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U. S. Nuclear Regulatory Commission
Document Control Desk
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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 2000

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 the reporting period January 1, 2000, through December 31, 2000.

ANO-1: The ANO-1 licensing basis for the year 2000 was the CRAFT2-based evaluation model (EM). For ANO-1, there were no significant errors or changes that resulted in an increase in the PCT or non-conformance to the criteria set forth in 10CFR50.46(b) in the CRAFT2-based Babcock and Wilcox (B&W) Emergency Core Cooling System (ECCS) EM. Also, during 2000, there were no input errors detected that changed the results of completed LOCA analyses.

In addition for information only, Entergy Operations is providing a description of the analyses and evaluations that have been completed during the reporting period for the RELAP5/MOD2 EM. Attachment A presents B&W plant generic items, and Attachment B presents the ANO-1 specific evaluations performed for 2000 using the B&W-approved LOCA EM(s).

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).

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This submittal contains no commitments. Should you have any questions regarding this submittal, please contact me.

Very truly yours,

Jimmy D. Vandergrift
Director, Nuclear Safety Assurance

JDV/nbm
Attachments

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Attachment A
B&W 177-FA Plant Generic Items

A.1 PSC 2-00

Preliminary Safety Concern (PSC) 2-00 was initiated by Framatome Technologies on July 28, 2000. It identified that the calculated consequences for a postulated core flood tank (CFT) line break for the B&W-designed plants could be worse if offsite power were available, and credit for operators tripping the reactor coolant pumps (RCPs) was performed at two minutes after loss of subcooling margin (LSCM). The NRC was informed via letter (Ref. 2) on September 26, 2000.

The CFT line break has historically been analyzed for the B&W-designed plants with a loss-of-offsite power (LOOP) at the time of reactor trip. The worst single failure following LOOP is generally a loss of an emergency diesel generator, such that a single high pressure injection (HPI) and low pressure injection (LPI) pump are initially unpowered. A single operating LPI pump and valve arrangement that results in all the LPI flowing to only one CFT line, which is assumed to be the broken line, leaves the event to be mitigated in the short term by the flow from one HPI pump and one intact CFT. This ECCS flow is sufficient, with the residual reactor vessel inventory from early RCP trip, to adequately cool the core. The minimum core mixture level generally remains near or above the top of the core with typical PCTs less than 800°F for this break with an immediate loss-of-offsite power.

If offsite power is available, the operators are instructed by the emergency operating procedures to manually trip the RCPs immediately following LSCM. Historical CRAFT2 analyses credited RCP trip at two minutes following LSCM. When the RCP trip is delayed by two minutes, the continued forced circulation in the RCS causes more RCS liquid to flow out the break, thereby decreasing the liquid inventory that remains in the reactor vessel. This reduced vessel inventory, with the ECCS flow from a single CFT and one HPI pump, results in additional core uncovering with higher cladding temperature excursions.

Analyses, performed with RELAP5/MOD2 using the NRC-approved EM reported in BAW-10192P-A (Ref. 6), predicted significant PCT increases for several of the 177-FA lowered-loop plants when the reactor coolant pumps are powered for the first two minutes following the loss of subcooling margin. More significantly, sensitivity studies showed that the calculated consequences are highly dependent upon the modeling of RCP performance under two-phase flow conditions. The severity of the predicted cladding temperature excursions is directly tied to the extent that pump head performance is degraded during two-phase flow. Increased degradation reduces the amount of liquid inventory lost through the break. Conversely, less degradation will increase inventory loss, with a significant adverse impact upon predicted PCT.

The NRC was informed via letter (Ref. 4) on December 20, 2000 that the analyses in support of the PSC 2-00 resolution were not going to be completed by the end of 2000. The results were reported in the final summary report for PSC 2-00 (March 30, 2001).

A.2 Stainless Steel Rod Evaluation

Some irradiated fuel assemblies may contain fuel rods that are not suitable for use in subsequent fuel cycles. Replacement of these fuel rods with non-heat producing stainless steel rods has been demonstrated to be an acceptable action (Ref. 5). The use of solid non-heat producing rods or fuel rods with naturally enriched uranium allows the modified fuel assemblies to be utilized in subsequent cycles. Reference 3 evaluates the affect on the results of a LOCA analysis of up to 10 solid stainless steel or natural uranium fuel pins per assembly, with a maximum of 200 total replacement rods in the core. The affect of the replacement rods on the initial stored energy, heat transfer and swell/rupture flow blockage was considered and the affect on the LOCA transient was evaluated. The results of this evaluation are generically applicable to all B&W plants.

A.3 Revision 1 to EM Limitations and Restrictions Document

The EM limitations and restrictions document (Ref. 1) discusses all limitations and restrictions placed on the BWNT LOCA EM and all associated computer codes. Additionally, it provides checklist style tables from which compliance to the limitations and restrictions on the LOCA input parameters and acceptable ranges of application are verified. The revision adds information related to the recently approved $M5_{TM}$ topical report, and PSC 1-99 modeling updates for RCP type and two-phase degradation. Also included in this revision was a description of the void-dependent cross-flow model option, additional information on the latest energy deposition factor methods and sample-input data related to grid modeling.

References

1. FTI Document 51-5001731-01, "BWNT LOCA EM Limitations and Restrictions," April 2000.
2. Letter, J.J. Kelly to USNRC Document Control Desk, FTI-00-2433, September 26, 2000.
3. FTI Document 51-5010065-00, "LOCA Evaluation of Stainless Steel and Natural UO_2 Replacement Rods," October 2000.
4. Letter, D.J. Firth to USNRC Document Control Desk, FTI-00-3085, December 20, 2000.
5. FCF Topical Report BAW-2149-A, "Stainless Steel Replacement Rod Methodology", September 1993.
6. FTI Topical Report BAW-101902P-A, "BWNT LOCA-BWNT Loss of Coolant Accident Evaluation Model for Once Through Steam Generator Plants," June 1998.

Attachment B
Entergy Operations (ANO-1) Specific Information

B.1 ANO-1 Mark-B9 RELAP5 LOCA Limits adjusted for current CFT Specifications

Entergy Operations requested that the B&W Owners Group (BWOG) 20% steam generator tube plugging LBLOCA analyses be reevaluated to support the current ANO-1 CFT technical specifications in support of Cycle 17. The Cycle 17 maneuvering analyses were performed in parallel to the LOCA calculations and assumed a 1 kW/ft LHR penalty on the base 20% tube plugging results through the middle-of-life of operation. This was considered a conservative estimate, and the justification analyses were completed in 2001.

B.2 ANO-1 Mark-B9 CRAFT2 LOCA Limits reduced for maneuvering analysis for Cycle 17

Entergy requested that conservative CRAFT2-based LOCA LHR limits be used for the Cycle 17 maneuvering analyses to support the possibility of switching to the RELAP5-based EM for reference in licensing calculations prior to or during Cycle 17 operation. The evaluation provided a conservative set of LOCA LHR limits, however the previous CRAFT2 LOCA limits and PCTs are still acceptable.