Attachment 5
Effect of Proposed Revisions on Evaluation Model Results

Introduction

An important consideration with regards to the revisions proposed by Reference 1, is the effect they may have on Evaluation Model results. This section documents the results of several sensitivity studies and estimation of the impact of these model changes and non-conservativisms. The information was derived from several sources including staff assessments, contractor studies, and reports obtained from vendors and the public literature.

The effect of proposed changes can be significantly different depending on break size. Therefore, two tables are provided; one for large break LOCA and another for small break LOCA. Because the large break LOCA analysis frequently limits core power, more work has been performed to investigate the effects of various parameters for that scenario.

Large Break LOCA

Table 1 lists the sensitivity estimates for large break LOCA. For large break, the effect on PCT of decreasing decay heat in the analysis by replacing the 1971 ANS Decay Heat Standard with a more realistic estimate has been found to be worth several hundred degrees F. Existing studies have used the 1979 ANS Standard, which is more conservative than the proposed 1994 ANS Decay Heat Standard. Thus, it is reasonable to expect that analyses performed with the 1971 ANS Standard replaced with the 1994 Standard will result in a large break PCT reduction of nearly 500 °F. (While highly plant dependent, this amount of LOCA margin would allow plant power upratings of 10 to 20%.)

The effect of replacing the Baker-Just correlation with the Cathcart-Pawel correlation for heat release due to clad-steam chemical reaction has been estimated to be less than 100 °F. The estimates assume that the clad temperatures are sufficiently high such that the metal-water reaction rate is important in the calculation (T_{clad} > 1900 °F). Thus, for plants where the peak clad temperatures remain low either due to decay heat model reduction or behavior of the plant itself, the effect of replacing Baker-Just with Cathcart-Pawel is negligible.

Three studies are cited in the table as providing a rough estimate of the non-conservatism associated with neglecting downcomer boiling. The Westinghouse estimate of +400 °F is the difference between the first and second reflood PCTs for a typical 4-loop PWR. The second PCT is the direct result of the downcomer boiling and would not have occurred if the process had been neglected in the Evaluation Model. The +700 °F estimate from the RELAP simulation is considered to be high due to excessively high interfacial drag calculated by that code. Together however, the two estimates suggest that the downcomer boiling effect is significant and may offset any benefit obtained by a reduction in decay heat.

The effect of fuel relocation is highly variable. In the earliest investigation available (by EG&G), experimental results were used to determine the +46 °F increase in PCT. This was used in the initial resolution of Generic Safety Issue 92 (Fuel Crumbling During LOCA). More recent work performed in France [9] however suggests the effect to be significantly larger and a strong function of the packing fraction that occurs when the fuel relocates. It is likely that new test data will be needed to fully resolve the issue.

There are several estimates of code uncertainty. In these estimates, realistic codes were used, which are considered to be significantly more accurate than Appendix K based Evaluation Models. From Reference 10 an estimate can be made of the code uncertainty by comparing the 95th and 50th percentile estimates of the PCT. For the case examined it was +340 °F. It is important to note however that the uncertainty based on a difference between a 99th (or 100th) and 50th percentile PCT is significantly larger. Since an Appendix K calculation is intended to bound all possible conditions, then the uncertainty based on the 99th and 50th percentile PCTs is more appropriate for an Appendix K code. In this case, the code uncertainty would be much larger than +340 °F.

The other estimates of code uncertainty [11, 12, 13] are similar to that in Reference 10. In some cases, the exact value is not listed in Table 1 in order to insure that proprietary information is not disclosed in this document. The precise values can be obtained in the References cited. The important point is that the magnitude of the uncertainties are large, and are comparable to the reduction in PCT obtained by decay heat relaxation. Thus, the "excess" conservatism in the 1971 ANS Decay Heat Standard is seen to compensate for code deficiencies.

Table 1: Large Break LOCA ΔPCT Estimates

Process	ΔPCT	Basis/Comments
Decay Heat	-260 to -450 °F	Recent Westinghouse estimate based on App. K EM calculations [2]. ANS 1971 + 1.20 replaced with ANS 1979 + 2σ. Calculations performed using BASH-EM.
Decay Heat	-372 °F	NRC contractor RELAP calculations for CE 2700 MWt (Millstone 2) plant [3]. ANS 1971 + 1.20 replaced with ANS 1979 + 2σ.
Decay Heat	-460 °F	1984 Westinghouse study on Appendix K relaxation [4] .
Metal Water Reaction	-45 to -55 °F	Recent Westinghouse estimate assuming the Baker-Just correlation is replaced with Cathcart-Pawel for metal-water reaction heat [2]. Calculations performed using BASH-EM.
Metal Water Reaction	-75 °F	NRC contractor RELAP calculations with Baker-Just replaced by Cathcart-Pawel [3].
Metal Water Reaction	-65 °F	1984 Westinghouse study on Appendix K relaxation [4] .
Downcomer Boiling	+400 °F	Westinghouse estimate from Best Estimate EM calculations for W 4-loop PWR [5].
Downcomer Boiling	+810 °F	NRC contractor calculations using RELAP5 for a CE System 80+ (3800 MWt) unit [6].
Downcomer Boiling + Reflood Bypass	+63 °F	Estimate based on WCOBRA/TRAC calculations for an uprated CE System 80+ unit [7]. Both downcomer boiling and ECC bypass during reflood were found to be important and contribute to increases in PCT.
Fuel Relocation	+46 °F	EG&G estimate based on experimental tests in PBF (Power Burst Facility) to address Generic Safety Issue (GSI) 92 [8].
Fuel Relocation	+313 °F	Results reported in technical paper by IPSN [9] using CATHARE for a Framatome PWR (similar to a Westinghouse 3-loop PWR). A burst zone 70% filling fraction assumed.
Code Uncertainty	+340 °F	W ΔPCT between 95th and 50th percentile uncertainty in a W 4-loop PWR for WCOBRA/TRAC calculation [10].
Code Uncertainty	+300 °F	Difference between the 95th and 50th percentile PCTs for a Westinghouse RESAR-3S plant using TRAC-PF1/MOD1 [11].
Code Uncertainty	> +275 °F	Framatome ANP large break code uncertainty using realistic version of RELAP [12]
Code Uncertainty	> +400 °F	GE code uncertainty using SAFER/GESTER [13]

Small Break LOCA

There has been relatively little work investigating the effect of the proposed changes, with the exception of decay heat. Table 2 lists estimates of the effect of several parameters on small break LOCA PCTs.

Available results show a very large reduction in PCT if the ANS 1971 Decay Heat Standard is replaced with the 1979 Standard. (The 1979 Standard is slightly more conservative than the 1994 Standard.) Reductions of a least 500 oF can be expected if the 1971 Standard is replaced by a more realistic estimate of decay heat in a small break LOCA analysis.

Because small break LOCA does not generally determine the core operating power, there has not been significant work done to determine code uncertainty to small break processes. Table 2 lists sensitivities obtained through communications with NRC staff and contractors.

Table 2: Small Break LOCA ΔPCT Estimates

Process	ΔPCT	Basis
Decay Heat	- 1000 °F	NRC contractor citation of CE sensitivity to decay heat using CE EM for CE 2700 MWt (Millstone 2) plant [3].
Decay Heat	- 859 °F	NRC contractor citation of W sensitivity EM to decay heat standard for CE 2700 MWt (Millstone 2) plant [3].
Decay Heat	-500 to -1000 °F	NRC contractor estimate based on RELAP5 calculations for typical plants [3].
Decay Heat + Metal Water Reaction	-500 °F	Calculations performed using a SBLOCA version of WCOBRA/TRAC for Indian Point Unit 2 [14]. The ΔPCT is the difference between the limiting SBLOCA case in the paper and current plant (Appendix K based) analysis of record.
Metal Water Reaction	-11 to -76 °F	NRC calculations using RELAP with Baker-Just replaced by Cathcart-Pawel.
Nodalization	+600 °F	NRC RELAP calculations w and w/o crossflow for CE 2700 MWt plant.
Fuel Relocation	Not known	Clad swell and rupture and fuel relocation may occur in SBLOCA. However, no calculations have been found documenting the effect.
Operator Action	+ several 100 °F	Pump trip with off site power available depends on operator recognition and adherence to EOPs. This is a known post-TMI pump trip issue. Trip at inopportune time can cause deep uncovery.
Level Swell Uncertainty	+ several 100 °F	NRC contractor (verbal) estimate. Mixture level swell (code interfacial drag) is highly ranked PIRT process.
Loop Seal Clearance	+/- several 100 °F	Affects pressure drop through loop(s) and core level depression.

References

[1] USNRC, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)," SECY-01-0133, July 23, 2001.

[2] DiMuzio, C. A., Westinghouse Electric Corp., "BASH-EM Parametric Study Results," Presentation to USNRC, June 28, 2001.

[3] Ward, L. W., Palazov, V., Prelewicz, D., and Lauben, G. N., "Assessment of Proposed Modifications to Appendix K of 10 CFR 50.46," ISL-NRC-403-00, Dec. 2000.

[4] Cadek, F. F., Gresham, J. A., Hochreiter, L. E., and McIntyre, B. A., "Potential Thermal Margin Available From Changes in the Appendix K Rule," *Proc. of the International Nuclear Power Plant Topical Meeting*, Taipei, Taiwan, 1984.

[5] Nissley, M. E., Westinghouse Electric Corp., "Comparison of Best-Estimate Plus Uncertainty and Appendix K LBLOCA Analyses," Presentation to USNRC, June 28, 2001.

[6] Palazov, V. V., and Ward, L. W., "A System 80+ RELAP5/MOD3 Model for Downcomer Boiling Following a Large Break LOCA," ISL-NSAD-NRC-01-009, Jan. 2002.

[7] Pottorf, J., "Realistic Simulation of a Loss of Coolant Accident in a Combustion Engineering System 80/80+ PWR," M.S. Thesis, Kansas State University, 2001.

[8] Emrit, R., Riggs, R., Milstead, W., Pittman, J., and Vandermoledn, H., "A Prioritization of Generic Safety Issues," NUREG-0933, June 2000.

[9] Grandjean, C., et al. "High Burnup UO2 Fuel LOCA Calculations to Evaluate the Possible Impact of Fuel Relocation After Burst," OECD Topical Meeting on LOCA Fuel Safety Criteria, Aix-en-Provence, March 2001.

[10] Nissley, M. E., Westinghouse Electric Corp., "Comparison of Best-Estimate Plus Uncertainty and Appendix K LBLOCA Analyses," Presentation to USNRC, June 28, 2001.

[11] Technical Program Group (TPG), "Quantifying Reactor Safety Margins," NUREG/CR-5249, Dec. 1989.

[12] Framatome ANP, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," EMF-2103(P), Rev. 0, August, 2001. (Proprietary).

[13] Hamon, D. A., "Duane Arnold Energy Center SAFER/GESTR - LOCA Loss of Coolant Accident Analysis," NEDC-31310P, August 1986 (Proprietary).

[14] Bajorek, S. M., et al.,"Realistic Small- and Intermediate-Break Loss-of-Coolant Accident Analysis Using WCOBRA/TRAC," *Nuclear Technology*, Vol. 136, Oct. 2001.