10 CFR 50.46

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June 5, 2000
5928-00-20149

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
10CFR50.46 ANNUAL REPORT

References:
1. FTI Topical Report BAW-10104PA, Rev. 5 "B&W's ECCS Evaluation Model," November 1988.
2. FTI Topical Report BAW-10154A, "B&W's Small-Break LOCA ECCS Evaluation Model," July 1985.
3. FTI Topical Report BAW-10192PA, Rev. 0, "BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," June 1998.
4. FCF Topical Report "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," BAW-10227P, transmitted by letter, J. H. Taylor (FCF) to U.S. NRC Document Control Desk, "Submittal of Topical Report BAW-10227P, Evaluation of Advance Cladding and Structural Material in PWR Reactor Fuel," dated September 30, 1997, JHT/97-36.
5. Letter, J. J. Kelly to USNRC, "10CFR50.46 Thirty Day Report on Significant PCT Change in ECCS Analyses," February 4, 1999, OG-1740.
6. Letter, J J. Cudlin to USNRC, "10CFR50.46 Thirty Day Report on Significant PCT Change in ECCS Analyses," May 25, 1999, FTI-99-1727.

Dear Sir or Madam:

10CFR50.46 (a)(3)(ii) 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 accepted emergency core cooling system (ECCS) evaluation model (EM) or in the application of such a model that affects the peak cladding temperature (PCT) calculation.

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For the reporting period from January 1999 through December 1999, Framatome Technologies, Inc. (FTI) has confirmed that no significant errors were reported in either the CRAFT2-based B&W ECCS EM (Reference 1 for LBLOCA and Reference 2 for SBLOCA) or the RELAP5/MOD2-B&W-based BWNT LOCA EM (Reference 3). However, changes to the RELAP5/MOD2-B&W code and the BWNT LOCA EM were implemented by FTI for the inclusion of the zirconium-based M5 alloy cladding described in BAW-10227P (Reference 4). Additionally, sensitivity studies performed in response to PSC 1-99 determined that the limiting RCP degradation parameters reported in the EM (based on 205-FA Raised Loop (RL) studies) were inconsistent for the 177-FA plants, as described below.

During 1999, there were two model application input errors reported that resulted in a significant change (>50° F) in the calculated peak clad temperatures. The first application input error is related to inputs for the limiting reactor coolant pump type and associated two-phase degradation for the 177-FA Lowered Loop (LL) plants. The second application input error is related to inputs for the fuel assembly grid blockage droplet breakup factors for the TMI Unit 1 20 percent tube plugging LBLOCA analyses. The NRC has previously been notified of these significant errors within 30 days of discovery (References 5 and 6). Additionally, these errors and the results of the reanalyses for TMI Unit 1 were identified in the previous GPU Nuclear letters to the NRC, dated May 12, 1999 (1920-99-20155 and July 30, 1999 (1920-99-20360). The reevaluated peak clad temperatures for affected analyses considering the cumulative effects of these errors were less than the 10 CFR 50.46(b)(1) limit of 2200°F.

The enclosed Appendix A provides a summary of the EM changes and error notifications applicable to TMI Unit 1 as identified and evaluated by FTI, and reported in accordance with 10CFR50.46(a)(3)(ii).

If any additional information is needed, please contact David J. Distel at (610) 640-6672.

Very truly yours,

John B. Cotton

John B. Cotton
Vice President, TMI Unit 1

JBC/djd

Enclosures: 1) Appendix A: EM Changes and Significant Error Notification

cc: Administrator, USNRC Region I
Senior Project Manager, TMI Unit 1
Senior Resident Inspector, TMI Unit 1
File No. 00068

Appendix A: EM Changes and Significant Error Notification

A.1 CRAFT2 Evaluation Models

No significant errors were reported in the CRAFT2-based B&W ECCS EM, BAW-10104PA, Rev. 5 for LBLOCA and BAW-10154A Rev. 0 for SBLOCA, during 1999.

A.2 RELAP5/MOD2-B&W Evaluation Model

No significant errors were reported in the RELAP5/MOD2-B&W-based BWNT LOCA EM (BAW-10192PA Rev. 0) during 1999. However, two changes related to the BWNT LOCA EM have been evaluated during the reporting period. The changes pertain to ECCS analyses based on the M5 alloy cladding type and the limiting reactor coolant pump two-phase degradation.

A.2.1 RELAP5/MOD2-B&W Evaluation Model Changes

ECCS Analysis of M5 Alloy Cladding

The topical report outlining the zirconium-based M5 alloy (BAW-10227P) was submitted to the USNRC for review and approval in the fall of 1997 (Reference A-5). The question and answer period has continued through 1999 (References A-6 through A-10), culminating in the issuance of a Safety Evaluation Report (SER) by the USNRC dated December 14, 1999 (Reference A-11). A revised Safety Evaluation Report was issued by the USNRC on February 4, 2000 (Reference A-12) to clarify the acceptability of the maximum pre-rupture strain for M5 and approve the code changes made to model the M5 alloy. ECCS analyses performed for the M5 alloy fuel pin cladding required changes to the material properties and mechanical characteristics within the RELAP5/MOD2-B&W computer code. Changes to the code were submitted as an attachment to responses to NRC requests for additional information on the M5 topical report in September 1999 (Reference A-9). These changes were described as Revision 04 of the RELAP5/MOD2-BAW topical report (BAW-10164P). The changes to the code include the capability to evaluate M5 alloy pins and the Zircaloy pins in a single calculation, automated supplemental pin capabilities and a void-dependent form loss model to simplify the SBLOCA assembly cross-flow. Subsequent changes to the BWNT LOCA EM (BAW-10192PA Rev. 0) will include reference to the M5 alloy topical report (BAW-10227P) methodology SER and Revision 04 to the RELAP5/MOD2-B&W topical report (BAW-10164P). The EM and code changes do not affect any ECCS analyses performed based on Zircaloy cladding types, however a change in cladding type to the M5 alloy would result in differences in the calculated PCT resulting from the material property and mechanical characteristic differences.

Reactor Coolant Pump Two-Phase Degradation

Sensitivity studies performed on the 177-FA plant revealed that the limiting RCP degradation parameters reported in the EM based on 205-FA RL studies would not produce limiting PCT results. This issue was reported as Preliminary Safety Concern 1-99 (Reference A-3). Table 9-2 of Volume I of the EM 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-4 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-FA 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 does not constitute a significant safety hazard and is not reportable under 10CFR21.

A.2.2 RELAP5/MOD2 - B&W Evaluation Model Application Input Errors

Two input errors have been discovered in ECCS analyses performed using the RELAP5/MOD2-B&W-based EM in 1999.

RC Pump-Type

In compliance with 10CFR50.46 (a)(3), the NRC has previously been notified (Reference A-1) of the following "significant" error in an input to the RELAP5/MOD2-B&W code, which is part of the BWNT LOCA EM (BAW-10192PA, Rev. 0).

Notification:

Some ECCS analyses performed with the BWNT LOCA EM underpredicted the PCT because the most limiting reactor coolant pump (RCP) 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.

Disposition:

FTI has identified all of the LBLOCA cases affected by the erroneous pump type and non-conservative two-phase pump degradation model, and the cases that presented the most serious challenges to the 10 CFR 50.46 limits have been reanalyzed. The affected cases were applicable to TMI Unit 1. Reanalyzed cases were used to develop PCT deltas that were applied to the non-limiting core elevations or times in life. Reference A-4 summarizes the cumulative results of the reactor coolant pump reanalyses and associated PCT deltas or LHR adjustments. When reanalyzed or reevaluated, the PCTs for all affected analyses were less than the 10CFR50.46 (b)(i) limit of 2200°F.

Fuel Assembly Grid Parameters

In compliance with 10CFR50.46 (a)(3), the NRC has previously been notified (Reference A-2) of the following "significant" application input error to the BEACH code, which is part of the BWNT LOCA EM (BAW-10192PA, Rev. 0).

Notification:
The TMI Unit 1 15 percent steam generator tube plugging (SGTP) LBLOCA analyses and the B&W Owners Group 20 percent SGTP LBLOCA analyses for TMI Unit 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.

Disposition:
The NRC was notified of the fuel assembly grid input errors and the associated PCT increases in May of 1999 (Reference A-2). Reanalyses of the bounding 20 percent SGTP 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 all affected analyses were less than the 10CFR50.46 (b)(i) limit of 2200°F.

References:

A-1. Letter, J. J. Kelly to USNRC, "10CFR50.46 Thirty Day Report on Significant PCT Change in ECCS Analyses", February 4, 1999, OG-1740.

A-2. Letter, J. J. Cudlin to USNRC, "10CFR50.46 Thirty Day Report on Significant PCT Change in ECCS Analyses", May 25, 1999, FTI-99-1727.

A-3. 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-4. FTI Document 51-5006132-00, "PSC 1-99 Resolution", January 1999.

A-5. FCF Topical Report "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", BAW-10227P, transmitted by letter, J. H. Taylor (FCF) to U.S. NRC Document Control Desk, "Submittal of Topical Report BAW-10227P, Evaluation of Advance Cladding and Structural Material in PWR Reactor Fuel," dated September 30, 1997, JHT/97-36.

A-6. Letter, T. A. Coleman, Framatome Cogema Fuels, to U.S. NRC Document Control Desk, GR99-031.doc, February 5, 1999.

A-7. Letter, T. A. Coleman, Framatome Cogema Fuels, to U.S. NRC Document Control Desk, GR99-089.doc, April 23, 1999.

A-8. Letter, T. A. Coleman, Framatome Cogema Fuels, to U.S. NRC Document Control Desk, GR99-156.doc, July 29, 1999.

A-9. Letter, T. A. Coleman, Framatome Cogema Fuels, to U.S. NRC Document Control Desk, GR99-194.doc, September 24, 1999.

A-10. Letter, T. A. Coleman, Framatome Cogema Fuels, to U.S. NRC Document Control Desk, GR99-212.doc, October 20, 1999.

A-11. Letter, Stuart A. Richards (NRC) to T. A. Coleman, (FCF), "Acceptance for Referencing of Topical Report BAW-10227P: Evaluation of Advance Cladding and Structural Material (M5) in PWR Reactor Fuel," TAC No. M99903, December 14, 1999.

A-12. Letter, Stuart A. Richards (NRC) to T. A. Coleman, (FCF), "Revised Safety Evaluation (SE) for Topical Report BAW-10227P: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," TAC No. M99903, February 4, 2000.