches (see APR1400-K-P-NR-013102-P).

1. The RCP seal LOCA probability, given a total loss of seal cooling, is assumed to be equal to 1×10^{-3} per pump. Since there are 4 RCPs, the total seal LOCA probability is estimated to be 4×10^{-3}. This value was based on engineering judgement since the PRA was completed prior to the completion of the RCP Seal LOCA model which was being developed for the RCP seals to be used in the APR1400. The basis for the engineering judgement includes prior industry PRA experience gained applying industry RCP seal LOCA models (e.g., WCAP-15603, Rhodes model, etc.) considering the fact that the APR1400 RCP seals are to be an improvement over the prior seal packages. The results from the APR1400 RCP Seal Model (APR1400-A-M-NR-16001-P, Revision 0) estimate a per seal probability of [ ]TS (depending upon RCS conditions) for a total of [ ]TS; these results support the original assumption that 1×10^{-3} per pump (4×10^{-3} total) is conservative.

2. The loss of cooling from Division I CCW to the RCP thermal barrier may damage the RCP seals and lead to a seal LOCA. The RCP seal LOCA is considered in the following event trees: PLOCCW, TLOCCW, PLOESW, TLOESW, SBO, and GRID-SBO.

   Upon a loss of cooling for the RCP seals, the operators will manually trip the reactor and then trip the RCPs to prevent damage to the pumps and to avoid a challenge to the RPS. Failure of the reactor trip will result in either core damage or an ATWS situation depending on the integrity of RCP seals. If the RCP seals remain intact, then the sequence is evaluated as an ATWS. However, failure of the RCP seals, which is a LOCA, in conjunction with failure of reactor trip is assumed to result in core damage.

   Secondary Heat Removal (SHR) is required via AFW or the startup feedwater pump and steam relief. If SHR is successful and RCP seals remain intact, then core damage is averted. However, if SHR is successful but RCP seals fail, then SI injection is required to maintain RCS inventory allow continued decay heat removal. With the release of reactor coolant through the failed RCP seals, long term containment heat removal is required.

   If SHR is unavailable, then feed and bleed can be used for decay heat removal. Because feed and bleed cooling results in release of reactor coolant, the status of RCP seals is irrelevant to scenarios where SHR fails. During feed and bleed operation, the decay heat is transferred to the IRWST and containment. In order to maintain long term feed and bleed, decay heat removal requires either IRWST or containment spray cooling.

3. Limiting leakage rates during normal operation is 10 gpm per Tech Specs LCO 3.4.12, and the leakage rates during accidents with RCPs tripped is less than [ ]TS gpm per RCP upon a seal cooling from APR1400-A-M-NR-16001-P, Revision 0. If the RCPs are not tripped, the leakage results in a small LOCA.

4. Given a loss of power or seal cooling, the timing of seal failure is assumed to be T = 0, and the conditional probability is assumed to be 4.00E-03 in the current PRA model.

Impact on DCD

The DCD will be marked up as shown in the Attachments 1 and 2.

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 Reports.

a. The HRA is performed under assumptions that the operating procedures and guidelines will be prepared and that operator training will be similar to that at existing operating plants.

b. HEPs for different operator actions are estimated for the SBO conditions and non-SBO conditions. It is assumed that operators will have clear direction about the crosstie of buses and equipment during SBO conditions.

c. The CVCS is not credited for an injection function to make up the lost inventory.

d. ~~RCP seal LOCA probability, given a total loss of seal cooling and the RCP trip, is assumed to be equal to 1 × 10^{-3} per pump.~~ Change as "A" in next page

e. The entire year is used for evaluation of the initiating event frequencies for operations at power. The plant capacity factor is assumed to be 0.95. This assumption needs to be verified when plant-specific shutdown information is available.

f. Consequential LOOP is modeled. It is assumed that the consequential LOOP probabilities as a result of plant trips and LOCA events are different.

g. A failure to trip the reactor during a small break LOCA or SGTR event is assumed to lead to core damage, and no further model development was made.

h. The digital I&C system model retains the current hardware model from the reference plant, except for the software events and the communication link models. The digital I&C model is retained as-is with a single event representing the software/communication links as a black-box event. The event probability in the fault tree model is based on engineering judgment. The dependency between hardware and software/communication links is not evaluated, but will be evaluated when design details are finalized.

19.1.4.1.2.6 Uncertainty Analysis

Uncertainty in the Level 1 internal events PRA results is quantified using SAREX. The results of parametric uncertainty for Level 1 internal events CDF are summarized below:

"A"

The RCP seal LOCA probability, given a total loss of seal cooling and RCP trip, is assumed to be equal to 1×10^{-3} per pump. Since there are 4 RCPs, the total seal LOCA probability is estimated to be 4×10^{-3} where the timing of seal failure is assumed to be T=0. This value was based on engineering judgement since the PRA was completed prior to the completion of the RCP Seal LOCA model which was being developed for the RCP seals to be used in the APR1400. The basis for the engineering judgement includes prior industry PRA experience gained applying industry RCP seal LOCA models (e.g., WCAP-15603, Rhodes model, etc.) considering the fact that the APR1400 RCP seals are to be an improvement over the prior seal packages where the seal leakage rates during accidents with the RCPs tripped is minimal per RCP upon less of a seal cooling (Reference 59), and a seal failure probability much less than the original assumption. If the RCPs are not tripped, the leakage results in a small LOCA.

42. NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines,” U.S. Nuclear Regulatory Commission, November 2009.

43. EPRI 1016735, “Fire PRA Methods Enhancements: Additions, Clarifications, and Refinements to EPRI 1019189,” Electric Power Research Institute, December 2008.

44. NUREG/CR-4527, “An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets, Part II: Room Effects Tests,” U.S. Nuclear Regulatory Commission, April 1987.

45. Regulatory Guide 1.102, “Flood Protection for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, September 1976.

46. EPRI 1021086, “Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments (PRAs),” Electric Power Research Institute, October 2010.

47. NUREG/CR-6144 (BNL-NUREG-52399), “Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1,” U.S. Nuclear Regulatory Commission, June 1994.

48. Inspection Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determination Process,” U.S. Nuclear Regulatory Commission, February 2005.

49. NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Nuclear Energy Institute, July 2000.

50. NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline,” Rev. 0, Nuclear Energy Institute, July 2005.

51. CAFTA 6.0b, Software Manual, EPRI, Palo Alto, CA, 2014.

52. NUREG/CR-7114, “A Framework for Low Power/Shutdown Fire PRA,” U.S. Nuclear Regulatory Commission, September 2013.

53. NUREG/CR-7150, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE),” May 2014.

Add

59. APR1400-A-M-NR-16001-P, Revision 0, "PRA Model for RCP Seal Failure Given Loss of Seal Cooling for APR 1400 KSB HDD-254 Type F RCP Seals," KHNP, July 2016.