Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.90

July 31, 2003
3F0703-14

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

Subject: Crystal River Unit 3 – Non-Proprietary Version to Response to Request for Additional Information Regarding Technical Specification Change Request For New Departure From Nucleate Boiling Correlation (TAC No. MB7035)

References:
1) PEF to NRC letter, dated December 19, 2002, Crystal River Unit 3 – License Amendment Request #277, Revision 0, "BHTP Departure From Nucleate Boiling Correlation"
2) PEF to NRC letter, dated May 9, 2003, Crystal River Unit 3 – Supplement to Proposed License Amendment Request #277, Revision 0, "BHTP Departure From Nucleate Boiling Correlation"
3) NRC to PEF letter, dated May 29, 2003, Crystal River Unit 3 - Request For Additional Information Regarding Technical Specification Change Request For New Departure From Nucleate Boiling Correlation (TAC No. MB7035)
4) PEF to NRC letter, dated July 15, 2003, Crystal River Unit 3 – Response to Request For Additional Information Regarding Technical Specification Change Request For New Departure From Nucleate Boiling Correlation (TAC No. MB7035)

Dear Sir:

In References 1 and 2, Progress Energy Florida, Inc. (PEF) submitted License Amendment Request (LAR) #277 and supplemental information. In Reference 3, the NRC forwarded a request for additional information (RAI) to PEF concerning this LAR. The proprietary response to the RAI was submitted by letter dated July 15, 2003 (Reference 4). This submittal transmits a non-proprietary version of that information for inclusion in the Public Document Room and is provided in the attachment to this letter.

No new regulatory commitments are made in this letter.

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

A001

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/pei

Attachment: Response to Request for Additional Information Regarding Technical Specification Change Request For New Departure From Nucleate Boiling Correlation — Non-Proprietary Version

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

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 31st day of July, 2003, by Dale E. Young.

Janet Schroeder

Signature of Notary Public
State of Florida

JANET SCHROEDER
MY COMMISSION # DD 128063
EXPIRES: June 20, 2006
Bonded Thru Notary Public Underwriters

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT

Response to Request for Additional Information Regarding Technical Specification Change Request For New Departure From Nucleate Boiling Correlation

Non-Proprietary Version

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Proposed Responses for "Request for Additional Information, Crystal River Unit 3, License Amendment, BHTP Departure from Nucleate Boiling Correlation"

1. By supplement dated May 9, 2003, Florida Power Corporation (the licensee) provided additional information to describe the normal and accident analyses performed in support of its amendment request for Crystal River Unit 3 (CR-3). The licensee identified and analyzed the limiting Departure from Nucleate Boiling (DNB) transient for each of the Condition I, II, III, and IV events for CR-3 in lieu of reanalyzing all the transients. The NRC staff requests the licensee technically justify why each of the events is the most limiting for DNB. Additionally, the NRC staff requests that the licensee state whether it will reanalyze all of the remaining events prior to reloading the core to ensure appropriate safety limits are met.

   RESPONSE

   The limiting DNB events were identified in assessments that examined the impact of placing Mark-B-HTP fuel into the CR-3 core for the events contained in the FSAR. These included:
   a) uncompensated operating reactivity changes,
   b) startup accidents,
   c) rod withdrawal at rated power,
   d) moderator dilution,
   e) cold water injection,
   f) loss of coolant flow,
   g) stuck-out, stuck-in, or dropped-in control rod assembly,
   h) load rejection, and
   i) loss of electric power

   The incorporation of the Mark-B-HTP fuel into the CR-3 core will result in the following impact for full cores (Mark-B-HTP versus Mark-B10)
   - a slight reduction ([ ]) in the predicted reactor coolant system flow rate,
   - a slight increase ([ ]) in the core bypass flow fraction, and
   - a slight increase ([ ]) in the core pressure drop.

   These changes are attributed to the higher hydraulic resistance of the Mark-B-HTP fuel design when compared to the resident Mark-B10 fuel design. Each event was examined relative to the event initiators, acceptance criteria, and event termination to determine whether the above changes in flow and pressure drop would significantly affect the RCS response.

   After examining the events and assessing the impact of the slight changes in predicted RCS flow rate, core bypass flow fraction, and core pressure drop, it was concluded there was no significant impact

on the RCS response versus time during the event that had been used for Cycle 13 analyses, therefore, the relative DNB performance between events would remain the same with the addition of the Mark-B-HTP fuel design. That is, the events found to be the more DNB limiting using the BWC CHF correlation for Cycle 13 would remain the limiting events using the BHTP CHF correlation for Cycle 14.

The explicit DNBR calculations performed for the limiting DNB events accounted for the changes in the core bypass flow fraction and the core pressure drop. The calculations were also based on a thermal design flow rate that was well below the predicted RCS flow rate for CR-3.

Although explicit DNB calculations were only performed for the limiting DNB events, the assessments of all the events are included in the reload analysis documentation. No additional DNBR calculations are planned in supporting the non-LOCA safety analyses conclusions.

2. In its analysis of the DNB limiting accidents for Condition I/II events, the licensee stated that the analysis was done assuming a full core of Mark-B-HTP fuel. The NRC staff requests the licensee provide a technical justification that demonstrates that the assumption of a Mark-B-HTP full core results in more limiting DNB accident analyses than either the Cycle 14 transition core or a full core of Mark-B10 fuel.

   <u>RESPONSE</u>

   The CR-3 Cycle 14 transition core is more limiting with respect to DNB performance during accident analyses than either an all Mark-B10 core or an all Mark-B-HTP core. This conclusion has been reached by examining both steady-state statepoints and transient events with models of full cores of each fuel design as well as various transition core configurations including the Cycle 14 specific configuration.

   As discussed in the previously supplied response in Reference 1, the application of the assumption of a full core of Mark-B-HTP is justifiable for predicted DNB performance when the transition core penalty is explicitly determined and accommodated within the DNB margin provided with the Thermal Design Limit. The flexibility of using the DNB margin within the Thermal Design Limit for this purpose is discussed on page 5-3 of BAW-10187P-A (Reference 2). In Reference 1, the transition core penalty, based on preliminary analyses, was [ ] or [ ] DNB points where 1 DNB point = 0.01. The Thermal Design Limit of [ ] using the BHTP correlation contained [ ] DNB points of excess DNB margin for the full core Mark-B-HTP condition of which [ ] DNB points were to be committed to offsetting the DNB transition core penalty.

3. In the licensee's discussion of the limiting Condition IV event, it stated that additional "unused" DNB margin existed above the amount required to satisfy its Alternative Source Term (AST) dose evaluations. However, the licensee provided numerical data that could be interpreted as demonstrating that the amount of margin available is less than that stated. The NRC staff requests the licensee identify the minimum DNB limit for Condition IV events that ensures its AST limits are not violated.

   RESPONSE

   The minimum DNB limit for Condition IV events that ensures the Alternative Source Term (AST) dose evaluations remain bounding is [ ] BHTP. If the minimum DNB limit were higher than [ ], then more fuel failure would be predicted than what is considered in the current AST dose evaluations for CR-3. If the minimum DNB limit were lower than [ ], then less fuel failure would be predicted and the current AST dose evaluations would become more bounding.

   For CR-3 Cycle 14 the Thermal Design Limit (TDL), or DNB limit, was selected to be [ ] using the BHTP correlation. This TDL value was selected prior to the DNB analysis of the transient events. It was expected to be sufficiently large (or high) to adequately accommodate the Statistical Design Limit, the DNB transition core penalty and additional margin deemed appropriate for addressing cycle-specific needs. Since the [ ] DNB limit necessary to maintain the applicability of the current AST dose evaluation is less than the TDL of [ ], then [ ] DNB points, or [ ], of DNB margin within the TDL are being committed to offset the need for a slightly lower DNB limit.

   The following diagram shows that once the DNB analyses were performed a TDL value of [ ] was found to conservatively include more DNB margin than was actually needed for CR-3 Cycle 14. The diagram shows that a TDL as low as [ ] could have been sufficient to support CR-3 Cycle 14. However, Progress Energy elected to maintain the TDL basis of a [ ] DNB limit.

DNB Margin Within the Thermal Design Limit (TDL)
for CR-3 Cycle 14

[ ]

4. In support of its licensee amendment request, the licensee cited various topical reports used to perform its analyses. The NRC staff requests the licensee review each of the topical reports referenced and provide a list of restrictions and requirements imposed on each by the NRC during the acceptance review. The licensee should include a response to each that describes how it complied with the restriction or requirement and justify their applicability to CR-3.

RESPONSE

The SER restriction and requirements for each referenced topical report will be listed and addressed.

BAW-10156P-A, Rev. 1, "LYNXT Core Transient Thermal-Hydraulic Program", B&W Fuel Company, Lynchburg, Virginia, August 1993.

SER Restriction 1

The application of the LYNXT, Rev. 1, PV algorithm is restricted to the following ranges:

Mass flux (absolute value) - 0.0 to 3.0 mlbm/(hr-ft^2), where m denotes 10^6,
System pressure - 500 to 3000 psia,
Local heat flux - 0.0 to 0.8 mBtu/(hr-ft^2).

It is the responsibility of the licensee to verify that the proper algorithm and algorithm input parameters have been selected for the analyses performed within these ranges.

Compliance: The PV algorithm was used in the determination of the DNB predictions for CR-3 using the LYNXT code. All local coolant conditions for DNB predictions were within the mass flux, system pressure, and local heat flux application ranges identified in the SER restriction.

SER Restriction 2

When the LYNXT, Rev. 1, incorporating the PV algorithm, is used, the licensee is responsible for verifying the adequacy of the crossflow resistance chosen whenever reverse and recirculating flows are observed in the analysis.

Compliance: There were no steady-state or transient conditions evaluated with LYNXT that exhibited reverse or recirculating flow for the CR-3 DNB analyses.

SER Restriction 3

The B&W-2, BWC, BWCMV, and W3 CHF correlations may be used with either the COBRA-IV-I implicit or PV solution algorithm. The application of each correlation is restricted to its range of applicability. Application of LYNXT, Rev. 1 to another CHF correlation (other than B&W-2, BWC, BWCMV, and W3) not developed using either of LYNXT, Rev. 1's flow solution algorithms will require a separate validation process.

Compliance: The DNB predictions for the Mark-B-HTP fuel design utilize the BHTP CHF correlation. This correlation is currently being reviewed by the NRC staff. The topical report BAW-10241P, "BHTP DNB Correlation Applied Using LYNXT", Reference 3, identifies the correlation's range of applicability for fuel design characteristics and local coolant

conditions when using it with the LYNXT thermal-hydraulic code. The BHTP correlation was used within its correlation ranges defined in BAW-10241 for LYNXT DNB predictions supporting CR-3.

BAW-10187P-A, "Statistical Core Design for B&W-Designed 177FA Plants", B&W Fuel Company, Lynchburg, Virginia, March 1994.

Specifically, the hot pin statistical design limit of 1.313 is acceptable with the following restrictions:

SER Restriction 1
The component uncertainties and their distributions are to be reviewed on a plant-specific basis to determine their applicability.

Compliance: The component uncertainties and their distributions applicable for CR-3 Cycle 14 were assembled and statistically treated according to the SCD methodology defined in BAW-10187 P-A (Reference 2) using the BHTP CHF correlation. The resulting Statistical Design limit, using the BHTP correlation, for Cycle 14 was found to be different than the [ ] BWC value that was applicable for Cycle 13. If the Statistical Design Limit (SDL) determined for Cycle 14 using these uncertainties is to be applied in future CR-3 cycles (beyond Cycle 14), then the uncertainties and their distributions will be reviewed for applicability with the future cycles.

SER Restriction 2
The "bounding" assembly-wise power distribution assumed in the core-wide SDL calculation should be shown to bound the expected operating power distributions on a cycle-specific basis.

Compliance: The conservative power distribution used in the determination of the core-wide SDL resulted in the core-wide SDL of [ ] becoming slightly more limiting than the hot pin SDL. Cycle-specific core-wide SDLs were calculated using the expected operating power distributions for CR-3 Cycle 14. Final results show the [ ] SDL was substantially more limiting than the SDLs based on CR-3 cycle-specific power distributions. Therefore, it was concluded the "bounding" power distribution remained bounding. Note: The above SDL of [ ] was based on final calculations whereas the SDL of [ ] shown in the Response to Item 3 was based on preliminary calculations consistent with Reference 1. .

SER Restriction 3
All core state variables that were not included in the statistical design must continue to be input to thermal-hydraulic computer codes at their most adverse allowable level rather than at their nominal value.

Compliance: All core state variables not included in the statistical design were applied at their most adverse level in the thermal-hydraulic code model for CR-3.

SER Restriction 4
The response surface model should be validated and revised (as necessary) when applied to new fuel assembly designs and extended operating conditions, and with new computer codes and DNB correlations. The approved codes are LYNXT, LYNX1, and LYNX2, and the approved correlation is the BWC DNB correlation.

Compliance: With the introduction of the BHTP CHF correlation, the response surface model (RSM) was revised to reflect the performance of the BHTP correlation. The RSM was validated over the range of operating conditions for which the BHTP SDL was being applied for CR-3.

BAW-10143P-A, "BWC Correlation of Critical Heat Flux", Babcock & Wilcox, Lynchburg, Virginia, May 1970.

SER Restriction 1

The BWC correlation is acceptable for CHF fuel licensing analysis of B&W Mark C 17x17 geometry fuel and 15x15 Mark BZ geometry fuel, provided

a. *The range of conditions fall within the range of the database,*

b. *The local subchannel conditions (mass flux, enthalpy, quality and heat flux) are determined using the LYNX-2 subchannel analysis code.*

Compliance: The BWC CHF correlation was used for predicting DNB performance for the Mark-B10 resident fuel design in CR-3. The SER for BAW-10156P-A, Rev.00 (Reference 4) acknowledges the acceptability to use the BWC CHF correlation with the LYNXT code. The local coolant conditions for performing DNB predictions using the LYNXT code remained within the application ranges specified in Restriction 3 shown below.

SER Restriction 2

Before the BWC correlation is used with a subchannel analysis code other than LYNX-2, it must be qualified by appropriate data comparisons.

Compliance: Justification for using the BWC CHF correlation with the LYNXT code and the NRC approval are contained in BAW-10156P-A, Rev.00.

SER Restriction 3

The BWC correlation is approved for use with the following design limits and ranges of conditions:

Mark C 17x17 Fuel

Design MCHFR (minimum critical heat flux ratio)	*1.14*
Pressure (P) psia	*1600 to 2400*
Mass velocity (G), 10^6 lbm/hr-ft^2	*1 to 3.5*
Quality (X), %	*-20 to +30*

Mark BZ 15x15 Fuel

Design MCHFR	*1.18*
Pressure (P) psia	*1600 to 2600*
Mass velocity (G), 10^6 lbm/hr-ft^2	*0.43 to 3.8*
Quality (X), %	*-20 to +26*

Compliance: As stated above, the application of the BWC CHF correlation for CR-3 DNB predictions using the LYNXT code was limited to the correlation ranges defined for the Mark-BZ 15x15 fuel.

BAW-10179P-A, Rev. 04, " Safety Criteria and Methodology for Acceptable Cycle Reload Analyses", Framatome ANP, August 2001.

SER Restriction 1

... the inclusion of the following operating limits in a COLR is acceptable and BAW-10179P is an acceptable TS reference for the BWFC methodology used to establish the value of these limits. If an NRC-approved methodology change results in an update to BAW-10179P, the approved revision number at the time the COLR parameters are determined will be identified in the COLR.

(1) control rod physical insertion, sequence, and overlap limits
(2) control rod program (locations and group designations)
(3) axial power shaping rod (APSR) insertion limits
(4) axial power imbalance operating limits
(5) quadrant power tilt limits
(6) end-of-cycle moderator temperature coefficient (MTC)
(7) nuclear heat flux hot channel factor limit, F_Q *(or maximum allowable linear heat rate)*
(8) nuclear enthalpy rise hot channel factor limit, $F^N_{\Delta H}$
(9) refueling boron concentration
(10) axial power imbalance protective limits
(11) trip setpoint for nuclear overpower based on RCS flow

Compliance: The CR-3 Improved Technical Specifications (ITS) relocated the Shutdown Margin requirements for Modes 3, 4, and 5 required by LCO 3.1.1 to the COLR and also relocated the acceptance criterion for absolute position indication/relative position indication agreement required by Surveillance Requirement 3.1.7.1 to the COLR. Additional parameters have been relocated from the CR-3 ITS to the COLR since the SER for BAW-10179P-A was issued. Specifically, ITS Amendment 204 (LAR #263) relocated the following parameters to the COLR:

LCO 3.3.1 RCS Variable Low Pressure Setpoint Equation (Table 3.3.1-1, Item 5)
SR 3.4.1.1 Reactor Coolant System Pressure DNB Limits
SR 3.4.1.2 Reactor Coolant System Temperature DNB Limits
SR 3.4.1.3 Reactor Coolant System Flow DNB Limits

These core operating limits, together with ten of the eleven parameters noted above (control rod program – locations and group designations are not specified in the CR-3 COLR) are provided in the current CR-3 COLR. These operating limits are determined during the reload safety evaluation on a cycle-specific basis for each core reload using the approved methodology identified in Specification 5.6.2.18, which states that the approved revision number for BAW-10179P-A shall be identified in the COLR.

The CR-3 Cycle 13 (current cycle) core operating limits were determined based on the approved methodology specified in Revision 3 of BAW-10179P-A, as noted in the references documented in BAW-2391, Revision 1 (Crystal River Unit 3 Cycle 13 Reload

Report), dated September 2001. In compliance with Specification 5.6.2.18, Section 1.0 of the CR-3 Cycle 13 COLR notes that BAW-10179P-A Revision 3 documents the analytical methods used to determine the core operating limits.

When BAW-10241P is approved, it is anticipated that an SER for Revision 5 of BAW-10179P will be issued. Thus, Revision 5 of BAW-10179P-A (Reference 5) would include the BHTP DNB correlation and would be the appropriate revision level to specify in the COLR for reloads containing Mark-B-HTP fuel assemblies.

SER Restriction 2
For fuel designs other than Mark-B, BWFC has committed to providing a specific submittal for NRC approval of those other designs.

Compliance: The Mark-B-HTP fuel design is a variation of the Mark-B design that utilizes the HTP spacer grid design and the FUELGUARD lower end fitting design. The Mark-B design has several variations in operation within the B&W-designed 177FA plants such as the Mark-B9, Mark-B10, Mark-B11, and Mark-12.

SER Restriction 3
Also, in accordance with Appendix A to Standard Review Plan 4.2, the NRC will also require an evaluation of fuel assembly structural integrity considering the effects of seismic and LOCA loads for transition cores consisting of different fuel types using time history numerical techniques based on the plant-specific safe shutdown earthquake (SSE). These plant-specific results must show that the grids will not buckle under the combined impact forces of a seismic/LOCA event, the core coolable geometry is maintained, and the stresses resulting from the seismic/LOCA-induced deflections are within acceptable limits.

Compliance: The Mark-B-HTP and resident Mark-B fuel assemblies were evaluated for loads and stresses resulting from safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) conditions. NRC-approved methodology per BAW-10133PA Revision 1 Addendums 1 and 2 (Reference 6) including representative time history techniques and appropriate core models, were used for the evaluation. Transition core and full core configurations, representing Crystal River 3 cycles 14 and subsequent cycles, were modeled and evaluated. Results show that for all fuel types, the spacer grids will not buckle for the SSE and LOCA impact forces; the core coolable geometry is maintained; and the fuel component stresses resulting from the SSE and LOCA loadings and deflections are within acceptable limits.

SER Restriction 4
The criteria and methodology reviewed herein are applicable to B&W-designed plants. When B&W reloads a non-B&W plant additional implementation will be required.

Compliance: The criteria and methodology contained within BAW-10179, and by reference, is being applied to the B&W-designed plant of CR-3. Revision 05 of BAW-10179P includes, by reference, the BHTP CHF correlation in BAW-10241. Both revision 05 of BAW-10179P and BAW-10241P are currently being reviewed by the NRC.

5. The licensee's original submittal and its supplement did not describe how the licensee demonstrated compliance with 10CFR50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." The NRC staff requests the licensee provide information demonstrating that it has performed an appropriate analysis for its core loaded in the proposed Cycle 14 configuration.

RESPONSE

FANP performed Mark-B-HTP LOCA analyses for CR-3 using the NRC-approved BWNT LOCA Evaluation Model (BAW-10192P-A Rev. 0, Reference 7) using the methods described in the NRC-approved RELAP5/MOD2-B&W code (BAW-10164P-A Rev. 04, Reference 8). The NRC-approved methodology states that the LOCA analyses will use the same CHF correlation that is used for the fuel pin DNB analyses. The BHTP CHF correlation (Reference 3) was therefore implemented into the RELAP5/MOD2-B&W code for the Mark-B-HTP fuel assembly LOCA analyses that support CR-3 Cycle 14.

LOCA analyses for both mixed-core and whole-core configurations with the Mark-B-HTP fuel were performed to demonstrate compliance with 10 CFR 50.46. Five beginning-of-life (BOL) mixed-core LBLOCA cases with axial peaks simulated at the 2.506, 4.264, 6.021, 7.779, and 9.536 ft elevations were completed. In addition, eleven whole-core Mark-B-HTP LBLOCA analyses are simulated in accordance with the approved EM. Five BOL cases and five middle-of-life analyses, with axial peaks at the identified elevations, were performed along with one representative 2.506-ft analysis at the maximum fuel pin burnup. The LBLOCA analyses also included analysis of the 3, 6, and 8 weight percent gadolinia fuel pins. The limiting SBLOCA analyses were also analyzed to demonstrate that the Mark-B-HTP fuel will not produce a higher PCT than the previously analyzed Mark-B10 fuel.

The RELAP5/MOD2 blowdown mixed-core LBLOCA analyses completed to support Cycle 14 conservatively placed the Mark-B-HTP fuel with the higher HTP grid form losses in the hot channel and simulated the average channel with the lower resistance Mark-B10 fuel assemblies. The core bypass flow was conservatively maximized in the mixed-core analysis by simulating the core as though it was comprised entirely of higher resistance Mark-B-HTP fuel. The mixed-core REFLOD3B (BAW-10171P-A Rev. 03, Reference 9) analysis of the reflooding phase also conservatively simulated the resistance of a full core of Mark-B-HTP fuel to increase the flow losses and minimize the core reflooding rate.

Consistent with its normal practice for B&W plants, FANP performed the mixed-core and whole-core LOCA analyses by iterating on LHR to achieve a targeted PCT in the 2000 F +/- 50 F range. This targeted range contains margin to the 10 CFR 50.46 acceptance criterion of 2200 F. LOCA cases for a 177-FA LL B&W-designed plant with a PCT within the targeted range have substantial margins to the local oxidation and whole-core hydrogen generation criteria. It should be noted, however, that the targeted PCT range is a simple guide and not a strict requirement. Compliance with the 10 CFR 50.46 acceptance criteria remains the requirement.

REFERENCES

1. Progress Energy's earlier transmittal of explanation information to the NRC.

2. BAW-10187P-A, "Statistical Core Design for B&W-Designed 177FA Plants", March 1994.

3. BAW-10241P, "BHTP DNB Correlation Applied with LYNXT", December 2002.

4. BAW-10156P-A, Rev. 0, "LYNXT: Core Transient Thermal Hydraulic Program", February 1986.

5. BAW-10179P, Rev. 5, Safety Criteria and Methodology for Acceptable Cycle Reload Analyses", December 2002.

6. BAW-10133PA, Rev.1, Addendums 1 and 2, "Mark-C Fuel Assembly LOCA-Seismic Analyses," December 2000.

7. BAW-10192P-A Revision 0, "BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," June 1998.

8. BAW-10164P-A Revision 4, "An Advanced Computer Program for LWR LOCA and Non-LOCA Transient Analysis," November 2002.

9. BAW-10171P-A Revision 3, "REFLOD3B - Model for Multinode Core Reflooding Analysis," December 1995.