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Subject: Arkansas Nuclear One - Units 1 and 2
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6
Errors or Changes in the Emergency Core Cooling
System Evaluation Model; Annual Report for 1999

Gentlemen:

10CFR50.46(a)(3)(ii) requires licensees to report annually each change to or error discovered in an acceptable evaluation model or in the application of such model for the emergency core cooling system (ECCS) that affects the peak cladding temperature (PCT). Included in the submittal is the estimated effect these changes or errors have on the limiting ECCS analysis. The purpose of this submittal is to provide that required information for Arkansas Nuclear One (ANO).

For ANO-1, the CRAFT2-based evaluation model is the current licensing basis; however, information on the RELAP5/MOD2-Babcock and Wilcox (B&W) evaluation model is also presented herein. The RELAP5/MOD2-B&W-based model was approved for generic use as documented in BAW-10192P-A, dated June 1998. Information related to the RELAP5/MOD2-B&W-based loss-of-coolant accident (LOCA) evaluation model is presented for information only, since CRAFT2 is the ANO-1 licensing basis.

ANO-1 CRAFT2 MODEL: There were no errors or changes to the CRAFT2 ECCS evaluation model or the application of this model that resulted in an increase in the PCT or non-conformance to the criteria set forth in 10CFR50.46(b).

RELAP5/MOD2-B&W Model (for ANO-1): A model change was made regarding the inclusion of the zirconium-based M5 alloy cladding (which is not in use at ANO-1) described in BAW-10227P. Additionally, sensitivity studies performed in response to Preliminary Safety Concern (PSC) 1-99 determined that the limiting reactor coolant pump degradation parameters reported in the evaluation model (based on 205-fuel assembly raised-loop studies)

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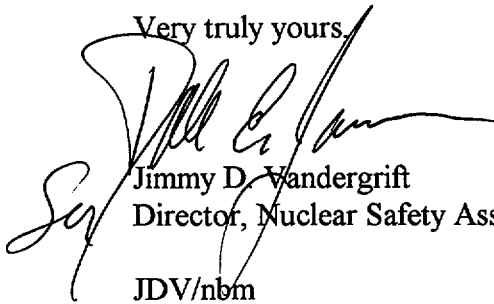
were inconsistent for the 177-fuel assembly plants. A more detailed discussion of the reactor coolant pump two-phase degradation is provided in Appendix A.

Also, there were two input errors in ECCS analyses using the RELAP5/MOD2-B&W-based evaluation model that resulted in a significant changes in the calculated PCTs. The first error is related to inputs for the limiting reactor coolant pump type and associated two-phase degradation for the 177-fuel assembly lowered-loop plants. The second error is related to inputs for the fuel assembly grid blockage droplet breakup factors for the ANO-1 20 percent tube plugging large break LOCA (LBLOCA) analyses. A more detailed discussion of these input errors is provided in Appendix B.

Finally, a brief summary of additional RELAP/MOD2-B&W-based evaluation model analyses and evaluations is included in Appendix C. These changes are not specifically required to be reported under 10CFR50.46, but are provided for completeness to changes in the RELAP/MOD2 model. These changes include: moderator temperature coefficient evaluations for end-of-cycle T_{AVE} reductions, screening criteria to show compliance with the LOCA evaluation model restriction on axial peaking, and additional justification for the applicability of the BEACH code for cladding temperatures above 1640°F.

ANO-2: For ANO-2, there were no errors or changes to the ABB-CE ECCS evaluation model or the application of this model that resulted in an increase in the PCT or non-conformance to additional criteria set forth in 10CFR50.46(b).

Very truly yours,



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Appendix A – Evaluation Model Changes and Significant Error Notification

The ANO-1 licensing basis for LOCA is based on CRAFT2-based evaluation models. No significant errors were reported in the CRAFT2-based B&W ECCS evaluation model, BAW-10104PA, Rev. 5 for LBLOCA and BAW-10154A, Rev. 0 for SBLOCA, during 1999. No significant errors were reported in the RELAP5/MOD2-B&W-based LOCA evaluation model (BAW-10192PA, Rev. 0) during 1999. However, two changes related to the RELAP5/MOD2-B&W-based LOCA evaluation model have been evaluated during the reporting period. The changes pertain to ECCS analyses based on the M5 alloy cladding type (which is not in use at ANO-1) and the limiting reactor coolant pump two-phase degradation. The RELAP5/MOD2-B&W information is presented for information only.

Sensitivity studies performed on the 177-fuel assembly plant revealed that the limiting reactor coolant pump degradation parameters reported in the evaluation model based on 205-fuel assembly raised-loop studies would not produce limiting PCT results. This issue was reported as PSC 1-99 (Reference A-1). Table 9-2 of Volume I of the evaluation model must be modified in the future to indicate that the limiting two-phase degradation should be determined by plant-specific sensitivity studies. Studies summarized in Reference A-2 determined that the minimum pump degradation (modeled by the M1 multiplier and the two-phase difference curves from RELAP5) provided conservative results for the 177-fuel assembly plants. It was determined that the PCTs increased by greater than 50°F. However, they remained below the 10CFR50.46(b)(i) limit of 2200°F. Finally, it was concluded that PSC 1-99 did not constitute a significant safety hazard and was not reportable under 10CFR21.

References

- A-1. Letter, J. J. Kelly to USNRC, "Report of Preliminary Safety Concern Related to Use of an Inappropriate RCP Two-Phase Degradation Model," March 5, 1999.
- A-2. FTI Document 51-006132-00, "PSC 1-99 Resolution," January 1999.

Appendix B – ECCS Analysis Input Errors

The ANO-1 licensing basis for LOCA is based on CRAFT2-based evaluation models. Two input errors have been discovered in ECCS analyses using the RELAP5/MOD2-B&W-based evaluation model in 1999. The RELAP5/MOD2-B&W information is presented for information only.

Some ECCS analyses performed with the LOCA evaluation model underpredicted the PCT because the most limiting reactor coolant pump type and two-phase degradation model were not used. Upon reanalysis with the most limiting RCP parameters, the calculated PCT increase was more than 50°F. In compliance with 10CFR50.46(a)(3), the NRC has previously been notified (Reference B-1) of this “significant” error in an input to the RELAP5/MOD2-B&W code, which is part of the LOCA evaluation model (BAW-10192PA, Rev. 0). The LBLOCA cases affected by the erroneous pump type and non-conservative two-phase pump degradation model have been identified, and the cases that presented the most serious challenges to the 10CFR50.46 limits have been reanalyzed. The affected cases were applicable to ANO-1. Reanalyzed cases were used to develop PCT deltas that were applied to the non-limiting core elevations or times-in-life. Reference B-2 summarizes the cumulative results of the reactor coolant pump reanalyses and associated PCT deltas or linear heat rate (LHR) adjustments for the affected plants. When reanalyzed or reevaluated, the PCTs for the affected analyses were less than the 10CFR50.46(b)(i) limit of 2200°F. However, to maintain the final PCTs within the desired 1950°F to 2050°F range, LHR reductions of 0.3kW/ft and 0.5kW/ft were imposed at the 4.264 ft and 6.021 ft elevations, respectively.

The B&W Owners Group 20 percent steam generator tube plugging LBLOCA analysis for ANO-1 underpredicted the PCTs because of an input error affecting the hot pin heat removal during the reflood phase. Specifically, the grid blockage factors were input incorrectly, i.e., a mixing vane grid input was modeled instead of the correct non-mixing vane grid input. In compliance with 10CFR50.46(a)(3), the NRC has previously been notified (Reference B-3) of this “significant” error in an input to the BEACH code, which is part of the LOCA evaluation model (BAW-10192PA, Rev. 0).

Reanalyses of the bounding 20 percent steam generator tube plugging LBLOCA cases with corrected grid input parameters resulted in a calculated PCT increase in excess of 50°F. However, when reanalyzed and reevaluated, the PCTs for the affected analyses were less than the 10CFR50.46(b)(i) limit of 2200°F. The maximum PCT increase due to the grid error was 165°F at the 9.536 ft elevation and decreased by 13°F at the 2.506 ft elevation. The middle-of-life and end-of-life PCTs were conservatively increased by the amount determined for the beginning-of-life analyses, since the lower PCTs at middle-of-life and end-of-life would generate lower PCT increases in response to the corrected grid input data.

References

- B-1. Letter, J. J. Kelly to USNRC, "10CFR50.46 Thirty Day Report on Significant PCT Change in ECCS Analyses," February 4, 1999, OG-1740.
- B-2. FTI Document 51-5006132-00, "PSC 1-99 Resolution," January 1999.
- B-3. Letter, J. J. Kelly to USNRC, "10CFR50.46 Thirty Day Report on Significant PCT Change in ECCS Analyses," May 25, 1999, FTI-99-1727.

Appendix C – Additional Analyses and Evaluations

The following information is related to the RELAP5/MOD2-B&W-based LOCA evaluation model and is presented for information only.

End-of-Cycle T_{AVE} Reduction Maneuver Analysis for LOCA Applications

Analyses for the end-of-cycle average reactor coolant system temperature (T_{AVE}) reduction maneuver were completed to provide a new bounding (negative) moderator temperature coefficient (MTC) value that will limit the calculated LOCA consequences, such that the reduced T_{AVE} results are bounded by the nominal T_{AVE} LOCA results. Negative MTC curves conservative for Mark-B11, Mark-B10-OL, Mark-B10K, and Mark-B9 fuel were developed at values of -10, -15, -20, -25 and -30 pcm/F. These MTC curves are appropriate for both the 177-fuel assembly lowered-loop and 177-fuel assembly raised-loop B&W plant designs. The LOCA analyses (References C-1 and C-2) iterated on the MTC curves to find the least negative MTC that provided results that were bounded by the calculated nominal T_{AVE} consequences. The outcome of the analyses determined that a -10 pcm/F MTC limit curve must be adhered to such that no LHR limit penalties are necessary to accommodate the EOC T_{AVE} maneuver. These results are applicable to Mark-B fuel types in the 177-fuel assembly cores, which are licensed with the LOCA evaluation model (BAW-10192PA, Rev. 0).

Axial vs. Radial Core Peaking Factors

The RELAP5/MOD2-B&W-based LOCA analyses are performed with a core axial peaking factor of 1.7, as outlined in the LOCA evaluation model. At the time the LOCA evaluation model was approved, modeling of a core axial peaking factor of 1.7 acceptably represented the actual axial and radial core peaking factors seen in the core power distribution analyses for B&W plants. However, the third restriction of the NRC Safety Evaluation Report (SER) on the LOCA evaluation model (BAW-10192AP Rev. 0) states that Framatome Technologies, Inc. (FTI) must revalidate the acceptability of the evaluation model peaking methods if: (1) significant changes are found in the core elevation at which the minimum core LOCA margin is predicted or (2) the core power distribution analyses radial and axial peaks that approach the LOCA LHR limit differ appreciably from those used to demonstrate Appendix K compliance.

Separate screening criteria needed to show compliance with the LOCA evaluation model restriction on peaking were developed for both the B&W lowered-loop and raised-loop plant designs (References C-3 and C-2, respectively). The methods are valid for any current or past Mark-B fuel type (including but not limited to Mark-B4Z, Mark-B8, Mark-B9, Mark-B10(F), Mark-B11(M5)) that is ruptured-node limited or has similar ruptured-node or unruptured-node PCTs predicted with the LOCA evaluation model. The methods either validate compliance with the restriction or define a LOCA LHR limit penalty associated with LHR limits calculated based on the LOCA evaluation model to verify that the limiting LOCA consequences are predicted.

LOCA Evaluation Model Limits and Restrictions, BAW 10192PA

The development of an FTI document to verify compliance with the limitations and restrictions contained in the LOCA evaluation model (BAW-10192PA, Rev. 0) was completed near the end of 1998 (Reference C-5) and was summarized in the 1999 annual report. As part of the SER on Revision 2 of the BEACH code (Reference C-6), the applicable range of initial cladding temperature is specified as 950°F to 1640°F. Additional justification of the applicable ranges for which the empirical constants on the revised grid and rupture models are appropriate was provided in Appendix A of that document.

References

- C-1. FTI Document 86-5006590-00, "BWOG TAVE Reduction LOCA Summary", January 2000.
- C-2. FTI Document 86-5006232-00, "Davis-Besse LOCA Summary", February 2000.
- C-3. FTI Document 51-5004541-00, "Radial vs Axial Core Peaking for LOCA", June, 1999.
- C-4. FTI Document 51-5006132-00, "PSC 1-99 Resolution", January 2000.
- C-5. FTI Document 51-5001731-00, "BWNT LOCA EM Limitations and Restrictions", December 1998.
- C-6. N. H. Shah, et al., "BEACH – A Computer Program for Reflood Heat Transfer During LOCA," BAW-10166P-A, Revision 4, B&W Nuclear Technologies, Lynchburg, Virginia, October 1992.