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:	**15.00.02 – Review of Transient and Accident Analysis Methods**
Application Section:	**15.0.2**
Date of RAI Issue:	**03/08/2016**

Question No. 15.00.02-16

The CE-ABB SBLOCA methodology being used by KHNP was approved for power ratings up to 3800 MWt, per the letter from Parr (NRC) to Stern (CE), dated June 13, 1975. The APR1400 SBLOCA calculations for the APR1400 design are conducted at a power level of 4062.7 MWt (102% of 3983 MWt), which is significantly higher than the 3800 MWt restriction on the CE-ABB SBLOCA methodology. Document the revision of the SBLOCA methodology being used and provide the justification for using the methodology and the computer codes at this higher power level. The justification should include references to other instances where the CE-ABB SBLOCA methodology has been approved by the NRC for higher power levels. The applicant needs to demonstrate the conservatism of the methodology at the APR1400 power level, and establish that there are no phenomenological departures due to APR1400 power level or design differences that would invalidate the use of the SBLOCA methodology or the computer codes approved up to 3800 MWt applications.

Response

The APR1400 Small-Break Loss-of-Coolant Accident (SBLOCA) methodology uses the Supplement 1 Method (S1M) (Reference 5) for a power level of 4062.7 MWt (including uncertainty). The APR1400 design is essentially the same as the Combustion Engineering System 80+ for the SBLOCA transient calculations. The S1M was used for System80+ analysis to obtain its Safety Evaluation Report (SER) (Reference 7). The last paragraph for SBLOCA on Page 15-21 of Reference 7 begins with ‘NRC-approved methods were used to analyze small- and large-break LOCAs:’ The S1M is the NRC-approved method mentioned here. There are no known phenomenological departures or design differences that would invalidate the use of the S1M methodology for APR1400 design. The S1M utilizes the computer codes: CEFLASH-4AS, STRIKIN-II, COMPERC-II, and PARCH-EM.

The Nuclear Regulatory Commission (NRC) SER (Reference 1) on the Combustion Engineering Topical Report (Reference 2 for CEFLASH-4A) has the following constraint:

Power rating up to 3800 MWt

Since the CEFLASH-4AS code was derived from the large break LOCA version (Reference 2), the limitation of 3800 MWt applies to the small break methodology (Reference 5) using the CEFLASH-4AS computer code.

The 3800 MWt limit originated from Regulatory Guide (RG) 1.49 (Reference 3). According to RG 1.49, the power limit was set until sufficient experience is gained with design, construction and operation of large plants. The limitation is not related to any small break LOCA modeling features, plant designs and fuel types. This is evident in the statement in Section 4 of the SER for the Palo Verde Nuclear Generating Station (PVNGS) stretch power license submittal (Reference 4, p. 42). It states that 'the staff has determined that sufficient experience exists with large plants and licensed units in excess of the administrative limit, the proposed amendment complies with RG 1.49.'

Section 4 of Reference 4 also states that 'In SECY 94-025, the staff informed the Commission of the acceptability of power levels above 3800 MWt for certain evolutionary reactor designs. The design of General Electric's advanced boiling water reactor was found acceptable for a power level of 3926 MWt and Asea Brown Boveri Combustion Engineering's System 80+ was found acceptable for a power level of 3914 MWt (as stated in Section 1.2.2 of NUREG-1462 "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," dated August 1, 1994).'

The Reference 2 Topical Report was prepared to support analyses for Combustion Engineering (plants which at the time included Calvert Cliffs Units 1 and 2, Millstone Unit 2, St. Lucie Units 1 and 2, Palisades, Fort Calhoun Station, San Onofre Units 2 and 3, Arkansas One Unit 2, and Waterford Unit 3. The San Onofre Units and Waterford Unit 3 were licensed at a power level of 3458 MWt (including uncertainty) which is the highest power level of the CE plants at that time.

Waterford Unit 3 has since uprated to 3735 MWt (including uncertainty). Thus, the 3800 MWt constraint on Reference 2 application bounded the existing plants at the time it was issued. Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3 have since come on line and have been uprated to a licensed rated power of 4070 MWt (including uncertainty) presently. A precedent has been established by the NRC to accept an Emergency Core Cooling System (ECCS) Analysis for PVNGS Units 1, 2, and 3 at a rated core power greater than 3800 MWt.

The S1M CEFLASH-4AS, which has the power limitation in its SER, does not directly use power level for input. Technical specification cold leg temperature with uncertainty is used and a hot leg temperature is calculated from an enthalpy increase with technical specification low RCS flowrate and power. The enthalpies based on these temperatures are input to the CEFLASH-4AS computer code for initialization. Reactor trip is modeled to occur at the time of the LOCA and power level drops off. The initial power level determines the decay heat at

initialization. The S1M is an Appendix K method which uses a conservative 1.2 multiplier on the 1971 ANS decay heat curve.

Westinghouse revised the methodology from the S1M to the Supplement 2 Method (S2M) evaluation model (Reference 6) in 1997. The S2M made the following changes to S1M:

1. Improved heat transfer correlation to steam
2. Added radiation heat transfer to steam
3. Improved coupling between the fuel and channel nodes
4. Replaced the Dougall-Rohsenow correlation with the Groenveld correlation for film boiling.

These updates for the S2M provided additional margin from the S1M and were approved by the NRC. Results from the S1M are conservative relative to results from the S2M for same plant conditions (see Table 3-4 on Page 3-14 of Reference 6). As noted above, the APR1400 analysis used the S1M.

In summary, the above discussion shows the following three points:

1. The power limit is an administrative limit and it is not related to models in the SBLOCA methodology, specific plant design and fuel. There is no stated power limitations in Reference 1 which are related to specific thermal hydraulic models of the methodology. Therefore, there is no need for model descriptions for this purpose.
2. S1M is an NRC-approved methodology.
3. S1M is conservative compared to S2M.

Based on this discussion, the analysis with S1M for APR1400 at 4062.7 MWt (including uncertainty) is considered to be acceptable.

References

1. O.D. Parr (NRC) to F.M. Stern (C-E), “NRC Staff Review of the Combustion Engineering ECCS Evaluation Model,” June 13, 1975.
2. CENPD-133P, “CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis,” August 1974.
3. Regulatory Guide 1.49, “Power levels of Water-Cooled Nuclear Power Plants,” Revision 1, issued in 1973.
4. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 108 to Facility Operating License No. NPF-41, Amendment No. 100 to Facility Operating License No. NPF-51, and Amendment No. 80 to Facility Operating License No. NPF-74. Arizona Public Service Company, et al. Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3. Docket Nos. STN 50-528, STN 50-529, and STN 50-530 (Accession No. ML021710572).
5. CENPD-137, Supplement 1-P, “Small Break Model, Calculative Methods for the C-E Small Break LOCA Evaluation Model,” January 1977.
6. CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model,” April 1998.
7. NUREG-1462, Vol. 1 and 2, “Final Safety Evaluation Report Related to the Certification of the System 80+ Design Docket No. 52-002,” August 1994.

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.