Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

November 10, 1999
3F1199-01

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

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

Reference: FPC to NRC letter, 3F0199-02, dated January 7, 1999, "Change in Analysis of Record for Small Break Loss of Coolant Accident and 10CFR50.46 Notification"

Dear Sir:

Pursuant to 10 CFR 50.46(a)(3), Florida Power Corporation (FPC) hereby provides notification of a greater than 50°F decrease in the calculated peak clad temperature (PCT) for the small break loss of coolant (SBLOCA) analysis for Crystal River Unit 3 (CR-3). The change in PCT is a result of reanalysis of the SBLOCA to reflect modifications of the Engineered Safeguards Actuation System and High Pressure Injection system during the current refueling outage. A summary of the changes to the SBLOCA analysis is provided as Attachment A.

Additionally, this submittal provides notification pursuant to 10 CFR 50.46(a)(3) of a less than 50°F decrease in the calculated PCT for the large break LOCA (LBLOCA) analysis for CR-3. FPC has replaced the CRAFT2 evaluation model (EM) with RELAP5/MOD2-B&W (RELAP5) EM as the code of record for the CR-3 LBLOCA. FPC had previously changed the SBLOCA EM to RELAP5/MOD2-B&W (RELAP5) in the referenced letter. The transition from CRAFT2 to RELAP5 for the LBLOCA involves a decrease in PCT of approximately 40°F for the limiting LBLOCA transient. Attachment B provides a summary of the evaluation that concludes RELAP5 is acceptable for use in CR-3 specific LBLOCA applications.

These attachments also provide information that demonstrate continued compliance with the emergency core cooling system requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K. No further action is required by FPC and no NRC action is requested. This letter establishes no new regulatory commitments.

A001

PDR ADOCK 05000302 P

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Manager, Nuclear Licensing at (352) 563-4883.

Sincerely,

Daniel L. Roderick
Director, Nuclear Engineering and Projects

DLR/dah

xc: Regional Administrator, Region II
NRR Project Manager
Senior Resident Inspector

Attachments:

A. Change in SBLOCA Analysis to Reflect Modifications of ESAS and HPI
B. Change of Evaluation Model Methodology for LBLOCA

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

3F1199-01

ATTACHMENT A

CHANGE IN SBLOCA ANALYSIS
TO REFLECT MODIFICATIONS OF ESAS AND HPI

CHANGE IN SBLOCA ANALYSIS TO REFLECT MODIFICATIONS OF ESAS AND HPI

I. Introduction

A. Purpose and Overview

The purpose of this attachment is to provide notification to the NRC in accordance with 10 CFR 50.46(a)(3) of a peak clad temperature (PCT) decrease of greater than 50°F for the small break loss of coolant accident (SBLOCA) evaluation model (EM) for Crystal River Unit 3 (CR-3). A reanalysis of the SBLOCA has been performed to reflect modifications being made to the Engineered Safeguards Actuation System (ESAS) and High Pressure Injection (HPI) system. The calculated PCT for the limiting SBLOCA transient has decreased by approximately 180°F using the EM based on the RELAP5/MOD2-B&W code (Reference 1).

This attachment provides a summary of the change in the analysis for the SBLOCA at CR-3. The modifications, which are being performed during the current refueling outage, include an increase in the ESAS reactor coolant system (RCS) low pressure actuation setpoint for HPI, as well as changes to the HPI flow paths which enhance the delivery of pumped injection.

As background, this attachment provides information on the emergency core cooling system (ECCS) acceptance criteria, and the methods for SBLOCA analyses. A discussion is also included comparing the current ESAS and ECCS configurations described in the CR-3 Final Safety Analysis Report with the modified configurations. A comparison of the PCT for the current and modified configurations is also provided for the limiting SBLOCA scenarios.

B. Background

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 must meet the following criteria:

1. The calculated PCTs are less than 2200°F.
2. The maximum calculated local cladding oxidation is less than 17%.
3. The maximum calculated core-wide oxidation does not exceed 1% of the fuel cladding.
4. The cladding remains amenable to cooling.
5. Long-term cooling must be established and maintained after the LOCA.

The first four criteria are demonstrated by analytical methodology, while the last criterion is demonstrated by the combination of ECCS performance, equipment availability, and operational practices.

The classic distinction between the SBLOCA and LBLOCA has been one of break size. In the RELAP5 based EM, the SBLOCA break size ranges from 0.75 ft^2 to smaller areas whose flowrate exceeds the makeup capability. This EM uses a LBLOCA transition break methodology between break sizes of 0.75 and 2.0 ft^2 and a pure LBLOCA methodology above 2.0 ft^2. Spectra of break sizes are then analyzed to determine the most limiting break size. This attachment focuses on the SBLOCA analyses using the RELAP5-based EM.

A SBLOCA results in a relatively slow RCS depressurization. The transition from a period of relatively high core flow to one of more quiescent flow is fairly smooth. During the quiescent period, a two-phase froth may develop above, and then in, the core region. The ECCS is designed to provide sufficient flow to ensure that at a minimum, core decay heat energy removal is assured, and that the 10 CFR 50.46 criteria are met. To demonstrate compliance with 10 CFR 50.46, Babcock and Wilcox (now Framatome Technologies, Incorporated) developed ECCS performance analysis methodologies for SBLOCA analyses (Reference 1). A methodology based on the RELAP5/MOD2-B&W code and EM has since been approved for use in licensing applications.

C. Engineered Safeguards Actuation System and Emergency Core Cooling System (HPI) Modifications

A description of modifications that were to be performed during refueling outage 11 (RFO-11) for ESAS and HPI were provided to the NRC as part of License Amendment Request (LAR) #241 (Reference 4). LAR #241 has been subsequently approved by the NRC as Amendment No. 178 (Reference 5).

Previously, the CR-3 Improved Technical Specification ESAS setpoint on RCS low pressure for HPI actuation was 1500 psig. The HPI flow paths were configured such that operator action was necessary to maximize injected flow, such as the action to isolate reactor coolant pump (RCP) seal injection following ESAS initiation. Moreover, if the break was at an HPI injection point, it became necessary to isolate the broken line. Additionally, certain single failures would require operator action, again to ensure the maximum available flow to the RCS. While this configuration did perform its accident mitigation function, significant operational and analytical improvements have been made through design modifications.

As described in Reference 4, these modifications have been focused on both the ESAS HPI actuation setpoint and the HPI piping. The ESAS RCS low pressure actuation setpoint has been raised to 1625 psig and flow path changes (valves and piping) have been made to the HPI system. The HPI flow path now includes features such as cross-tie lines, automatic isolation of makeup and RCP seal injection, and redistribution of flow resistance. These changes ensure the timely actuation of ESAS, as well as the improved delivery of HPI flow. In combination, the modifications result in decreased reliance on operator action, as well as improved analytical results.

II. Limitations and Conditions

The NRC Safety Evaluation Report on BAW-10192-P (Reference 2) contained eleven conditions related to the use of the RELAP5/MOD2-B&W EM. Each of these conditions was addressed in Reference 6 when CR-3 revised the EM for the SBLOCA to RELAP5/MOD2-B&W (RELAP5). The changes to the analysis associated with the ESAS and HPI modifications do not change the CR-3 compliance to the limitations and conditions as described in Reference 6.

III. Summary of Key Results

The use of the RELAP5/MOD2-B&W methodology for this SBLOCA analysis (Reference 3) demonstrated compliance with the analytical criteria of 10 CFR 50.46. A comparison of the results for the limiting Cold Leg Pump Discharge break size is shown below:

Parameter	Limit	Pre-Modification	Post-Modification
Break Size (ft^2)	-	0.120	0.07
Peak Clad Temperature (°F)	< 2200	1583	1400
Local Oxidation (%)	< 17	0.646	0.4267
Whole Core Oxidation (%)	< 1	0.053	0.0154

The core geometry also remained amenable to cooling, since resultant fuel deformations were found to maintain coolable configurations. Finally, long term core cooling is assured through demonstrating that the core is quenched, that pumped injection is available, and that long-term cooling challenges (e.g., post-LOCA boron precipitation) have been addressed.

IV. Conclusion

In conclusion, the reanalysis of the SBLOCA has shown that CR-3 maintains compliance with the requirements of 10 CFR 50.46 for ECCS.

V. References

1. Framatome Technologies, Inc. Topical Report, BAW-10192P-A Revision 0, "BWNT LOCA Evaluation Model for OTSG Plants," June 1998.

2. Letter from J. E. Lyons (NRC) to J. H. Taylor (FTI), "Acceptance for Referencing of Topical Report BAW-10192-P, "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," February 18, 1997.

3. Florida Power Corp. Calculation, F-98-0008 Revision 2, "SBLOCA Analysis for HPI Upgrades," (FTI Document 86-5001942-02).

4. FPC to NRC letter, 3F1198-03, dated November 23, 1998, "License Amendment Request #241, Revision 0, High Pressure Injection System Modifications."

5. NRC to FPC letter, 3N0599-11, dated May 21, 1999, "Crystal River 3 – Issuance of Amendment Regarding Reactor Protection System and Engineered Safeguards Actuation System Setpoints and Surveillance Requirements (TAC No. MA3614)."

6. FPC to NRC letter, 3F0199-02, dated January 7, 1999, "Change in Analysis of Record for Small Break Loss of Coolant Accident and 10 CFR50.46 Notification."

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

3F1199-01

ATTACHMENT B

CHANGE OF EVALUATION MODEL METHODOLOGY FOR LBLOCA

CHANGE OF EVALUATION MODEL METHODOLOGY FOR LBLOCA

I. Introduction

A. Purpose and Overview

The purpose of this attachment 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) (decrease of less than 50°F) for the large break loss of coolant accident (LBLOCA) analyses for Crystal River Unit 3 (CR-3). 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 CR-3 LBLOCA analyses.

The primary emphasis of this attachment is to demonstrate that the CR-3 specific application of RELAP5 for LBLOCA analyses meets the limitations and conditions of the NRC Safety Evaluation Report (SER) for the RELAP5 EM. A comparison of the CRAFT2 and RELAP5 results regarding the 10 CFR 50.46 acceptance criteria is also provided for the limiting LBLOCA scenario (Double Ended Guillotine Rupture at Cold Leg Pump Discharge). The change in the analyses to RELAP5 results in a decrease in peak clad temperature (PCT) for the limiting transient of approximately 40°F.

CR-3 has previously changed to a RELAP5 based EM for small break LOCAs (SBLOCA) as detailed in Reference 1.

B. Background

The function of the Emergency Core Cooling System (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 maximum calculated PCT is less than 2200°F.
2. The maximum calculated local cladding oxidation is less than 17%.
3. The maximum calculated core-wide oxidation does not exceed 1% of the fuel cladding.
4. The cladding remains amenable to cooling.
5. Long-term cooling must be established and maintained after the LOCA.

The first four criteria are demonstrated by analytical methodology, while the last criterion is demonstrated by the combination of ECCS performance, equipment availability, and operational practices.

The CR-3 LBLOCA methodology has been based on the CRAFT2 code, as described in Reference 2. The CRAFT2-based analyses represented conservative and valid licensing results based on the boundary conditions analyzed. However, these analyses have been replaced with new analyses that use the improved calculational methods contained in the RELAP5 EM. The RELAP5 EM, described in References 3 and 4, was devised to provide flexibility for addressing new fuel designs, and to accommodate component performance issues (e.g., reduced ECCS flows, steam generator tube plugging, etc.). The RELAP5 EM also includes requirements that are generally more restrictive than those used to license the CRAFT2-based EM. The transition to the improved RELAP5 calculational methods and meeting the prescribed requirements for the use of the new EM provide assurance that the CR-3 LBLOCA analyses remain valid and demonstrate continued compliance with the 10 CFR 50.46 acceptance criteria.

II. Limitations and Conditions

NRC Safety Evaluation Report (SER) on BAW-10192-P (Reference 3) contains restrictions and conditions related to the use of the RELAP5/MOD2-B&W EM. Compliance with these restrictions and conditions is demonstrated in References 5 and 10 and is discussed below.

A. Restrictions on BWNT LOCA Evaluation Model: BAW-10192-P

1. *The LOCA methodology should include any NRC restrictions placed on the individual codes used in the evaluation model (EM).*

   The NRC has approved the series of codes used in the BWNT LOCA EM and has issued SERs for each. Sections 2.2 through 2.5 of Reference 10 describe the NRC restrictions contained in the SER for the RELAP5/MOD2-B&W (Reference 6), BEACH (Reference 7), REFLOD3B (Reference 8) and CONTEMPT (Reference 9) codes used in the BWNT LOCA EM and demonstrate how each restriction is checked for appropriate usage.

   SER restrictions were incorporated into generic checklists documented in Tables 4 through 7 of Reference 10. These tables were completed for the specific LBLOCA analyses summarized in Reference 5. Each restriction has been validated by the controlled code input and by use of the appropriate LOCA checklists.

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-10192P should be used in LBLOCA and SBLOCA evaluation model applications, respectively.*

   The guidelines, code options, and prescribed input specified in Tables 9-1 and 9-2 of Volume I of BAW-10192P (Reference 4) for use in the LBLOCA analyses have been verified to be included in the analyses summarized in Reference 5.

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 evaluation model 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.

The axial and radial peaks used in the Reference 5 analyses were similar and approximately 1.7 for all elevations and linear heat rates analyzed. The core maneuvering analyses for each reload ensure the minimum predicted LOCA margin and the predicted LOCA peaks using cycle specific radial and axial peaking factors at the limits of normal operation do not differ appreciably from the linear heat rate (LHR) limits used to demonstrate 10 CFR 50.46 and Appendix K compliance.

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 ($\geq$ 2.0 ft^2) methodology since the demonstration calculations and sensitivities were run with this model.*

   As outlined in BAW-10192 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 ($\geq$ 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. The CR-3 LBLOCA analyses (Reference 5) used the nonmechanistic model.

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

   Time-in-life cases were explicitly examined for this application. Conditions appropriate to the specific time in life were used in the hot channel, while the beginning-of-life (BOL) parameters were maintained in the average channel. The results demonstrated that the BOL LHR limits can be maintained up to 45 GWd/mtU. From 45 GWd/mtU to 60 GWd/mtU the LHR limit is reduced linearly to 11.7 kW/ft in accordance with the TACO3 pressure limit of 800 psi above system pressure.

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.*

   Analyses have been performed to demonstrate that three pump operation is bounded by four pump LHR limits. Specifically, these analyses concluded that the four pump LHR limits are appropriate for three pump LHR limits at 80 percent power and a moderator temperature coefficient of +1.0 pcm/°F or less for the 177-FA lowered-loop (LL) plants.

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

   For this application, the minimum containment pressure derived from a maximum ECCS flow was used in the LOCA analysis that considered minimum ECCS injection. This composite approach conservatively considers the worst containment pressure with the minimum ECCS refill capacity to ensure the LBLOCA calculated consequences are bounding for any combination of available ECCS pumps.

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.*

   This restriction is related to SBLOCAs and is not applicable to the CR-3 LBLOCA analyses.

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.*

   This restriction is related to SBLOCAs and is not applicable to the CR-3 LBLOCA analyses.

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.*

    This restriction is related to SBLOCAs and is not applicable to the CR-3 LBLOCA analyses.

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 evaluation model credits the RVVV steam flow with the loop steam venting for LBLOCA analyses. The possibility exists for a cold leg pump suction seal to clear during blowdown and then reform during reflood before the evaluation model 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.

This verification analysis was performed using the RELAP5 system model for the 177-FA LL plant design and confirmed that a loop seal does not reform prior to whole core quench. These results were obtained using the 177-FA LL model and it can be concluded that Restriction #11 of the evaluation model is met for the CR-3 plant.

B. Preliminary Safety Concern 1-99

An additional condition incorporated into the analyses associated with the transition to a RELAP5 based EM was Framatome Technologies, Incorporated (FTI) Preliminary Safety Concern (PSC) 1-99. This PSC resulted from non-conservative reactor coolant pump type modeling and degradation effects on LBLOCA analyses. As documented in the PSC, the CR-3 Mark-B9 analyses (Reference 5) include the worst-case degradation differences and the M3 multiplier curve. Therefore, since the analyses which form the basis for the transition to a RELAP5 based EM incorporate the most limiting pump type and degradation curve, the issue associated with PSC 1-99 does not impact the CR-3 RELAP5 LOCA analyses.

III. <u>Summary of Key Results</u>

The use of the RELAP5 methodology for the LBLOCA is contained in Reference 5. The analyses documented in Reference 5 incorporate the requirements of 10 CFR 50 Appendix K and demonstrate compliance with the acceptance criteria of 10 CFR 50.46. A comparison of the limiting CRAFT2 and RELAP5 analyses, for the Mark-B9 fuel design, is shown below:

Parameter	Limit	CRAFT2	RELAP5
Peak Clad Temperature (°F)	< 2200	2051	2010
Local Oxidation (%)	< 17	3.95	2.47
Whole Core Oxidation (%)	< 1	< 0.557	< 0.3

The table above demonstrates 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.

IV. Conclusion

In conclusion, FPC has demonstrated that the RELAP5 methodology as outlined in BAW-10192P-A is appropriate for use in CR-3 LBLOCA analyses. This attachment addresses the restrictions and conditions imposed by the associated SERs. This submittal is notification that the approved RELAP5 methodology is being used for the LBLOCA analysis of record for CR-3.

V. References

1. FPC to NRC letter, 3F0199-02, dated January 7, 1999, "Change in Analysis of Record for Small Break Loss of Coolant Accident and 10 CFR 50.46 Notification."

2. Babcock & Wilcox Topical Report, BAW-10103A, Rev. 3, "ECCS Analysis of B&W'S 177-FA Lowered-Loop NSS," July 1977.

3. Letter from J. E. Lyons (NRC) to J. H. Taylor (FTI), "Acceptance for Referencing of Topical Report BAW-10192-P, 'BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants' (TAC NO. M89400)," February 18, 1997.

4. Framatome Technologies, Inc. Topical Report, BAW-10192P-A, Revision 0, "BWNT LOCA Evaluation Model for OTSG Plants," June 1998.

5. FTI document 86-5003297-02, "CR-3 R5/M2 LBLOCA Summary Report," August 1999.

6. J. A. Klingenfus, et al, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," BAW-10164P-A, Revision 03, Babcock & Wilcox, Lynchburg, Virginia, October 1996.

7. N. H. Shah, et al., "BEACH - A Computer Program for Reflood Heat Transfer During LOCA," BAW-10166P-A, Revision 4, B&W Nuclear Technologies, Lynchburg, Virginia, October 1992.

8. C. K. Nithianandan, "REFLOD3B - Model for Multinode Core Reflooding Analysis," BAW-10171P-A, Revision 03, Babcock & Wilcox, Lynchburg, Virginia, September 1989.

9. Y. H. Hsii, "CONTEMPT—Computer Program for Predicting Containment Pressure-Temperature Response to a LOCA," - B&W-Revised Version of Phillips Petroleum Co. Program (L. C. Richardson, et. al., June 1967), BAW-10095-A, Revision 1, Babcock & Wilcox, Lynchburg, Virginia, April 1978.

10. FTI Document 51-5001731-00, "BWNT LOCA EM Limitations and Restrictions," November 1998.