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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
10 CFR 50.46 ANNUAL REPORT

References:

1. FTI Topical Report BAW-10104P-A, Rev. 5, "B&W's ECCS Evaluation Model," November 1988.
2. FTI Topical Report BAW-10154-A, Rev. 0, "B&W's Small-Break LOCA ECCS Evaluation Model," July 1985.
3. 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.
4. FTI Topical Report BAW-10192P, Rev. 1, "BWNT LOCA- BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," July 2000.
5. FTI Topical Report BAW-10227P-A, Rev. 0, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000.
6. FTI Topical Report Clarification BAW-10192C-00,"Letter to NRC Documenting the Statistical Determination of the Initial Fuel Temperature Uncertainty, Applied to 43-10192 Rev.0", 12/21/00.
7. FTI Topical Report BAW-10166PA-04,"BEACH-Best Estimate Analysis Core Heat Transfer; A Computer Program for Reflood Heat Transfer During LOCA", February 1996.
8. FTI Topical Report BAW-2374, "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once-Through Steam Generators," July 2000.
9. FTI Document 51-5001731-01, "BWNT LOCA EM Limitations and Restrictions," April 2000.

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10. Letter, J. J. Kelly to USNRC Document Control Desk, FTI-00-2433, September 26, 2000.
11. FTI Document 51-5010065-00, "LOCA Evaluation of Stainless Steel and Nat. $UO_2$ Replacement Rods," October 2000.
12. Letter, D. J. Firth to USNRC Document Control Desk, FTI-00-3085, December 20, 2000.
13. FCF Topical Report BAW-2149-A, "Stainless Steel Replacement Rod Methodology," September 1993.

Dear Sir or Madam:

10 CFR 50.46 (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.

For the reporting period from January 1, 2000 through December 31, 2000, Framatome ANP 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 as described Enclosure A. Enclosure A provides a summary of the EM changes applicable to TMI Unit 1 as identified and evaluated by Framatome ANP, and reported in accordance with 10 CFR 50.46(a)(3)(ii).

Enclosure B provides a description of generic items involving the ECCS EM as identified by Framatome ANP.

No new regulatory commitments are established in this submittal.

If any additional information is needed, please contact David J. Distel at (610) 765-5517.

Very truly yours,

James A. Hutton
Director - Licensing
Mid Atlantic Regional Operating Group

Enclosures: A) EM Changes and Significant Error Notification
B) B&W 177-FA Plant Generic Items

JAH/djd

cc: H. J. Miller, USNRC Administrator, Region I
T. G. Colburn, USNRC Senior Project Manager, TMI Unit 1
J.D. Orr, USNRC Senior Resident Inspector, TMI Unit 1
File No. 00068

## ENCLOSURE A

## EM CHANGES AND SIGNIFICANT ERROR NOTIFICATION

### A.1 CRAFT 2 Evaluation Model

No errors were reported in the CRAFT2-based B&W ECCS EM, BAW-10104P-A, Rev. 5 for LBLOCA (Reference 1) and BAW-10154-A, Rev. 0 for SBLOCA (Reference 2), during 2000.

### A.2 RELAPS/MOD2-B&W Evaluation Model

No errors were reported in the RELAP5/MOD2-B&W-based BWNT LOCA EM, BAW-10192P-A Rev. 0, (Reference 3), during 2000.

Changes to the BWNT LOCA EM or associated computer codes have been proposed and have either been approved by the NRC or are currently under review. These changes are related to LOCA modeling of the $M5_{TM}$ advanced cladding, RELAP5/MOD2-B&W code changes, a clarification to the BWNT LOCA EM, and a revision to the BWNT LOCA EM that updates the LBLOCA methodology. Additionally, two-phase reactor coolant pump degradation SBLOCA sensitivity studies with offsite power available are described.

ECCS Analysis of $M5_{TM}$ Alloy Cladding

Topical report BAW-10227P-A describes modifications to the BWNT LOCA EM and the associated computer codes for application to the $M5_{TM}$ cladding and guide tube material. The topical report was submitted to the NRC for review and approval in the fall of 1997, and was approved by the NRC in February 2000 (Reference 5). The approval, however, only covered the M5 cladding method changes.

RELAP5/MOD2-B&W Topical Report Changes

Revision 4 of BAW-10164P included several other changes to the RELAP5/MOD2-B&W computer code besides the $M5_{TM}$ cladding models (Reference 5), for which approval has not yet been received. The changes included EM pin model improvements necessary to model multiple cladding material types, an option for multiple pin channels in a single core fluid channel, a void-dependent core cross-flow option for SBLOCA applications, and an automated limit of the rupture flow blockage for droplet breakup in BEACH applications.

The option for user input cladding material properties was added to allow modeling of the approved $M5_{TM}$ properties (Reference 5). In addition, supplemental pin capability was added to facilitate the modeling of multiple EM pin channels within a single hydrodynamic fluid channel (i.e., use of a hot pin or burnable poison rod in one assembly). The relationship between the supplemental pin and the remainder of the pins in a common fluid channel is one in which the supplemental pin swell and rupture will not define the rupture flow blockage for the entire channel. These parameters are controlled by the larger group, or primary pin channel. The same analysis may model fuel rods with one of two cladding material types, the default $ZR_4$ properties or a user-input set. The supplemental rod modeling is particularly useful for gadolinia or lead test pin (e.g. $M5_{TM}$) analyses. It may also be used in future EM revisions for hot pin applications, in which the hot pin has a different radial peak or perhaps a different initial fuel temperature.

A special void-dependent form-loss option was developed to automate user input of the BWNT EM cross-flow model for certain multi-core channel SBLOCA applications. This option allows the code to alter the user input constant form-loss coefficient based on the void fraction in the upstream volume. The specific EM applications that will use this model are BWNT LOCA SBLOCA analyses. This model allows the regions of the core covered by a two-phase mixture or pool to have a resistance that is different from that in the uncovered or steam region, as described in the approved EM.

The final change limits the code-calculated pin rupture droplet breakup to 60% blockage for primary pin channels as required by the Safety Evaluation Report on Revision 2 of Reference 7. This SER limit was automatically included in the code to assure that the limit on droplet breakup blockage could not be violated.

<u>BAW 10192P Revision 0 Clarification BWNT LOCA EM</u>

Clarifications to Revision 0 of BAW-10192P-A (Reference 6) were submitted to the NRC in February 2000. The clarification addresses a refinement of the modeling and use of the hot rod within the hot assembly in the LBLOCA EM applications. The hot rod and hot assembly power peaking remains identical, but the initial fuel temperatures are different based on 95/95 statistical uncertainties. Notification of NRC approval of the submitted clarification was requested, however this notification has not yet been received.

<u>BAW 10192P Revision 1 BWNT LOCA EM</u>

Revision 1 of BAW-10192P was submitted to the NRC in July 2000 (Reference 4), however, notification of approval has not yet been received. The revision pertains to the removal of the REFLOD3B code from the LBLOCA package. Refill/reflood calculations will be performed entirely by RELAP5/MOD2 in a single system analysis that also calculates the hot rod thermal response. BEACH routines within RELAP5/MOD2 will continue to be used for the fuel rod thermal response, but they will now be dynamically coupled to the entire RCS. This EM revision streamlines and closely couples current LBLOCA calculation methods and does not affect the SBLOCA calculation. Sensitivity studies and a representative plant application of the new EM model is presented in Appendix B of Volume 1 of the submittal.

RCP Two-Phase Degradation Model for SBLOCA

The NRC-approved SBLOCA EM (Reference 3) calculates two-phase reactor coolant pump (RCP) performance curves using the RELAP5 head difference and degradation multipliers that were derived from the Semiscale pump tests. Examination of the Semiscale pump degradation curves, which are based upon tests run at relatively low pressures, indicates that the RELAP5 model can overpredict 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 2-00). Comparison of the EM curves to representative data, specifically the CE 1/5-scale steam-water tests (which were run at higher pressures), confirms that the EM pump model overpredicts pump head degradation during two-phase flow early in the event. Since less pump degradation results in additional core uncovering and higher PCTs, the approved EM model cannot be judged to be conservative for this application. When a bounding pump performance curve (the lower bound "M3-modified" curve used in the approved large break LOCA model) is modeled, the predicted consequences are much more severe. Therefore, the selection of a RCP two-phase degradation model in future SBLOCA analyses will be justified by sensitivity studies similar to those used for LBLOCA applications, or reference to applicable studies that determine the conservative model for application to specific analyses.

# ENCLOSURE B

# B&W 177-FA PLANT GENERIC ITEMS

## B.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 (Reference 10) 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 HPI and 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 (EOPs) 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 evaluation model (EM) reported in BAW-10192P-A (Reference 3), 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 (Reference 12) 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 to NRC in Framatome's final summary report for PSC 2-00 (Letter to NRC, FANP-01-988, April 2, 2001) and AmerGen letter to the NRC 5928-01-20103, dated April 11, 2001; and will be reported in the 2001 10 CFR 50.46 Annual Report.

## B.2 **PSC 2-98/LBLOCA Tube Load**

Preliminary Safety Concern 2-98 (PSC 2-98) is related to a concern that the Once Through Steam Generator (OTSG) tube tensile loads resulting from a postulated Small Break Loss of Coolant Accident (SBLOCA) may be larger than the currently recognized limiting load. The limiting load was originally established as resulting from a Main Steam Line Break (MSLB). The results of the evaluation of PSC 2-98 include the determination of limiting loads that may result from SBLOCA, MSLB, as well as an examination of other events. The operability of the steam generator and impact of loads on steam generator repair products was also assessed in the evaluation of the safety concern. Topical Report BAW-2374 (Reference 8) was submitted to the NRC on July 7, 2000. The Topical report has not received approval to date and Revision 1 to BAW-2374 was released in March 2001 for NRC approval.

## B.3 **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 (Reference 13). 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 11 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 both on a best-estimate and evaluation model basis. The results of this evaluation are generically applicable to all B&W plants.

In order to apply the $UO_2$ LOCA evaluation LHR and/or Fq limits to a core containing replacement rods (stainless steel solid filler rods or natural uranium fuel rods), three criteria must be met as listed below.

1) The total number of replacement (stainless steel or natural uranium) rods within the core must not exceed 200, with a maximum of 10 per assembly.

2) The LOCA LHR and/or $F_q$ limits are based on the peak pin in the core. The core maneuvering analyses must verify that the core peaking based on the exact

configuration of the natural $UO_2$ or stainless steel rods is within the constraints set by the $UO_2$ LOCA limits. LOCA LHR and/or $F_q$ limits for the original assembly with Zircaloy or $M5_{TM}$ cladding are applicable to the reconstituted assembly with natural $UO_2$ rods having either Zircaloy cladding, $M5_{TM}$ cladding, or stainless steel pins.

3) The rod average power history must be bounded by the conservative envelope modeled in the fuel pin initialization (currently performed by TACO3) for time-in-life LOCA evaluations used in determining the applied LOCA limits.

## B.4 <u>Revision 1 to EM Limitations and Restrictions Document</u>

The EM Limitations and Restrictions document (Reference 9) 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 can be 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.