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Subject: Notification of Change in the Analysis of Record for both Large Break and Small Break Loss of Coolant Accident and of the Resulting Change in Peak Clad Temperature in Accordance with 10 CFR 50.46(a)(3)

Ladies and Gentlemen:

The FirstEnergy Nuclear Operating Company (FENOC) is replacing the CRAFT2 evaluation model (EM) with the RELAP5/MOD2-B&W (RELAP5) EM as the evaluation model of record for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS) loss of coolant accident (LOCA) analyses. The new analysis of record involves a greater than 50 °F change in peak clad temperature for both large break and small break loss of coolant accident (LBLOCA and SBLOCA) analyses results. Pursuant to 10 CFR 50.46(a)(3), the DBNPS hereby provides notification of these changes in the calculated peak clad temperature for the DBNPS. The calculated peak clad temperature remains below the 10 CFR 50.46(b)(1) limit of 2200 °F.

Enclosure 1 provides a summary of the new analysis, including an evaluation which concludes that the RELAP5 EM is acceptable for use in DBNPS-specific LOCA applications. This enclosure also provides information which demonstrates continued compliance with the emergency core cooling system requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K.

No NRC response is required or requested.

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Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 321-8466.

Very truly yours,

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Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
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CHANGE OF EVALUATION MODEL METHODOLOGY FOR LARGE AND SMALL BREAK LOCA

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NO. 1

I. Introduction

A. Purpose and Overview

The purpose of this enclosure is to provide notification to the NRC in accordance with 10 CFR 50.46(a)(3) of a change in the calculated peak clad temperature (PCT) for both the large and small break loss of coolant accident (LBLOCA and SBLOCA) analyses for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). This change is the result of a transition from a CRAFT2-based evaluation model (EM) to a RELAP5/MOD2-B&W (RELAP5) based EM for the DBNPS LOCA analyses.

The revised LBLOCA and SBLOCA analyses for the DBNPS are described in this document. Section I provides an introduction. Section II summarizes the key results. Section III discusses specific EM issues and conservatisms applicable to the LOCA analyses. Section IV summarizes the compliance with the acceptance criteria of 10 CFR 50.46. Section V summarizes compliance with the EM Safety Evaluation Report (SER) restrictions that have been met or must be monitored for the revised analyses. Section VI concludes that the RELAP5 EM is appropriate for use in the DBNPS LOCA analyses. Finally, Section VII provides a list of references.

B. Background

The FirstEnergy Nuclear Operating Company operates the Babcock and Wilcox (B&W)-designed DBNPS. Previously, the DBNPS had been shown to be in compliance with the five criteria of 10 CFR 50.46 based on the NRC-approved CRAFT2-based Emergency Core Cooling System (ECCS) EM principally described in BAW-10104-PA, "B&W's ECCS Evaluation Model" (Reference 1), and BAW-10105, "ECCS Evaluation of B&W's 177-FA Raised-Loop NSS" (References 2), for LBLOCA analyses, and in BAW-10154-P, "B&W's Small-Break LOCA ECCS Evaluation Model" (Reference 3), and BAW-10075A, "Multinode Analysis of Small Breaks for B&W's 177-Fuel-Assembly Nuclear Plants with Raised Loop Arrangement and Internals Vent Valves" (References 4), for SBLOCA analyses. Framatome Technologies, Incorporated (FTI), the B&W

successor, has since developed and received NRC approval for use of the RELAP5-based EM described in BAW-10192-PA, "BWNT LOCA Evaluation Model for OTSG Plants" (Reference 5), to replace the CRAFT2-based EM. The LOCA analyses of record for the DBNPS are being revised with new analyses using modified plant boundary and modeling techniques described in the RELAP5-based EM. The new RELAP5/MOD2-B&W LOCA analyses model 20 percent steam generator tube plugging levels (15 percent in the intact loop, 25 percent in the broken loop) and a core power level of 102 percent of 2966 MWt (which includes a 7 percent uprate from the current operating power level of 2772 MWt). The uprated power level was used to allow for future flexibility. However, the uprate is not currently being sought.

The new RELAP5/MOD2-B&W LOCA analyses are suitable for replacement of the ECCS analyses of record for the DBNPS. They were performed in compliance with the EM methods and the limitations and restrictions stated in the NRC Safety Evaluation Reports (SERs) for BAW-10192-PA and for BAW-10227-PA, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (Reference 24).

The LOCA generic noding and sensitivity studies documented in BAW-10192-PA have been shown to apply to the 177 fuel assembly (FA), raised-loop (RL) plant design. FTI also performed the necessary plant-specific sensitivity studies to confirm that the worst break size and most limiting set of plant boundary conditions were applied to the licensing analyses for large LOCAs. LBLOCA analyses were performed with five axial power shapes at various times in life. The required spectrum of limiting SBLOCA break sizes were performed using a composite set of core operating parameters that envelope the entire range of core operation.

II. Summary of Results

The function of the ECCS is to protect the core in the event of a LOCA. 10 CFR 50.46 requires that the evaluation of ECCS performance for a commercial nuclear power plant meet the following criteria:

1. The calculated peak cladding temperature (PCT) is less than 2200 °F.

2. The maximum calculated local cladding oxidation is less than 17.0 percent.

3. The maximum amount of core-wide oxidation does not exceed 1.0 percent of the fuel cladding.

4. The core remains amenable to cooling.

5. Long-term cooling is established and maintained after the LOCA.

The large and small break LOCA calculations documented in References 6 through 11, that were performed with the approved RELAP5/MOD2-B&W EM (Reference 5) and with the M5 Topical (Reference 24), demonstrate compliance with these criteria for breaks up to and including the double-ended severance of the largest primary coolant pipe for the DBNPS. The spectrum also included a 0.44-ft^2 core flood line break and three high pressure injection (HPI) line break sizes of 0.02463-, 0.02- and 0.015-ft^2. An initial core power level of 102 % of 2966 MWt was modeled for both spectrums with an axial peaking factor (APF) of 1.7 peaked at the midpoints for each set of fuel spacer grids. This is 109 % of the currently licensed 2772 MWt core power.

The use of the RELAP5 methodology for the LBLOCA and SBLOCA is contained in References 6 and 11, respectively. The analyses documented in these references incorporate the requirements of 10 CFR 50 Appendix K and demonstrate compliance with the acceptance criteria of 10 CFR 50.46, as shown below:

LBLOCA		
Parameter	**Limit**	**RELAP5**
Peak Clad Temp. (F)	< 2200	2102
Local Oxidation (%)	< 17	4.78
Average Oxidation (%)	< 1	0.205

SBLOCA		
Parameter	**Limit**	**RELAP5**
Peak Clad Temp. (F)	< 2200	1408
Local Oxidation (%)	< 17	0.576
Average Oxidation (%)	< 1	0.155

The RELAP5-based peak clad temperature for the LBLOCA, as indicated above, is 2102 °F for the Mark-B10K fuel design. For comparison, the peak clad temperature based on the current CRAFT2-based LBLOCA analyses are 2176 °F (Reference 29) and 2050 °F (Reference 30) for the Mark-B8A and Mark-B10A fuel designs, respectively.

The RELAP5-based peak clad temperature for SBLOCA, as indicated above, is 1408 °F for the Mark-B10K fuel design. The current peak clad temperature for the CRAFT2-based SBLOCA analysis is 1707 °F (Reference 31), which is bounding for all fuel types prior to the Mark-B10K design.

The tables above demonstrate that using the RELAP5 methodology continues to meet the 10 CFR 50.46 acceptance criteria that pertain to the analytical methodology. The core geometry also remains amenable to cooling, since the fuel assemblies retain their pin-coolant-channel arrangement and are capable of passing coolant along the pins to provide cooling for all regions of the assemblies. Finally, long-term core cooling is assured through demonstrating the core is quenched, cladding temperature is returned to near saturation temperature, and pumped injection is available.

III. Evaluation Model Issues and Conservatisms

A. RELAP5/MOD2-B&W Pump Degradation Study

This study was performed as part of the generic Evaluation Model (EM) sensitivity studies contained in BAW-10192-PA (Volume I, Appendix A, Section A.2.6). The results established a limiting, maximum pump degradation multiplier set to be used in all EM analyses. Preliminary Safety Concern (PSC) 1-99 identified that the 177-FA lowered loop (LL) plants could produce significantly higher PCTs when a minimum two-phase pump degradation model is used. These mixed conclusions resulted in subsequent sensitivity studies for the DBNPS to determine the most limiting degradation model (Reference 13).

The results of the DBNPS sensitivity study (Reference 13) clearly demonstrated that the minimum two-phase degradation case produces more severe results than the maximum degradation case. The minimum degradation multiplier reduces the resistance of the pumps in the HVN octant. As a result, the core flow reverses direction later in the transient and produces lower core flow rates. The decrease in removal of fuel stored energy leads to higher fuel temperatures at the end of blowdown than for the maximum degradation case. Furthermore, there is less liquid available for input to REFLOD3B in the lower plenum of the reactor vessel. As a result, the adiabatic heatup time will be longer resulting in a PCT increase. Therefore, based on these results from the DBNPS sensitivity study, it is concluded that for the DBNPS, the minimum pump two-phase degradation will produce more severe results than the maximum pump degradation and will be used in the LOCA analyses.

B. Energy Deposition Factors

The energy deposition factor (EDF) is defined as the energy absorbed (thermal source) in the fuel pellet and clad divided by the energy produced by the pellet (nuclear source).

$$EDF = P_{\text{thermal source}} / P_{\text{nuclear source}}$$

BAW-10192PA reports that an EDF of 0.973 will be used for the steady-state initialization and during the blowdown portion of the transient, and an EDF of 0.96 will be used during reflood for LBLOCA analyses. New methods and predictions for the EDFs appropriate for use in LOCA analyses at various times in life have recently been evaluated by Framatome Cogema Fuels (FCF) (Reference 15). These calculations do not totally support the 0.973 or 0.96 values for high burnup, low power fuel or fuel that may be surrounded by higher power fuel. Therefore, based on Reference 15, the LOCA evaluations use more conservative EDFs, depending on the time in life and fuel pin type (in some cases, the EDF exceeds a value of 1.0).

C. Fuel Burnup and Thermal Conductivity

The NRC-approved TACO3 fuel performance code uses a conductivity model that varies only with temperature and not with burnup. Recently, SIMFUEL data has become available that demonstrates that fuel thermal conductivity decreases with extended burnup, as documented in BAW-10186-PA, "Extended Burnup Evaluation," (Reference 16). Since the TACO3 model is based on a beginning-of-life conductivity curve, LOCA initialization fuel volume-average temperatures calculated at high burnups are nonconservative. Justification for not using a variable thermal conductivity versus burnup model in TACO3 is supported by increasing the fuel volume-average temperature uncertainty factor for pin burnups exceeding 40,000 MWd/mtU. The NRC, as discussed in the technical evaluation report (TER), has approved this method for BAW-10186-PA.

D. SBLOCA Core Crossflow Resistance Study

Core crossflow is modeled in the base model through the use of RELAP5/MOD2-B&W crossflow junctions between the hot and average channels in the core region. The crossflow areas are calculated based upon the actual flow area exposed by the three-by-four matrix of fuel assemblies in the hot channel, and the junction form loss factors are input based on the method discussed in the EM (BAW-10192-PA, Volume II, Appendix A). This scheme was found to increase the flow diversion out of the hot channel while restricting the flow of lower temperature steam from the average to the hot channel during core uncovering, thereby, maximizing the hot channel peak clad temperature prediction. The only variation between the cases used for the EM and the DBNPS analyses is the implementation of void-dependent crossflow logic. FTI recently developed a new RELAP5/MOD2-B&W code option that used the EM crossflow modeling philosophy to standardize the crossflow modeling implementation by allowing the core crossflow to vary depending on the mixture level (Reference 20) as opposed to the fixed crossflow resistances shown in Table A-3 of the SBLOCA EM (Reference 5).

The void-dependent crossflow model improvement removes the likelihood of PCT variation because of the fixed nature of the constant crossflow model specification. This modeling choice, while in compliance with the modeling philosophy described in the EM, uses a new code model described in Reference 20. This model was added based on the EM discussions, but it was added after the EM was approved by the NRC. To demonstrate that this model change did not constitute an EM change, a sensitivity study was performed using the most limiting case where the fixed core crossflow model defined by the "Base Case" crossflow resistances shown in Table A-3 of the SBLOCA EM was used in lieu of the void-dependent crossflow model (Reference 11). The two cases produced very similar results. In particular, the system pressure response and system draining were very similar. The resulting PCT for the full-area HPI line break with the fixed crossflow model produced a PCT that was 28 °F less than the void-dependent crossflow model. This study demonstrates that the void-dependent crossflow model produces appropriate results. Therefore, the studies performed for the EM remain applicable and do not need to be repeated.

IV. Compliance with 10 CFR 50.46 Criteria

The compliance of the LBLOCA and SBLOCA analyses results with the five 10 CFR 50.46 ECCS criteria is discussed in this section.

1. Peak Cladding Temperature

The first criterion of 10 CFR 50.46 requires that the calculated peak cladding temperature remain below 2200 °F. For all LOCA cases, the PCT was calculated to be less than 2200 °F. The limiting LBLOCA PCT was calculated to be 2102 °F at the 9.536-ft elevation at beginning of life (BOL). The limiting SBLOCA PCT was calculated to be 1408 °F for a double-ended HPI line break.

2. Local Cladding Oxidation

The second criterion of 10 CFR 50.46 requires that the maximum local degree of cladding oxidation not exceed 17 percent. Compliance with this criterion is obtained by evaluating the results of the calculation of peak cladding temperature. In the calculation, local cladding oxidation is computed as long as the cladding temperature remains above 1000 °F.

The hot channel local cladding oxidation values for the Mark-B10K and Mark-B10A LBLOCA analyses are summarized in References 6 and 10, respectively. In all cases, the hot channel local cladding oxidation was significantly less than 17 percent. For SBLOCAs (Reference 11), the results confirmed that the amount of local cladding oxidation for small break LOCAs is also significantly less than 17 percent.

The oxidation values were calculated using a conservative initial oxide thickness to maximize the cladding temperature response due to metal-water reaction. Since the hot channel local cladding oxidation values for the DBNPS analyses did not approach 17 percent for any case, it is concluded that the 17 percent oxidation limit would not be reached even if a maximum initial oxide thickness were used in the large and small break analyses. These results and conclusions confirm that the amount of local cladding oxidation for the LOCAs analyzed meet the 10 CFR 50.46 local cladding oxidation requirement.

3. Whole-Core Oxidation and Hydrogen Generation

The third criterion of 10 CFR 50.46 states that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel reacted, excluding the cladding surrounding the plenum volume.

The whole-core hydrogen generation was estimated using a simple approximation of the detailed method outlined in Section 6.0 of the EM. The maximum whole-core hydrogen generation for the Mark-B10K fuel was estimated to be less than 0.6 percent for all cases. The maximum whole-core hydrogen generation for the Mark-B10A fuel was estimated to be less than 0.4 percent for all cases. For the SBLOCA analyses (Reference 11), the maximum whole-core hydrogen generation rate was estimated to be less than 0.1 percent.

The LOCA cases performed and documented in References 6, 10, and 11 cover the entire range of possible power distributions and fuel burnup conditions that can occur in the plant. The maximum possible oxidation increase that can occur during a LOCA has been enveloped for the DBNPS, and a significant margin has been demonstrated to the one percent limit contained in the third criterion of 10 CFR 50.46. Therefore, this criterion is satisfied.

4. Core Geometry

The fourth acceptance criterion of 10 CFR 50.46 states that calculated changes in core geometry shall be such that the core remains amenable to cooling. The RELAP5/MOD2-B&W PCT calculations directly assess the alterations in core geometry at the most severe location in the core that result from a LOCA. These calculations demonstrate that the fuel pin cooled successfully. Clad swelling and flow blockage due to rupture can be estimated based on the models presented in Reference 24. For the DBNPS, the hot assembly flow area reduction at rupture is less than 51 percent for all large break LOCA cases, while the small break LOCA cases did not predict rupture. Furthermore, the upper limit of possible channel blockage, based on Reference 24, is less than 90 percent. Neither 90 percent blockage nor 51 percent blockage constitutes total subchannel obstruction. Since the position of rupture in a fuel assembly is distributed within the upper part of a grid span, subchannel blockage will not become coplanar across the assembly. Therefore, the assembly retains its pin-coolant-channel arrangement and is capable of passing coolant along the pin to provide cooling for all regions of the assembly.

The effects of fuel rod bowing on whole-core blockage are considered in the fuel assembly and fuel rod designs, which minimize the potential for rod bowing. The minor adjustments of fuel pin pitch due to rod bowing do not alter the fuel assembly flow area substantially, and the average subchannel flow area is preserved. Therefore, due to the axial distribution of blockage caused by rupture, no coplanar blockage of the fuel assembly will occur, and the core will remain amenable to cooling. Deformation of the fuel pin lattice at the core periphery is allowed to occur from the combined mechanical loading of the LOCA and a seismic event. Using leak-before-break (LBB) methodology, the spacer grid impact loads are within the spacer grid elastic load limit and no permanent grid deformation is predicted (Reference 17). Therefore, the coolable geometry requirements are met for all fuel assemblies within the core.

The consequences of both thermal and mechanical deformation of the fuel assemblies in the core have been assessed, and the resultant deformations have been shown to maintain coolable core configurations. Therefore, the coolable geometry requirements of 10 CFR 50.46 have been met and the core has been shown to remain amenable to core cooling.

5. Long-Term Cooling

The fifth acceptance criterion of 10 CFR 50.46 states that the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. Successful initial operation of the ECCS is shown by demonstrating that the core is quenched, and the cladding temperature is returned to near saturation temperature. Thereafter, long-term cooling is achieved by the pumped injection systems. These systems are redundant and are able to provide a continuous flow of cooling water to the core fuel assemblies so long as the coolant channels in the core remain open.

Compliance with this criterion is demonstrated for the systems and components specific to the DBNPS. The initial phase of core cooling has been shown to result in low cladding and fuel temperatures. A pumped-injection system capable of recirculation is available and operated by plant personnel to provide extended coolant injection. For a cold leg break, the uncontrolled concentration of boric acid within the core is postulated to be capable of forming crystalline precipitation, which could prevent the coolant from reaching certain portions of the core. The concentration of dissolved solids has been shown to be limited to acceptable levels through the use of hot leg injection through the pressurizer spray or by use of the backup decay heat drop line flow path to provide flow from the RCS hot leg to the suction of a DHR/LPI pump (Reference 21). This methodology and limitations are described in Reference 27 and 28.

The Mk-B10K fuel assemblies have fine-mesh debris resistant lower end fittings that are intended to catch debris on the upstream face, preventing incursion into the core. There is some concern that excessive flow resistance or blockage of the core flow area may occur from use of these filters. An examination of the consequences of incorporating debris resistant filters was conducted for post-accident scenarios (Reference 25). No adverse affects were determined. Therefore, the capability of long-term cooling has been established and appropriate compliance to 10 CFR 50.46 has been demonstrated.

V. Compliance with RELAP5/MOD2-B&W EM SER Restrictions

The NRC Safety Evaluation Report (SER) on BAW-10192-PA contained eleven restrictions related to the use of the RELAP5/MOD2-B&W EM. Compliance with these eleven restrictions is demonstrated in Reference 18 and summarized below. The NRC SER on BAW-10227-PA contained no additional restrictions pertaining to LOCA associated with the use of the M5 cladding material.

1. The LOCA methodology should include any NRC restrictions placed on the individual codes used in the EM.

   Response: Sections 2.2 through 2.5 of Reference 18 detail the NRC restrictions placed on the codes used in the BWNT LOCA EM. For LBLOCA analyses, the RELAP5/MOD2-B&W (includes BEACH), the REFLOD3B and CONTEMPT codes are utilized. For SBLOCA analyses, only the RELAP5/MOD2-B&W code is utilized. All of these codes used in the BWNT LOCA EM comply with the NRC restrictions.

2. The guidelines, code options, and prescribed input specified in Tables 9-1 and 9-2 in both Volume I and Volume II of BAW-10192-P should be used in LBLOCA and SBLOCA EM applications, respectively.

   Response: Table 9-1 in Volume I (LBLOCA) of BAW-10192PA is verified via use of Table 4 in Reference 18. Compliance to the Table 4 restrictions for the LBLOCA analyses is listed in Appendix D of Reference 6. The only exception to these inputs is any M5 cladding related inputs or the minimum reactor coolant pump (RCP) degradation model. Table 9-2 in Volume II (SBLOCA) of BAW-10192-PA is verified via use of Table 6 in Reference 18. Compliance to the Table 6 restriction for the SBLOCA analyses is listed in Reference 11. These tables also include input and restrictions placed on the individual codes that make up the BWNT LOCA EM as discussed in detail in Reference 18.

3. The limiting linear heat rate for LOCA limits is determined by the power level and the product of the axial and radial peaking factors. An appropriate axial peaking factor for use in determining LOCA limits is one that is representative of the fuel and core design and that may occur over the core lifetime. The radial peaking factor is then set to obtain the limiting linear heat rate. For this demonstration, calculations were performed with the axial peak of 1.7. The general approach is acceptable for demonstrating the LOCA limits methodology. However, as future fuel or core designs evolve, the basic approaches that were used to establish these conclusions may change. FTI must revalidate the acceptability of the EM peaking methods if: (1) significant changes are found in the core elevation at which the minimum core LOCA margin is predicted or (2) the core maneuvering analyses radial and axial peaks that approach the LOCA LHR limits differ appreciably from those used to demonstrate Appendix K compliance.

   Response: This restriction is related only to LBLOCAs. The axial and radial peaks used in the Reference 6 and 10 analyses were similar and approximately 1.7

for all elevations and linear heat rates analyzed. The restriction states that FTI must revalidate the acceptability of the EM peaking methods if: (1) significant changes are found in the core elevation at which the minimum core LOCA margin is predicted or (2) the core maneuvering analyses radial and axial peaks that approach the LOCA LHR limit differ appreciably from those used to demonstrate Appendix K compliance.

Reference 19 defines several layers of screening criteria needed to show compliance with the BWNT LOCA EM restriction on peaking. The methods provided are valid for any current or past Mark-B fuel type (including but not limited to Mark-B4Z, Mark-B8, Mark-B9, Mark-B10) that is ruptured-node limited or has similar ruptured- or unruptured-node PCTs predicted with the BWNT LOCA EM.

Four criteria were developed in Reference 19 from which to show compliance or to define a LOCA linear heat rate (LHR) limit penalty associated with LHR limits calculated based on the RELAP5 EM (Reference 5). These criteria are summarized as follows:

a. The fuel burnup must be compared to the LOCA LHR limits versus burnup. If the burnup is on the PCT-limited portion of the LOCA limit curve, then proceed to Step 2. If the burnup range is on the pin-pressure-limited portion of the curve, the restriction is met without any other conditions. That is, no axial peaking checks or linear heat rate limit adjustments are needed for pin pressure limited LHRs.

b. If the burnup is on the PCT-limited portion of the curve, then the power distribution analysis LOCA margins must be checked at all core elevations. If there is less than 5% LOCA margin, proceed to Step 3. If there is more than 5% margin, the restriction is met and no further checks are needed because the PCT at the maximum power distribution LHR will be lower than the BWNT LOCA EM PCT.

c. If the burnup is on the PCT-limited portion of the curve and there is less than 5% LOCA margin, then variations in the augmented peaking factor versus the 1.7 axial used in the LOCA analyses must be considered. The axial peak must be 1.65 or greater for 0 to 4 ft power peak elevations, 1.7 ± 0.5 for 4 to 8 ft elevations, and 1.75 or less for 8 to 12 ft elevations. If these axial peaks are in compliance, the restriction is met and no further checks are needed. If they are not met, then proceed to Step 4 for the LOCA LHR limit reductions.

d. If the burnup is on the PCT-limited portion of the curve, there is less than 5% LOCA margin, and the axial peak is not in compliance, then the power distribution analysis must assign a LOCA LHR limit penalty to ensure that the BWNT LOCA EM PCT (based on the given LHR and APF of 1.7) is not underpredicted. The LHR limit penalty compensates for the known deviation between the augmented axial peak and the required peak. The LHR limit reductions, ΔLHR, are core elevation dependent:

<u>At 2772 MWt</u>

$\Delta LHR_{0\rightarrow 4\,ft} = 0.0$

$\Delta LHR_{4\rightarrow 12\,ft} = \min\{0.0, [(1.75 - APF_{power\ distribution\ analysis\ augmented\ peak})*5.0\ kW/ft]\}$

(NOTE: if APF > 1.75, there is a penalty.)

<u>At 2966 MWt</u>

$\Delta LHR_{4\rightarrow 12\,ft} = \min\{0.0, [(APF_{power\ distribution\ analysis\ augmented\ peak} - 1.55)*1.5\ kW/ft]\}$

$\Delta LHR_{4\rightarrow 12\,ft} = \min\{0.0, [(1.75 - APF_{power\ distribution\ analysis\ augmented\ peak})*5.0\ kW/ft]\}$

4. The mechanistic ECCS bypass model is acceptable for cold leg transition (0.75 ft^2 to 2.0 ft^2) and hot leg break calculations. The nonmechanistic ECCS bypass model must be used in the large cold leg break (≥ 2.0 ft^2) methodology since the demonstration calculations and sensitivities were run with this model.

   Response: As outlined in BAW-10192-PA Volumes I and II, different bypass models are used for large break and small break analyses. The nonmechanistic ECCS bypass model is used in large break analyses (≥ 2.0 ft^2). The mechanistic ECCS bypass model is used for cold leg transition (0.75 ft^2 to 2.0 ft^2), hot leg, and all smaller sized cold leg breaks.

5. Time-in-life LOCA limits must be determined with, or shown to be bounded by, a specific application of the NRC-approved EM.

   Response: Time-in-life cases were explicitly examined for the LBLOCA analyses. Conditions appropriate to the specific time in life were used in the hot channel, while the BOL parameters were maintained in the average channel.

   Time-in-life calculations for SBLOCA applications are not required unless the fuel pin heatup is sufficient to cause cladding rupture. FTI evaluates the likelihood of rupture by analyzing the SBLOCA with a composite set of pin conditions that provide a conservative PCT prediction. End-of-life pin pressures are used to maximize the cladding hoop stresses, thereby improving the likelihood of rupture for those cases that do experience heatup. To maximize the cladding temperatures, the beginning-of-life (BOL) fuel stored energy and BOL oxide thicknesses are used.

However, any case that predicts clad rupture with these conditions would be further parameterized by adjusting the time of rupture (via pin pressure or normalized heating ramp rate changes) to push rupture to the time of peak clad temperature. This composite method ensures that the calculated PCT will bound any PCT predicted by a consistent time in life (TIL) analysis with appropriate TIL pin parameters. A pure TIL calculation (with fuel stored energy, pin pressure, and cladding oxide thickness consistent with the TIL that produces the worst rupture time) would be performed if the composite case is judged to be overly conservative. The consistent case would also use the plastic-weighted normalized heating ramp rate to predict the fuel pin swell and rupture performance.

SBLOCA sensitivity studies performed and documented in Reference 22 indicated that for PCTs less than 1606 °F, the most limiting PCT results are produced when rupture is not predicted because rupture tends to cool the node. Reference 26 studies showed that with PCTs near 1800 °F, the ruptured node becomes limiting because of the metal-water energy contribution. While these assertions are based on studies performed with Zr-4 cladding, they are equally applicable to M5 cladding, because the rupture behavior and metal-water reaction are not significantly different between the cladding materials. The SBLOCA analyses from Reference 11 used a constant normalized heating ramp rate limit of one to minimize the likelihood of cladding rupture. Since the limiting SBLOCA PCT was below 1606 °F, limiting PCT results are assured. The possibility of cladding rupture can not be ruled out, however the PCT predicted from a ruptured node condition would not be limiting.

6. LOCA limits for three pump operation must be established for each class of plants by application of the methodology described in this report. An acceptable approach is to demonstrate that three pump operation is bounded by four pump LHR limits.

   Response: LBLOCA analysis of a three-pump case at a core power in the range of 65 to 85 percent is performed (or reference to a three-pump analysis for a similar plant design is made) to demonstrate that three-pump operation is bounded by four-pump LHR limits. The hot channel three-pump peak LHR limit is set equivalent to the 100 percent power 4-pump LHR limit. Because this analysis is performed at a power level less than 95 percent, it must consider the possibility of positive moderator temperature coefficient (MTC). A 75 percent full power three-pump analysis with a +1.0 pcm/F MTC was performed for the DBNPS in Reference 8 to show that the altered configuration would not predict more limiting results than the 4-pump 100 percent full power case. These results concluded that the 4-pump LHR limits are appropriate for 3-pump LHR limits at 75 percent power and an MTC of +1.0 pcm/F or less for the 177-FA RL plants.

Three-pump SBLOCA analyses are not performed, because the core power is reduced but the ECCS capacity remains at the 100 percent full power levels. Therefore, four-pump full-power SBLOCA PCTs will bound the PCTs for similar three-pump partial power cases.

7. The limiting ECCS configuration, including minimum versus maximum ECCS, must be determined for each plant or class of plants using this methodology.

   Response: This restriction is primarily related to LBLOCAs. The main problem in SBLOCAs is loss of RCS inventory, therefore, a single ECCS train is always more limiting than two ECCS trains. The minimum containment pressure derived from a maximum ECCS flow for LBLOCAs was shown in the EM topical (Reference 3) to be more limiting in some cases. A study was performed (Reference 14) which determined that for the DBNPS, maximum ECCS flow is limiting for LBLOCA analyses.

8. For the small break model, the hot channel radial peaking factor to be used should correspond to that of the hottest rod in the core, and not to the radial peaking factor of the 12 hottest bundles.

   Response: There are twelve assemblies modeled in the hot bundle, and each pin is peaked to the hot pin radial value.

9. The constant discharge coefficient model (discharge coefficient = 1.0) referred to as the "High or Low Break Voiding Normalized Value," should be used for all small break analyses. The model which changes the discharge coefficient as a function of void fraction, i.e., the "Intermediate Break Voiding Normalized Value," should not be used unless the transient is analyzed with both discharge models and the intermediate void method produces the more conservative result.

   Response: This restriction is related only to SBLOCA analyses. A constant discharge coefficient is used for SBLOCA analyses. Verification of this input is performed for each SBLOCA analysis.

10. For a specific application of the FTI small break LOCA methodology, the break size which yields the local maximum PCT must be identified. In light of the different possible behaviors of the local maximum, FTI should justify its choice of break sizes in each application to assure that either there is no local maximum or the size yielding the maximum local PCT has been found. Break sizes down to 0.01 ft^2 should be considered.

    Response: This restriction is related only to SBLOCA analyses. The SBLOCA break spectrum (down to at least 0.01 ft^2) is performed to determine the local maximum PCT. The break sizes analyzed are chosen to ensure that the local peak has been appropriately defined. The full spectrum of break sizes performed for the DBNPS covers this requirement.

11. B&W-designed plants have internal reactor vessel vent valves (RVVVs) that provide a path for core steam venting directly to the cold legs. The BWNT LOCA EM credits the RVVV steam flow with the loop steam venting for LBLOCA analyses. The possibility exists for a cold leg pump suction to clear during blowdown and then reform during reflood before the EM analyses predict average core quench. Since the REFLOD3B code cannot predict this reformation of the loop seal, FTI is required to run the RELAP5/MOD2-B&W system model until the whole core quench, to confirm that the loop seal does not reform. This demonstration should be performed at least once for each plant type (raised loop and lowered loop) and be judged applicable for all LBLOCA break sizes.

    Response: This restriction is related only to LBLOCA analyses. This verification analysis was performed using the RELAP5 system model for the DBNPS in Reference 12. The results of that analysis confirmed that a loop seal does not reform prior to whole core quench.

VI. Conclusion

In conclusion, FENOC has demonstrated that the RELAP5 methodology as outlined in BAW-10192-PA and BAW-10227-PA is appropriate for use in the DBNPS LOCA analyses. This Enclosure addresses the restrictions and conditions imposed by the associated SERs as well as each of the five 10 CFR 50.46 ECCS criterion. This submittal is notification that the approved RELAP5 methodology is being used for the LBLOCA and SBLOCA analysis of record for the DBNPS.

VII. References

1. FTI Topical Report, "B&W's ECCS Evaluation Model," BAW-10104-PA, Rev. 5, November 1986.

2. B&W Topical Report, "ECCS Evaluation of B&W's 177-FA Raised-Loop NSS," BAW-10105, Rev. 1, July 1975.

3. FTI Topical Report, "B&W's Small-Break LOCA ECCS Evaluation Model," BAW-10154-P, Rev. 0, November 1982.

4. B&W Topical Report, "Multinode Analysis of Small Breaks for B&W's 177-Fuel-Assembly Nuclear Plants with Raised Loop Arrangement and Internals Vent Valves," BAW-10075A, Rev. 1, March 1976.

5. FTI Topical Report, "BWNT LOCA Evaluation Model for OTSG Plants," BAW-10192-PA, Rev. 0, June, 1998.

6. FTI Document 32-5004708-00, "D-B1 Mk-B10k LOCA Limits," FTI Proprietary.

7. FTI Document 32-5005804-00, "DB-1 Mk-B10k Gad Analysis," FTI Proprietary.

8. FTI Document 32-5005425-00, "177-RL 3-Pump Partial Power," FTI Proprietary.

9. FTI Document 32-5004974-00, "Davis Besse Partial Power MTC Study," FTI Proprietary.

10. FTI Document 32-5005503-00, "DB-1 Mk-B9A R5/M2 LOCA Limits."

11. FTI Document 32-5004328-00, "DB MkB-10K SBLOCA Spectrum Analysis," FTI Proprietary.

12. FTI Document 32-5004922-00, "177-RL RELAP5 Loop Seal Clearing," FTI Proprietary.

13. FTI Document 32-5002653-00, "Davis-Besse 1 RC Pump Study," FTI Proprietary.

14. FTI Document 32-5003372-00, "D-B1 ECCS Configuration Study," FTI Proprietary.

15. FCF Document 86-1267229-01, "Pre and Post LOCA EDFs," FTI Proprietary.

16. FCF Topical Report, "Extended Burnup Evaluation," BAW-10186-PA, June 1997.

17. FCF Document 32-5006837-00, "Mk-B10K Fuel Rod Faulted Analysis."

18. FTI Document 51-5001731-00, "BWNT LOCA EM Limitations and Restrictions."

19. FTI Document 51-5004541-00, "Radial vs Axial Core Peaking."

20. FTI Document 43-10192Q-04, "BWNT LOCA Requests/Ad.Inform."

21. FTI Document 86-5006059-00, "Post-LOCA Boron Concentration Management for DB-1."

22. FTI Document 32-1234842-01, "Oconee Mk-B11 75% FP SBLOCA."

23. FTI Document 32-5006465-00, "D-B 1 CFT Configuration Study."

24. FCF Topical Report, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," BAW-10227-PA, February 2000.

25. FTI Document 51-5006320-00, "Debris Filter Design Criteria LOCA & SA."

26. FTI Document 32-1234842-02, "Oconee Mk-B11 75% FP SBLOCA."

27. Letter, FirstEnergy to NRC, "Request for Exemption from 10 CFR 50, Appendix K, for Boric Acid Precipitation Control Methodology (TAC No. MA7831)," Serial Number 2633, March 15, 2000.

28. Letter, FirstEnergy to NRC, "Supplemental Information Regarding the Request for Exemption from 10 CFR 50, Appendix K, for Boric Acid Precipitation Control Methodology (TAC No. MA7831)," Serial Number 2652, April 3, 2000.

29. FTI Document 86-1176204-01, "DB-1 Cycle 7 LOCA Limits."

30. FTI Document 86-1224873-00, "Mark-B10A LL Spectrum Study."

31. FTI Document 32-1173541-02, "SBLOCA 2% SS Power Error Assessment."

COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8466) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	**DUE DATE**
None	N/A