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.:	**431-8504**
SRP Section:	**15.06.05 – Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary**
Application Section:	**15.6.5**
Date of RAI Issue:	**03/08/2016**

Question No. 15.00.02-11

NUREG-0800, Standard Review Plan (SRP) Section 15.0.2, "Review of Transient and Accident
Analysis Methods," specifies that an evaluation model must be able to predict all important physical phenomena determined to be necessary for the accident under consideration reasonably well from both qualitative and quantitative points of view. However, it is not clear that the applicant's small break loss of coolant accident (SBLOCA) evaluation model is meeting this guidance with respect to loop seal clearing, and the staff is concerned that the modeling of the loop seal clearing phenomena may not be conservative.

The staff's review of APR1400 DCD Section 15.0.2, "Review of Transient and Accident Analysis
Methods," and the referenced Technical Report (TeR) APR1400-F-A-NR-14001-P, Rev.0,"Small
Break LOCA Evaluation Model," has raised several questions as submitted in this RAI. The information being sought is part of the analytical procedures that the staff uses to establish that the evaluation model treats loop seal clearing phenomena realistically or conservatively. The regulatory bases identified above are applicable to all subsequent questions in this RAI. The applicant is also requested to update the DCD and the TeR as appropriate to ensure that the analysis method and results are documented.

The TeR describes the APR1400 SBLOCA evaluation model in broad terms. In Section 1 of the
TeR, "Introduction", it is stated that the SBLOCA methodology used for APR1400 is very similar to the conventional Combustion Engineering (CE) SBLOCA methodology used for currently operating US CE-fleet PWRs. However, the report did not provide a discussion of the differences between the CE methodology and the KHNP methodology used for APR1400.

SRP Section 15.0.2 suggests review of any changes to the previously approved evaluation models to ensure the changes do not invalidate the previous approval. The applicant is requested to specify the exact approved CE SBLOCA methodology revision upon which the KHNP SBLOCA methodology is based, and discuss any differences between the two. The staff needs to understand all the changes made to the SBLOCA methodology and computer codes since they were last approved, in order to ensure that there is nothing new in the methodology that could invalidate the previous approval of the applicability or limitations of the methodology. The staff needs to review all the modifications made in the mathematical modeling, computer codes (CEFLASH-4AS, COMPERCII, STRIKIN-II, and PARCH) used to analyze the APR1400 SBLOCA, as well as any differences in data transfer between the codes since they were last approved. The staff also needs to know if the material properties for ZIRLO and M5 were incorporated into STRIKIN and PARCH. If so, the documentation and verification of the code changes should be submitted to the staff. If not, justification for not incorporating the ZIRLO and M5 material properties should be provided.

Response

KHNP uses the NRC-approved Westinghouse Small Break Loss of Coolant Accident (SBLOCA) Analysis Methodology (S1M) for CE-designed Pressurized Water Reactors (PWRs) except for the following change:

- Cladding properties for Zirlo

S1M is documented in References 1 and 2. The change above is documented in References 14. The change made to S1M by KHNP did not require any modifications to the thermal hydraulic models employed in S1M. This updated S1M will be referred as 'S1M-K' in the RAI responses for the convenience of discussion.

Reference 1 was issued in 1974 and Reference 2 was issued in 1977 and since then CE had used S1M to license Emergency Core Cooling System (ECCS) performance for SBLOCAs at CE-designed PWRs. S1M uses the following four computer codes:

1. CEFLASH-4AS for system thermal hydraulic transient calculation up to the time when Safety Injection Tanks (SITs) begin to inject
2. COMPERC-II for system thermal hydraulic transient calculation beyond the time when SITs begin to inject
3. STRIKIN-II for hot rod calculation during the initial blowdown phase
4. PARCH/EM for hot rod calculation during the pool boiling period

In 1979, the accident at Three Mile Island Unit 2 (TMI-2) occurred. To address findings from the accident for SBLOCA models, NRC issued NUREG-0635 (Reference 3) requesting to address the following concerns related to CE-designed PWRs:

1. Demonstration of acceptability of the condensation heat transfer correlation in the steam generator.
2. Effect of non-condensable gases in condensation heat transfer

3. Justify the conservatism of ECC mixing model as an equilibrium process.
4. Validate program performance with LOFT tests L3-1 and L3-6, and with Semiscale test S-UT-8.
5. Validate the steam generator model.
6. Validate the core heat transfer and the liquid level models.
7. Confirm proper accounting of stored energy in the primary system metal structures.
8. Validate acceptability of 1.0 discharge coefficient on both sub-cooled and saturated break flow models.

Eventually, NRC issued NUREG-0737 (Reference 4) to specify TMI Action Plan Requirements which were approved by the Commission for implementation. Section II.K.3.30 of Enclosure 3 to NUREG-0737, lists requirements for SBLOCA models to continue to comply with the requirements of Appendix K to 10CFR50.

Specifically, II.K.3.30 describes the rationale and issues to address SBLOCA transient calculations as shown in the following statements:

"… review the analytical predictions of feedwater transients and small-break LOCAs for the purpose of assuring the continued safe operation of all operating reactors, including a determination of acceptability of emergency guidelines for operators.

As a result of the task force reviews, a number of concerns were identified regarding the adequacy of certain features of small-break LOCA models, particularly the need to confirm specific model features (e.g., condensation heat transfer rates) against applicable experimental data. These concerns, as they applied to each light-water reactor (LWR) vendor's models, were documented in the task force reports for each LWR vendor. In addition to the modeling concerns identified, the task force also concluded that, in light of the TMI-2 accident, additional systems verification of the small-break LOCA model as required by II.4 of Appendix K to 10 CFR 50 was needed. This included providing predictions of Semiscale Test S-07-10B, LOFT Test (L3-1), and providing experimental verification of the various modes of single-phase and two-phase natural circulation predicted to occur in each vendor's reactor during small-break LOCAs.

Based on the cumulative staff requirements for additional small-break LOCA model verification, including both integral system and separate effects verification, the staff considered model revision as the appropriate method for reflecting any potential upgrading of the analysis methods.

The purpose of the verification was to provide the necessary assurance that the small-break LOCA models were acceptable to calculate the behavior and consequences of small primary system breaks. *The staff believes that this assurance can alternatively be provided, as appropriate, by additional justification of the acceptability of present small-break LOCA models with regard to specific staff concerns and recent test data. Such justification could supplement or supersede the need for model revision".*

The last two sentences in the above quotation were italicized by the author of this response to high light the approach CE took to respond to the requirements. As the last paragraph of Page 1 of the Safety Evaluation Report (SER) (Reference 5) for Reference 6 states, CE elected to justify the continued acceptance of the S1M for small break LOCA evaluation as responses to the II.K.3.30 requirements.

The overall effort to respond to the NRC requests were documented in a series of CEN-203 reports which NRC accepted as shown in CEN-203-P, Revision 1-P-A, Supplements 1-P-A, 2-P-A, 3-A and 4-A (References 6 through 10).

References 6 addressed the following items:

1. Condensation heat transfer including non-condensable gas effect on condensation heat transfer
2. ECC injection modeling
3. Code validation using LOFT L3-1 and L3-6
4. Flow regime influence on pressure drop
5. Core heat transfer model
6. Metal heat transfer
7. Break flow multiplier

Reference 7 provided additional information in the following subjects:

1. Non-equilibrium effect on ECCS performance.
2. Two-phase mixture level and steam cooling heat transfer in core
3. Metal wall heat transfer
4. Break flow multiplier
5. System transient results for System 80

Reference 8 provided further additional information for steam generator modeling in the following subjects:

1. Applicability of Semiscale Test S-UT-8 (Reference 11) to C-E NSSS
2. Impact of liquid holdup in steam generator tubes on CE SBLOCA analyses
3. Steam generator nodalization, condensation heat transfer and multi-tube flooding

Using the information provided in References 6 through 8, the NRC staff found Action Item II.K.3.30 is resolved for CE plants pending acceptable benchmarking against Semiscale Test S-UT-8. Under this condition, the NRC staff concluded that no new plant analyses are required because S1M continued to be applicable without changes. The main justification of continued use of S1M is its conservatism compared to more realistic calculations.

Reference 9 addressed the pending issue stated in the conditional SER for References 6 through 8: “acceptable benchmarking against Semiscale Test S-UT-8.” A special interest of this simulation was to confirm that CEFLASH-4AS could calculate the core level depression before loop seal clearing observed in the test.

CE developed a special code version of CEFLASH-4AS for Semiscale test simulation to account for unique features of the test facility and implemented best-estimate features to make the code behave like a best-estimate (BE) code.

This code calculated results agreed with the test data very well. Then the code was converted to a semi-BE code by reverting BE features important for water hold-up and core uncovery to their original CEFLASH-4AS counterparts. The simulation with this semi-BE code showed that the results were in acceptable agreement with test data. Reference 10 responded to further questions on the analysis reported in Reference 9.

This study showed that the CEFLASH-4AS models which are important to steam generator liquid hold-up and core uncover adequately predicted the vessel liquid level depression observed in S-UT-8. Therefore, the NRC staff concluded that CEFLASH-4AS can acceptably calculate the pre-loop seal clearing core water level depression of S-UT-8.

Finally, the SER (Reference 12) with no condition for S1M stated that the currently approved CE Small Break LOCA Evaluation Model results in conservatively high cladding temperatures for the break spectrum analysis of a Nuclear Steam Supply System (NSSS).

The NRC review of the responses concluded that CE could continue to use the S1M considering the models' substantial conservatism in predicting SBLOCA transients.

Westinghouse had been using the S1M until 1998 when the methodology was revised to the S2M evaluation model and it was accepted by NRC (Reference 13). The S2M has the following three improvements:
1. Improved heat transfer correlation to steam
2. Added radiation heat transfer to steam
3. Improved coupling between the fuel and channel nodes

In addition to these improvements, it replaced the Dougall-Rohsenow correlation for film boiling with the Groenveld correlation. This is in compliance with the Requirement 5.c of 10CFR50 Appendix K for evaluation models approved after October 17, 1988 under the conditions where overall conservatism is reduced by more than 50 °F. It is noted that this requirement is not applicable to S1M.

These updates for the S2M provide additional margin from the S1M. However, it must be noted that S2M does not invalidate S1M as an NRC-approved evaluation model. Results from the S1M are conservative relative to results from the S2M for the same conditions (see Table 3-4 on Page 3-14 and Figure 3-5 on Page 3-15 of Reference 13).

Reference 13 presented the break spectrum analyses with S1M and S2M for a plant which was characterized as a 'Typical ABB CE PWR.' The analyses showed that the S2M core heat transfer model improvements resulted in a substantial reduction in peak cladding temperature for the limiting break. This observation was accepted to be applicable for all CE-designed PWRs because they have the same loop configuration and the changes to S1M are only for core heat transfer models.

References

1. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," August 1974.
2. CENPD-137, Supplement 1-P, "Small Break Model, Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977.
3. NUREG-0635 "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering - Designed Operating Plants," January 1980.

4. NUREG-0737, “Clarification of TMI Action Plan Requirements,” November, 1980.
5. Letter from C. O. Thomas (NRC) to R. W. Wells (CEOG), “Conditional Acceptance for Referencing of Licensing Topical Report CEN-203(P) Rev. 1, “Response to NRC Action Plan Item II.K.3.30 Justification of Small Break LOCA Methods,” June 20, 1985.
6. CEN-203-P, Revision 1-P-A, “Response to NRC Action Plan, Item II.K.3.30, Justification of Small Break LOCA Methods,” March, 1982.
7. CEN-203-P, Revision 1-P, Supplement 1-P-A, “Response to NRC Request Number 1 for Additional Information on C-E Report CEN-203-P, Rev. 1-P (Response to NRC Action Plan, Item II.K.3.30, Justification of Small Break LOCA Methods),” February, 1984.
8. CEN-203-P, Revision 1-P, Supplement 2-P-A, “Further Response to NRC Request Number 1 for Additional Information on C-E Report CEN-203-P, Rev. 1-P (Response to NRC Action Plan, Item II.K.3.30, Justification of Small Break LOCA Methods),” November, 1984.
9. CEN-203-P, Revision 1-P, Supplement 3-A, “Post-Test Analysis of Semiscale Test S-UT-8, Response to NRC’s Conditional SER Issued June 20, 1985 on the Justification of C-E Small Break LOCA Methods,” December, 1985.
10. CEN-203-P, Revision 1-P, Supplement 4-A, “Response to NRC Request for Additional Information for Verification of Analysis Methods for Small Break LOCA’s ,” November, 1986.
11. W. W. Tingle, “Test Data Report on Westinghouse Reactor Vessel Level Indicating System Performance during Semiscale Test S-UT-8,” EGG-SEMI-5827, March 1982.
12. Letter from D. M. Crutchfield (NRC) to J. K. Gasper (CEOG), “Acceptance for Referencing of Licensing Topical Report,” February 11, 1987.
13. CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model,” April 1998.
14. KNF-TR-NFR- 99001, Rev.0, “Evaluation of the Acceptability of ZIRLO Cladding for Korea Standard Nuclear Power Plant Fuel,” February 2001.

Impact on DCD

There is no impact on the DCD.

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 Environment Report.