Progress Energy

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

Ref: 10 CFR 50.90

July 25, 2003
3F0703-07

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

Subject: Crystal River Unit 3 – Proposed License Amendment Request #276, Revision 1, "Use of M5 Advanced Alloy Fuel Cladding and Response to Request for Additional Information"

References:
1) PEF to NRC letter dated October 23, 2002, Crystal River Unit 3 – License Amendment Request #276, Revision 0, "Use of M5 Advanced Alloy Fuel Cladding"
2) NRC to PEF letter dated May 29, 2003, "Crystal River Unit 3 - Request for Additional Information Regarding Technical Specification Change Request on the Use of M5 Advanced Alloy Fuel Cladding" (TAC No. MB6590)

Dear Sir:

Progress Energy Florida, Inc. (PEF) submitted License Amendment Request (LAR) #276, Revision 0 (Reference 1) to allow use of an improved fuel design in the next operating cycle for Crystal River Unit 3. In Reference 2, the NRC staff issued a request for additional information (RAI) concerning this LAR. Attachment A provides the response to the RAI and a minor revision to LAR #276, made at the request of the NRC staff. The revision removes a statement that would have allowed the use of lead test fuel assemblies. The statement was included for consistency with NUREG-1430, Revision 2, Standard Technical Specifications Babcock and Wilcox Plants, but is not specifically needed for the next fuel cycle. This revision to LAR #276 does not impact the conclusions of the No Significant Hazards Consideration Determination or the Environmental Evaluation included in Reference 1.

No new regulatory commitments are made in this letter.

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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

Attachments:

A. Response to Request for Additional Information and Proposed Revised License Amendment Request
B. Proposed Revised Improved Technical Specification Page

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 25th 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 A

LICENSE AMENDMENT REQUEST #276, REVISION 1
Use of Advanced Alloy M5 Fuel Cladding

Response to Request for Additional Information and Proposed Revised License Amendment Request

NRC Request:

1) In section 4.0 of the licensee's submittal of October 23, 2002, the licensee indicates that:

> The cycle-specific reload report associated with Cycle 14 will include a plant-specific LOCA [Loss-of-Coolant Accident] reanalysis prior to the use of M5 alloy fuel assemblies at CR-3. This LOCA analysis will be done in accordance with ITS 5.6.2.18, "Core Operating Limits Report (COLR)" and BAW-10179P-A.

The NRC staff requests that the licensee identify the specific LOCA methodology (including topical reports with revisions numbers and dates) that will be used to perform these analyses. The applicability of the methodology to CR-3 and Cycle 14 conditions, including mixed core penalties, must be justified.

PEF Response:

Framatome ANP (FANP) performed Mark-B-HTP LOCA analyses for CR-3 using the NRC-approved B&W Nuclear Technology (BWNT) LOCA Evaluation Model (BAW-10192P-A Rev. 0, Reference 1) using the blowdown methods and models described in the NRC-approved RELAP5/MOD2-B&W code (BAW-10164P-A Rev. 04, Reference 2). The RELAP5/MOD2-B&W code references the NRC-approved methods for applications of M5 cladding (BAW-10227P-A, Rev. 0, Reference 3). The NRC-approved Evaluation Model (EM) blowdown 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 4) was therefore implemented into the RELAP5/MOD2-B&W code for analysis of the Mark-B-HTP fuel assemblies to support Cycle 14. The system reflooding phase of the LBLOCA analyses were completed using the NRC-approved REFLOD3B code (BAW-10171P-A Rev. 03, Reference 5). The refill and reflood cladding temperature response was completed with the NRC-approved BEACH code (BAW-10166P-A Rev. 4, Reference 6). It should be noted that additional information was provided to the NRC in Appendix H of Revision 5 to BAW-10166P (Reference 7) to request acceptance for amendment of the range of initial bottom of core recovery cladding temperatures. The approval was obtained in a safety evaluation report (SER) transmitted from Herbert Berkow, USNRC to James Mallay, FANP (Reference 8).

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 (cold leg pump discharge double-ended guillotine break with a discharge coefficient of 1.0) 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. Five BOL cases and five middle-of-life analyses, with axial peaks at the identified elevations, are performed along with one representative 2.506-ft analysis at the maximum fuel pin burnup. The LBLOCA analyses also include analysis of the 3, 6, and 8 weight percent gadolinia fuel pins.

The RELAP5/MOD2 blowdown mixed-core LBLOCA analyses that support Cycle 14 conservatively placed the Mark-B-HTP fuel with the higher form losses for the HTP grids in the

hot channel and simulated the average channel with the Mark-B10 lower resistance 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 analyses 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. The increase of flow diversion, flow losses and bypass flow conservatively reduces the fluid flow through the Mark-B-HTP assembly in the mixed-core configuration.

The mixed-core Mark-B-HTP limiting LBLOCA peak cladding temperature (PCT) was determined to be 2022.2 F (based on 16.8 kW/ft peak power at a burnup of 45 GWd/mtU with an axial peak at the 4.264 ft elevation). The whole-core Mark-B-HTP LBLOCA limiting PCT was calculated to be 2050.8 F (based on 17.0 kW/ft peak power at a burnup of 45 GWd/mtU with an axial peak at the 4.264 ft elevation). The mixed-core peaking penalties, which were determined by analysis, varied axially in the core from a minimum of 0.1 kW/ft for the core mid-plane and exit-skewed shapes to 0.4 kW/ft for the core inlet-skewed peaks. The maximum local oxidation was less than 4 percent for all cases and the whole-core hydrogen generation rate was less than 0.2 percent. The analyses performed to demonstrate compliance to the 10 CFR 50.46 acceptance criteria have substantial margins to PCT, local oxidation, and whole-core hydrogen generation criteria of 2200 F, 17 percent, and 1 percent, respectively.

Potentially limiting SBLOCA break sizes were also analyzed with the NRC-approved BWNT LOCA Evaluation Model (BAW-10192P-A Rev. 0, Reference 1) and NRC-approved RELAP5/MOD2-B&W code (BAW-10164P-A Rev. 04, Reference 2) using the void-dependent core cross flow model and the BHTP CHF correlation that is used for the fuel pin DNB analyses. These cases were analyzed in a mixed-core simulation using a 9.536-ft axial peak of 17.0 kW/ft to demonstrate 10 CFR 50.46 compliance for the Mark-B-HTP fuel. The mixed-core results will also be reported for the SBLOCA whole-core results. The mixed-core and whole core results are similar for SBLOCA transients because the quiescent core flow and lower core decay heat rate during the core uncovering phase of the transient do not result in substantial core flow diversion and changes in calculated PCT. The SBLOCA limiting PCT for the Mark-B-HTP fuel was 1248 F for the 0.07-ft^2 cold leg pump discharge break. The maximum local oxidation was less than 1 percent and the whole-core hydrogen generation rate was less than 0.1 percent for all SBLOCA cases.

The BWNT LOCA EM and associated code and method topicals have been approved for LOCA analysis of B&W 177 fuel assembly lowered-loop and raised-loop plant designs, as well as B&W 205 raised-loop plant types. Since CR-3 is a B&W 177 lowered-loop plant design, the CR-3 plant-specific Mark-B-HTP LOCA analyses are acceptable for application to the CR-3 10 CFR 50.46 licensing basis. Reference to the application of the BWNT LOCA EM for M5 cladding is made through Appendices N and U of BAW-10179P, Revision 5 (Reference 9), with application of the DNB correlation (Reference 4) submitted in Appendix V of BAW-10179P, Revision 5. The CR-3 plant-specific LOCA analyses were completed with a mixed-core configuration input that conservatively represents Cycle 14 conditions. Analyses were also completed with a whole-core configuration of Mark-B-HTP fuel for use in future cycles.

References

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

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

3. BAW-10227P-A Revision 0, "Evaluation of Advanced Cladding and Structural Material in PWR Reactor Fuel," February 2000.

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

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

6. BAW-10166P-A Revision 4, "BEACH – Best Estimate Analysis Core Heat Transfer – A Computer Program for Reflood Heat Transfer During LOCA," February 1996.

7. BAW-10166P Revision 5, "BEACH – Best Estimate Analysis Core Heat Transfer – A Computer Program for Reflood Heat Transfer During LOCA," December 2001.

8. Letter Herbert N. Berkow (USNRC) to James F. Mallay (FANP), "Acceptance for Referencing of Appendices H and I to BAW-10166P-A, "BEACH – Best Estimate Analysis Core Heat Transfer, A Computer Program for Reflood Heat Transfer During LOCA," TAC No. MB7549, July, 2003.

9. BAW-10179P Revision 5, "Safety Criteria and Methodology for Acceptable Core Reload Analysis," December 2002.

NRC Request:

2) In section 2.0 of the submittal of October 23, 2002, the licensee requests the removal of some fuel design features (maximum fuel enrichment, nominal active fuel length, weight of uranium for fuel rods, and details of Control Rod content) from the CR-3 Technical Specifications (TS) and substitutes alternative language.

   The NRC staff requests that the licensee justify how the substituted language will ensure that only those fuel designs that have been analyzed with NRC-approved codes and methods applicable to CR-3 will be used in all future core reloads.

PEF Response:

The substituted language was chosen because it is consistent with NUREG 1430, Revision 2, Standard Technical Specifications Babcock and Wilcox Plants. The parameters that are being removed from ITS (maximum fuel enrichment, nominal active fuel length, weight of uranium for fuel rods, and details of Control Rod content) do not meet the criteria of 10 CFR 50.36 (c)(4),

Design features, for inclusion in Improved Technical Specifications (ITS). Therefore, these parameters were removed from ITS in NUREG 1430, Revision 0. These parameters are engineering design information and do not provide assurance that core safety limits are met.

The core safety limits are assured by the parameters and requirements of ITS 5.6.2.18, Core Operating Limits Report (COLR) and the Limiting Conditions for Operation, Safety Limits and Surveillance Requirements listed in this specification. Revisions to the COLR are submitted to the NRC per ITS 5.6.2.18. The COLR ensures that all core designs are analyzed with NRC-approved codes and methods applicable to CR-3. These codes and methods include inputs which account for the specific parameters that are being removed from ITS by this LAR. The value for parameters removed from ITS 4.2 will be maintained in the FSAR or other design documents controlled under the 10 CFR 50.59 process and the 10 CFR 50 Appendix B, Criterion III, Design Control, program.

<u>NRC Request:</u>

3) In a teleconference on January 16, 2003, the NRC staff questioned the basis of the licensee's request for changes to the TS pertaining to lead test assemblies (LTA) in the October 23, 2002 submittal, since no LTA will be used in Cycle 14.

   The NRC staff understands that the licensee will not use LTA in Cycle 14, and requests that this change request be withdrawn or the need for its inclusion be justified.

<u>PEF Response:</u>

The option to permit lead test assemblies was included in order to be consistent with NUREG 1430, Revision 2. CR-3 has no immediate need for lead test assemblies and, therefore, this portion of the request is being withdrawn. The revised ITS page is included in Attachment B. The last sentence of the revised ITS 4.2.1, proposed in License Amendment Request (LAR) #276, Revision 0, shown below, is being deleted.

> "A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions."

In addition, an editorial change is being requested to insert the word "ROD" in the term AXIAL POWER SHAPING ROD (APSR) assemblies in ITS 4.2.2.

The remaining changes requested in LAR #276, Revision 0, are not affected by this revision to the LAR.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #276, REVISION 1
Use of Advanced Alloy M5 Fuel Cladding

Proposed Revised Improved Technical Specification Page

4.0 DESIGN FEATURES

4.1 Site

The 4,738 acre site is characterized by a 4,400 foot minimum exclusion radius centered on the Reactor Building; isolation from nearby population centers; sound foundation for structures; an abundant supply of cooling water; an ample supply of emergency power; and favorable conditions of hydrology, geology, seismology, and meteorology.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4 or M5 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases.

4.2.2 Control Rods

The reactor core shall contain 60 safety and regulating CONTROL ROD assemblies and 8 AXIAL POWER SHAPING ROD (APSR) assemblies. The material shall be silver indium cadmium or Inconel as approved by the NRC.

(continued)