FirstEnergy

Davis-Besse Nuclear Power Station
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United States Nuclear Regulatory Commission
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Subject: Annual Report of Changes to the Emergency Core Cooling System Evaluation Model In Accordance With 10 CFR 50.46(a)(3)

Ladies and Gentlemen:

In accordance with 10 CFR 50.46(a)(3), the FirstEnergy Nuclear Operating Company (FENOC) herewith submits the attached annual report of the Emergency Core Cooling System (ECCS) Evaluation Model (EM) used at the Davis-Besse Nuclear Power Station (DBNPS). No new changes or errors to the EM occurred for the period of January 1, 2001 to December 31, 2001.

If you have any questions or require additional information, please contact Mr. Patrick J. McCloskey, Manager, Regulatory Affairs, at (419) 321-8450.

Very truly yours,

Lew W. Myers
Vice President - Nuclear
Davis-Besse Nuclear Power Station

AWB/s

Attachments

A001

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station in this document. Any other actions discussed in the submittal represent intended or planned actions by Davis-Besse. They are described only as information and are not regulatory commitments. Please notify the Manager – Regulatory Affairs (419-321-8450) at Davis-Besse of any questions regarding this document or associated regulatory commitments.

COMMITMENTS	DUE DATE
None	N/A

Annual Report of Changes to the 10 CFR 50.46 Emergency Core Cooling System Evaluation Model for the Davis-Besse Nuclear Power Station

10 CFR 50.46 (a)(3) states that each holder of an operating license shall report to the Nuclear Regulatory Commission (NRC) at least annually each change or error in an acceptable Emergency Core Cooling System (ECCS) Evaluation Model (EM) or in the application of such a model that affects the Peak Cladding Temperature (PCT) calculation.

For the period January 1, 2001 through December 31, 2001, no significant changes or errors (which resulted in a PCT difference greater than 50 degrees Fahrenheit (F)) were identified in the RELAP5 based EM that was used for licensing analysis for the Davis-Besse Nuclear Power Station (DBNPS). Specifically, no significant changes or errors were discovered in the RELAP5/MOD2-B&W-based Babcock & Wilcox Nuclear Technologies (BWNT) Loss of Coolant Accident (LOCA) EM, BAW-10192P-A, Revision 0 (Reference 1) that covers both Large Break (LB) LOCA and Small Break (SB) LOCA including applications with $M5_{TM}$ fuel cladding as described in BAW-10227P-A (Reference 2). Also, no input errors were detected that changed the results of the most limiting SBLOCA or LBLOCA analysis.

For the period January 1, 2001 to December 31, 2001 no new changes or errors are reported by the DBNPS. However, previously reported items that were included in the NSSS vendor annual report (Reference 3) are included to maintain consistency with the NSSS vendor letter.

<u>RELAP5/MOD2-B&W Once Through Steam Generator Large and Small Break LOCA Evaluation Model</u>

This model is applicable to all B&W designed pressurized water reactors for large and small break LOCA analyses. The NRC approved topical report for this evaluation model is BAW-10192P-A Revision 0 (Reference 1).

The large break LOCA Evaluation Model consists of four computer codes: (1) RELAP5/MOD2-B&W to compute the system, core, and hot rod response during blowdown, (2) REFLOD3B to calculate the time for refill of the lower plenum and core reflood rate, (3) CONTEMPT to compute the containment pressure response, and (4) BEACH (RELAP5/MOD2-B&W reflood heat transfer package) to determine the hot pin thermal response during refill and reflood phases. The small break LOCA Evaluation Model consists of two codes: (1) RELAP5/MOD2-B&W to compute the system, core, and hot rod response during the transient and (2) CONTEMPT to compute the containment pressure response, if needed. An NRC-approved fuel code (currently TACO3) is used to supply the fuel rod steady-state conditions at the beginning of the small or large break LOCA.

Correction of RCP Two-Phase Degradation Model for SBLOCA

The NRC-approved SBLOCA EM (Volume II of Reference 1) prescribes that the two-phase RELAP5 head difference and degradation multipliers, derived from the semi-scale pump tests, be used with the reactor coolant pump (RCP) performance curves. Examination of the semi-scale pump degradation curves, which are based upon tests run at relatively low pressures, indicate that the RELAP5 model can over-predict the amount of head degradation during the first several minutes of a SBLOCA transient with continued RCP operation (as analyzed in resolution of Preliminary Safety Concern (PSC) 2-00). Comparison of the RELAP5 curves to representative data, specifically the CE 1/5-scale steam-water tests (which were run at higher pressures) confirm that the RELAP5 model over-predicts pump head degradation during two-phase flow early in the event. Since less pump degradation results in additional RCS liquid loss that leads to more extensive core uncovering and higher PCTs, the approved RELAP5 model cannot be judged to be conservative for application to continued RCP operation during a SBLOCA. When a bounding pump performance curve (the lower bound "M3-modified" curves in the approved large break LOCA model) is modeled, the predicted consequences are much more severe. Therefore, the SBLOCA EM must be corrected to specify that the selection of a RCP two-phase degradation model in future SBLOCA analyses must be justified by sensitivity studies. This approach, which is similar to that used for LBLOCA applications, performs or makes reference to applicable studies that determined which RCP degradation model is conservative for application to plant-specific SBLOCA analyses.

RCP degradation model sensitivity studies were performed to resolve PSC 2-00 with RCP trip at two minutes after loss of subcooling margin. It was concluded from these analysis that the consequences for the Core Flood Tank (CFT) line break are most influenced by this change in degradation. CFT line break sensitivity studies confirmed that the M3-modified curve produces the most conservative results.

For the resolution of this issue at the DBNPS, the large SBLOCAs were analyzed with manual RCP trip at 2 minutes following Loss of Sub-Cooling Margin (Reference 5). The larger cold leg pump discharge breaks predicted more severe consequences than the case with loss of offsite power, however they did not challenge the limiting SBLOCA PCT of 1428 degrees F. The cases were non-limiting compared to other B&W designed plants because the DBNPS utilizes two, high-volume/low-head, high pressure injection (HPI) pumps that each have an equivalent flow capacity of two of the HPI pumps in operation at the other B&W units. Therefore, the PCT change for PSC 2-00 is 0 degrees F.

BEACH Initial Cladding Temperature Range

This item was previously reported to the NRC (Reference 4), and is being resolved by a revision to the BEACH topical report (Reference 6) that was submitted to the NRC in December of 2001. The revision (Reference 7) seeks to extend ranges on the Safety Evaluation Report (SER) restrictions currently in place on the use of BEACH.

Appendix H was added to the BEACH topical to extend the limitation on the cladding temperature at the onset of reflood. The BEACH SER restriction on Revision 2 limited the cladding temperature to between 950 and 1640 F. All of the demonstration cases presented in the BWNT LOCA EM (Reference 1) and subsequently approved by the NRC had cladding temperatures that exceeded this range. Framatome ANP completed an additional benchmark to assure that the code application could be extended to higher cladding temperatures. This benchmark (Full Length Emergency Core Cooling Heat Transfer - Separate Effects Tests And System Effects Tests Test 34420) provides additional confirmation that the mechanistic modeling of BEACH is adequate and acceptable for reflood heat transfer prediction for plant LOCA application temperatures to 2045 degrees F. This material was provided to the DBNPS for 10 CFR 50.46 reporting for calendar year 1999. After the NRC was informed that the BWNT LOCA EM demonstration cases were outside of this initial temperature range, they stated that this was an error and they requested that a formal revision to the BEACH code topical report be prepared and submitted for review and approval (Reference 7). The extension of the SER restriction on the initial cladding temperatures effectively corrects all application analyses that may have had temperatures outside of the initially approved range.

The PCT change for increasing the acceptable range of initial cladding temperatures in BEACH analyses is 0 degrees F for the DBNPS because no alteration of the code results or formulation was required.

References

1. FTI Topical Report BAW-10192P-A, Rev. 0 "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," June 1998.
2. FTI Topical Report BAW-10227P-A, Rev. 0, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000.
3. Letter to USNRC, "2001 – Annual Reporting of Changes and Errors in ECCS Evaluation Models," Framatome ANP NRC:02:012, February 28, 2002.
4. Letter to USNRC, "Annual Report of Changes to the Emergency Core Cooling System Evaluation Model in accordance with 10 CFR 50.46(a)(3)(ii)," Davis-Besse Serial Number 2716, July 5, 2001.
5. FRA-ANP Document 51-5009856-00, "Summary of PSC 2-00 Analyses," 4/13/01
6. FTI Topical Report BAW-10166PA-04. "BEACH – Best Estimate Analysis Core Heat Transfer; A Computer Program for Reflood Heat Transfer During LOCA," February 1996.
7. FTI Topical Report BAW-10166P-05, "BEACH – Best Estimate Analysis Core Heat Transfer; A Computer Program for Reflood Heat Transfer During LOCA," December, 2001.