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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)(ii)

Ladies and Gentlemen:

In accordance with 10 CFR 50.46(a)(3)(ii), the FirstEnergy Nuclear Operating Company (FENOC) herewith submits the attached annual report of changes to the Emergency Core Cooling System (ECCS) Evaluation Model (EM) used at the Davis-Besse Nuclear Power Station (DBNPS). None of these changes result in a change to the peak fuel cladding temperature exceeding 50 degrees Fahrenheit; therefore, these changes are classified as not significant in accordance with 10 CFR 50.46(a)(3)(i).

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

Very truly yours,

RMC/s

Attachments

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, DB-1 NRC/NRR Project Manager
K. Zellers, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

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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. Prior to the 12th refueling outage in April and May of 2000, the Davis-Besse Nuclear Power Station (DBNPS) continued to use the original CRAFT2-based ECCS evaluation model. The FirstEnergy Nuclear Operating Company (FENOC) informed the NRC via letter dated April 19, 2000 (FENOC Letter Serial Number 2655, Reference 4) that it had replaced the original CRAFT2-based EM with the newer RELAP5-based EM. This change was implemented with the Cycle 13 startup in May 2000 and was a first time application of the new methodologies of the revised EM at the DBNPS. In addition, the analysis was performed to accommodate a future 7 percent power uprate from the current licensed power level of 2772 MWt, and a unit-average 20 percent steam generator tube plugging. As reported in the Reference 4, the results of that application of the RELAP5-based EM satisfied the five acceptance criteria of 10 CFR 50.46. The PCT for the Large Break (LB) Loss of Coolant Accident (LOCA) decreased from 2176 to 2102 degrees Fahrenheit (F), while the Small Break (SB) LOCA PCT decreased from 1707 to 1408 degrees F, compared to the 2200 degree F acceptance criteria. The limiting SBLOCA PCT has since been increased by 20 degrees F to 1428 degrees. This occurred due to a small reduction in the required High Pressure Injection (HPI) pump capacity as described in the following.

For the period January 1, 2000 through December 31, 2000, no significant errors were identified in either the CRAFT2 or RELAP5-based EMs that were used for licensing analyses for the DBNPS. Specifically, no significant errors were discovered in the CRAFT2-based Babcock & Wilcox (B&W) ECCS EM, BAW-10104P-A, Revision 5 for LBLOCA (Reference 1), or BAW-10154-A, Revision 0 for SBLOCA (Reference 2). Additionally, no errors were discovered in the RELAP5/MOD2-B&W-based Babcock & Wilcox Nuclear Technologies (BWNT) LOCA EM, BAW-10192P-A, Revision 0 (Reference 3) that covers both LBLOCA and SBLOCA including applications with $M5_{TM}$ fuel cladding as described in BAW-10227P-A (Reference 5). Also, no input errors were detected that changed the results of the most limiting SBLOCA or LBLOCA analyses.

During this period, an issue pertaining to use of the RELAP5 EM was noted and reported to the NRC by the B&W Owners Group. The issue that pertained to the DBNPS is Preliminary Safety Concern (PSC) 2-00 (Reference 8, 10), which found that assumption of Loss-of-Offsite Power (LOOP) is not conservative for certain LOCAs. However, recalculation determined that the existing worst case LBLOCA and SBLOCA PCT results continued to bound the cases that were affected by this PSC.

A summary of the analyses, EM code changes, and evaluations for the period January 1, 2000 to December 31, 2000 are identified in the following sections:

I. Evaluation Model Input Changes (Reduced HPI Requirement)

The DBNPS SBLOCA analyses with reduced HPI flow was reevaluated to provide a basis for a relaxation in the HPI pump acceptance criteria in excess of what is currently allowable. The relaxation was evaluated as a contingency for HPI pump testing.

An HPI head flow reduction of 1.5 percent decreases the ECCS flow rate that is used to mitigate the consequences of a SBLOCA. The head flow reduction will have a slightly different consequence on every break size that has core uncovering, but it will have the largest impact on the smaller break sizes that recover prior to reaching the Core Flood Tank (CFT) injection pressure. Previous analyses showed that the full-area HPI line break was the limiting transient and this break would be affected most by a HPI flow reduction. Therefore, the HPI line break was reanalyzed with the 1.5-percent HPI head decrease to demonstrate that the DBNPS would remain in compliance with the 10 CFR 50.46 acceptance criteria. The most limiting case results were acceptable. The maximum SBLOCA spectrum PCT increased by 19.4 degrees F (~20 degrees F) for the double-ended HPI line break. It is expected that the other SBLOCAs would be bounded by this overall PCT, with PCT increases similar to or less than that calculated for this case. Therefore, a PCT increase of 20 degrees F should be applied to other SBLOCA break sizes that have core uncovering and PCT increases above the initial steady-state cladding temperature. The limiting SBLOCA PCT for the DBNPS is 1428 degrees F for the 0.02463 ft^2 HPI line break considering a 1.5 percent reduction in pressure for the HPI head-flow curve.

II. RELAP5/MOD2-B&W Topical Report Application

Topical Report BAW-10227P-A (Reference 5) 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. The approval, however, only covered the $M5_{TM}$ cladding method changes. This change to the RELAP5-based LOCA EM was included in the initial application of the EM to the DBNPS, as described in the Reference 4. Revision 4 of BAW-10164P (Reference 13) included several other changes to the RELAP5/MOD2-B&W computer code besides the $M5_{TM}$ cladding models, which are currently under review by NRC but for which approval has not yet been received. With the exception of method addition item 6 below, the current applications of the EM are considered to be consistent with the limitations and restrictions imposed on the EM (Reference 3) in the NRC Safety Evaluation Report (SER). However, the additions are required to adequately account certain analytical aspects of the current fuel design and system behavior. These method additions included 1) EM pin model improvements necessary to model multiple cladding material types and an option for multiple pin channels in a single core fluid channel, 2) a void-dependent core cross-flow option for SBLOCA applications, 3) an automated limit of the rupture flow blockage for droplet breakup in Best Estimate Analysis Core Heat Transfer (BEACH) applications, 4) RCP two-phase degradation model for SBLOCA, 5) End of Cycle (EOC) average reactor coolant system temperature (Tave) reduction maneuver analysis, and 6) justification for increased hot spot cladding temperatures at the time of bottom of core recovery.

The method additions are summarized as follows:

1) 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 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. The multiple pin option was used in the DBNPS analyses for the current operating cycle (cycle 13) to model the gadolinia fuel rods that are in the UO_2 fuel assembly. (It should be noted that this addition does not alter the results of the approved EM or change the fuel pellet material properties or

LOCA input parameters. The fuel parameters are explicitly addressed by approved Topical Report BAW 10184P-A, GDTACO – Urania Gadolinia Fuel Pin Thermal Performance.)

2) A special void-dependent cross-flow or form-loss option was developed to automate user input of the BWNT EM cross-flow model for 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 applications that will use this model are 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 void-dependent cross-flow model was used in the DBNPS application of the LOCA EM in anticipation of receiving NRC approval for use of this model. However, given that the void-dependent cross flow model has not yet been approved by the NRC, the limiting case (HPI line break) was performed with the fixed cross-flow resistance model in the currently approved EM to demonstrate that comparable and acceptable results are obtained in either case.

3) A change was also made to RELAP5 to automatically limit the code-calculated pin rupture droplet breakup to 60% blockage for primary pin channels as required by the NRC's SER on Revision 2 of Reference 6. This SER limit was automated in the code to assure that the limit on droplet breakup blockage could not be violated and thus remove a check that the analyst would otherwise have to perform.

4) 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 PSC 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 LBLOCA 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.

5) Analyses for the EOC 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. The LOCA analyses 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 Linear Heat Rate (LHR) limit penalties are necessary to accommodate the EOC T_{AVE} maneuver for the Mark-B fuel types.

6) The NRC has approved the BEACH code (Reference 6) for general use in EM analyses during the refill and reflood phases of a LBLOCA. It was found acceptable for predicting core heat transfer calculations (including the hot rod PCT), cladding oxidation, and swelling and rupture effects. The code is general in nature and can be used for analyses under a wide range of conditions; however, the NRC has

limited its use to ranges over which code performance has been assessed via existing benchmark cases. Those benchmark ranges, as given in the NRC's SER for BAW-10166P-A, Rev. 2, cover initial hot spot cladding temperatures up to 1640 degrees F. This temperature does not cover the initial cladding temperatures for the B&W-designed plant analyses at the time of bottom of core recovery (BOCREC). The NRC limitation states that additional justification is necessary for use outside of the assessed ranges. The BWNT LOCA EM and responses to requests for additional information provided the NRC with numerous plant demonstration cases that had BOCREC cladding temperatures in the range of 1800 degrees F. Although the NRC approved this EM for LOCA application on the B&W-designed plants, FTI has performed an additional benchmark to further validate the BEACH code for initial BOCREC cladding temperatures above 1640 degrees F. Because the additional validation was not provided to the NRC as a revision to the Topical Report, this method addition has been entered into the DBNPS Corrective Action Program Condition Report process to determine its safety significance and necessary corrective actions. Since there is no error in the use of the EM associated with this item, and no change in the predicted phenomenon, there is no change in the predicted PCT. However, there is a nonconformance associated with the failure to revise the Topical Report and receive approval for use in the EM.

The case selected for extending the initial cladding temperature range was taken from FLECHT-SEASET tests (Full Length Emergency Core Cooling Heat Transfer - Separate Effects Tests And System Effects Tests). The selected test has an initial cladding temperature of 2045 degrees F and is one test in a series of experiments that has been used extensively to validate the existing BEACH analysis ranges.

The key test parameters for this test (except for the initial clad temperatures) were similar to a previously reported benchmark. The results show a conservative comparison of measured-to-predicted PCT versus core elevation and quench front advancement. (BEACH matched the quench front data well up to 150 seconds, and conservatively underpredicted the advancement thereafter.)

This benchmark of a FLECHT-SEASET test case provides additional confirmation that the general formulation of BEACH is adequate and acceptable for reflood heat transfer prediction in LOCA applications. The adequacy of the code predictions is demonstrated in this benchmark for increased initial cladding temperatures of up to 2045 degrees F, which encompasses the expected range of B&W plant initial clad temperatures at BOCREC. This benchmark shows there is no safety significance in the use of BEACH with initial BOCREC temperatures above 1640 degrees F for B&W plant LOCA analyses performed in accordance with the NRC-approved methods in the BWNT LOCA EM (Reference 3).

III. Reactor Core Design Change Evaluations

These evaluations were performed to ensure the assumptions of the EM are satisfied and to confirm continuity of the analysis between the previous and current EMs.

1) Neither the DBNPS cycle 12 nor the current cycle 13 utilized reconstituted fuel assemblies that incorporate stainless steel rods. However, in the future, 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 9). 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. An evaluation has been performed to determine the effect on the results of a LOCA analysis using 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 effect of the replacement rods on the initial stored energy, heat transfer and swell/rupture flow blockage was considered and the effect

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 Nuclear Heat Flux Hot Channel Factor (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.

a) 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.

b) The LOCA LHR and/or Fq 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 Fq 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.

c) 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.

2) Present and future core designs may utilize a small number of fuel assemblies from previous batches, i.e. Mark-B8A fuel. These older fuel assemblies would typically (historically) occupy the core center location or low power peripheral locations. The Mark-B8A LOCA LHR limit evaluations were originally performed with the CRAFT2-based EM (Reference 1). The Mark-B8A LOCA LHR limits were adjusted to a basis consistent with the RELAP5-based EM (Reference 3) for application to Cycle 13. The existing Mark-B8A LHR limits, based on the CRAFT2 code, were adjusted to bound the effects of different plant parameters and the RELAP5/MOD2-B&W evaluation model. The LHR adjustments were performed in order to preserve the original CRAFT2 PCT predictions. For burnups of less than 24.5 GWd/mtU, this required a range of downward adjustments from a maximum change of 2kW/ft (for the 10-ft. core elevation at a CRAFT2 based 17kW/ft) to a zero adjustment at the 2-ft. core elevation. The LHR beyond a burnup of 24.5 GWd/mtU is reduced to a lesser extent since at higher burnups, lower LHRs are allowed, and there is significant PCT and pin pressure margin for both the CRAFT2 and RELAP5-based evaluation models.

IV. <u>Scenario or Configuration Changes Performed that Affect Core Cooling Analysis</u>

1. PSC 2-00/Core Flood Tank Line Break with 2 Minute Operator Action Time

PSC 2-00 was initiated by Framatome Technologies, Inc. (FTI) 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 RCPs was performed at two minutes after Loss of Subcooling Margin (LSCM). The NRC was informed of PSC 2-00 via FTI letter (Reference 8, 10) on September 26, 2000.

The CFT line break has historically been analyzed for the B&W-designed plants with a 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 Low Pressure Injection (LPI) pump are initially unpowered. A single operating LPI pump and valve arrangement that results in 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 B&W lowered loop plant PCTs less than 800 degrees F for this break with an immediate LOOP.

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 Reactor Coolant System (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 (Reference 3), predicted significant PCT increases for several of the 177-FA lowered-loop plants when the RCPs are powered for the first two minutes following LSCM. 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.

For the DBNPS analyses, the detrimental effect of the delayed RCP trip is mitigated by relatively low head (but high flow) HPI pumps, as compared to the lowered loop B&W designed plants. The high flow pumps are beneficial because the break size of concern is sufficiently large to produce rapid RCS depressurization. Analysis supporting this conclusion used the conservative "M-3" Reactor Coolant Pump degradation curves. The calculated change in the 0.44 sq. ft. DBNPS Core Flood Line Break PCT for a delayed RCP trip at two minutes increased to 962 degrees F from 715 degrees F, with the largest PCT impact observed on the 0.5 sq. ft. "Cold Leg Pump Discharge" break. This PCT increased from 715 degrees F to 1039 degrees F. However, these PCTs remain bounded by the 1428 degrees F PCT calculated for the DBNPS HPI line break. These results were reported to the NRC by FTI in April 2001 (Reference 10).

2. PSC 2-98/LBLOCA Tube Load

PSC 2-98 is related to a concern that the Once Through Steam Generator tube tensile loads resulting from a postulated 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 7) 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 the March 2001 for NRC approval. This item is included in this report since it is related to long term cooling capability, which is required by 10 CFR 50.46.

3. DBNPS Boron Precipitation Control Modification

During the 12^{th} refueling outage in April-May 2000, the DBNPS substantially modified its means of achieving post LOCA boric acid precipitation control. This modification involved increasing the capacity of the Auxiliary Pressurizer Spray (APS) line and adding the provision of a revised backup control methodology. This change was reviewed and approved by the NRC (Reference 11).

Evaluations were performed to identify boron concentration control methods and calculations used to show that the DBNPS operators can control the core boron concentration following a relatively large cold leg pump discharge LOCA. Compliance with 10 CFR 50.46 requirements was demonstrated by calculations that established initiation times for active boron dilution methods for a core power of 2772 MW_t with a decay heat of 1.2 times the ANS 1971 fission product standard plus the B&W heavy isotope actinide contribution. These calculations used power-level specific core mixing volumes to define limiting times for operator action. These operator action times are very conservative because of the decay heat conservatism plus the lack of credit for passive dilution contributions. Credit for passive boron dilution mechanisms such as Reactor Vessel (RV) vent valve liquid entrainment, hot leg nozzle gap flow, or the thermal shield annulus flows was not included because each is dependent upon the transient evolution. They each diminish in effectiveness as the time after post-trip increases either because of lower decay heat levels, long-term RV versus RV internal temperature changes, or potential small debris injected during the sump recirculation phase. Explicit credit for these means is difficult to quantify at the DBNPS, because there is no way for the operators to know the magnitude of the flows or their effectiveness without accurately knowing the sump boron concentration. Without credit for these passive mechanisms, calculations were performed to define the minimum times for operator actions to reconfigure ECCS flow paths to provide active methods for boron concentration control whenever the core is not adequately subcooled.

The two active methods, namely APS flow or Decay Heat Drop Line recirculation, have positive flow indication such that the operators can verify that the methods have been actuated. Either dilution method is capable of providing ample flows to successfully control the post-LOCA core boron concentrations when actuated based on the guidance provided in the evaluations.

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. Letter, Guy G. Campbell to USNRC Document Control Desk, Docket Number 50-346, Serial Number 2655, April 19, 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 BAW-10166P-A, Rev. 4, "BEACH – Best Estimate Analysis Core Heat Transfer; A Computer Program for Reflood Heat Transfer During LOCA", February 1996.
7. 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.
8. Letter, J.J. Kelly to USNRC Document Control Desk, FTI-00-2433, September 26, 2000.
9. FCF Topical Report BAW-2149-A, "Stainless Steel Replacement Rod Methodology", September 1993.
10. Letter, D.J. Firth (FTI) to USNRC Document Control Desk, FANP-01-988, April 2, 2001
11. Letter, USNRC to Guy G. Campbell, "Issuance of Exemption from the Requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K (TAC No. MA7831)," May 5, 2000.
12. FRA-ANP Topical Report BAW-2374, Rev. 1, "Risk-Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping," March 2001.
13. FTI Topical Report BAW-10164P, Rev. 4, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," September 1999 (M5 portions approved).