Florida Power
A **Progress Energy** Company

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

Ref: 10 CFR 50.90

December 19, 2002
3F1202-04

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

Subject: Crystal River Unit 3 - License Amendment Request #277, Revision 0, "BHTP Departure From Nucleate Boiling Correlation"

Reference: Crystal River Unit 3 - License Amendment Request #276, Revision 0, "Use of M5 Advanced Alloy Fuel Cladding," dated October 23, 2002

Dear Sir:

Pursuant to Title 10 of the Code of Federal Regulations, Section 50.90 (10 CFR 50.90), Florida Power Corporation (FPC) submits a request to amend the Crystal River Unit 3 (CR-3) Operating License, Appendix A, Improved Technical Specifications (ITS). The proposed changes involve ITS 2.1.1, Reactor Core Safety Limits (SLs). Current ITS 2.1.1 permits the use of the BAW-2 and BWC correlations. The proposed change will permit the use of the BHTP correlation which is needed to utilize the Framatome ANP (FRA-ANP) high thermal performance (HTP) spacer grid design. The BHTP correlation was submitted for NRC review by FRA-ANP in BAW-10241(P), "BHTP DNB Correlation Applied With LYNXT," dated December 18, 2002. FRA-ANP also requested approval of BAW-10179P, "Safety Criteria and Methodology for Acceptable Cycle Reload Analysis," Revision 5, to include use of the BHTP Correlation on December 18, 2002.

In addition to the high thermal performance spacer grid design, the CR-3 Cycle 14 fuel design will utilize M5 advanced alloy material for fuel cladding and other fuel assembly structural components. Therefore, adoption of the proposed fuel design also requires approval of License Amendment Request #276 (referenced above) concerning the use of M5 advanced alloy material. The Cycle 14 fuel utilizing the HTP fuel design and the M5 cladding is referred to as Mark-B/HTP fuel.

CR-3 is planning to utilize Mark-B/HTP replacement fuel assemblies beginning in the next refueling scheduled for October 2003. CR-3 respectfully requests that this amendment, the amendment and exemptions required for the M5 cladding, BAW-10241(P) and BAW-10179, Revision 5, be issued by August 1, 2003 to support this effort.

The CR-3 Plant Nuclear Safety Committee has reviewed this request and recommended it for approval.

A001

15760 West Power Line Street • Crystal River, Florida 34428-6708 • (352) 795-6486

This letter makes no new regulatory commitments.

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. Assessment of Proposed Changes (Introduction, Description, Background, Technical Analysis, Regulatory Analysis, References and Precedents)
B. Proposed Revised Improved Technical Specification Pages – Strikeout Version
C. Proposed Revised Improved Technical Specification Pages – Revision Line 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; 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 19th day of December, 2002, 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 ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #277, REVISION 0
BHTP Departure From Nucleate Boiling Correlation

Assessment of Proposed Changes
(Introduction, Description, Background, Technical Analysis, Regulatory Analysis, References and Precedents)

Description and Assessment of Proposed Changes

1.0 INTRODUCTION

This letter submits a License Amendment Request (LAR) to revise Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS) 2.1.1, Safety Limits (SLs). CR-3 is planning to use an enhanced Framatome ANP (FRA-ANP) fuel design for the replacement fuel assemblies beginning in Cycle 14. The enhanced fuel design utilizes a high thermal performance (HTP) spacer grid which requires the use of the BHTP correlation for departure from nucleate boiling (DNB) analyses.

2.0 DESCRIPTION

The proposed ITS change will add a safety limit value for use of the BHTP correlation for DNB analyses. The proposed wording for ITS 2.1.1.2 is as follows:

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained greater than the limits of 1.3 for the BAW-2 correlation, 1.18 for the BWC correlation and 1.132 for the BHTP correlation. Operation within this limit is ensured by compliance with SL 2.1.1.3 and with the AXIAL POWER IMBALANCE protective limits preserved by the RPS setpoints in LCO 3.3.1, as specified in the COLR.

3.0 BACKGROUND

HTP correlations have been approved for use in DNB analyses for Westinghouse fuel since the early 1990s. NRC approval for the HTP correlation was extended to Combustion Engineering and other fuel designs in EMF-92-153(P)(A), "HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel," Safety Evaluation dated December 28, 1993. The HTP DNB correlation was used in conjunction with the XCOBRA-IIIC computer code to compare DNB predicted to measured heat fluxes (Reference 2).

FRA-ANP developed the methodology for the BHTP correlation (use of the HTP correlation with the LYNXT computer code) and submitted the topical report BAW-10241(P), "BHTP DNB Correlation Applied With LYNXT," to the NRC for review and approval on December 18, 2002 (Reference 3). Framatome revised the topical report BAW-10179P, "Safety Criteria and Methodology for Acceptable Cycle Reload Analysis," Revision 5 (Reference 5), to include the BHTP Correlation (Attachment V), and submitted this to the NRC for review and approval on December 18, 2002. The LYNXT computer code was approved in BAW-10156-A, Revision 1, "LYNXT – Core Transient Thermal-Hydraulic Program," (Reference 4).

CR-3 intends to use the BHTP correlation for Mark-B/HTP fuel, which utilizes the Siemens HTP spacer grid design combined with the FRA-ANP Mark-B series fuel design and analysis methods. BAW-10241(P) essentially combines two approved methods of DNB analysis (HTP DNB correlation and LYNXT) to accommodate the combination of fuel design features. The proposed BHTP Correlation must be reviewed and approved by the NRC prior to the approval for this LAR and implementation at CR-3.

4.0 TECHNICAL ANALYSIS

The HTP correlation (Reference 1) has been used with the XCOBRA-IIIC thermal-hydraulic computer code (Reference 2) for the reload analyses of the HTP fuel designs. The incorporation of the HTP spacer grid into the Mark-B fuel design series reflects the integration of a Siemens developed spacer grid design into a Framatome developed fuel assembly design.

BAW-10241(P) provides the technical justification for using the HTP correlation with the LYNXT thermal-hydraulic code (Reference 3). The HTP data base has been evaluated using the LYNXT code and a 95/95 critical heat flux (CHF) design limit of 1.132 has been established. Although the original HTP CHF correlation form has been retained, the re-correlation has yielded changes to some of the coefficients that reflect the use of LYNXT. Since some coefficients have changed, the correlation has been given the distinct name of BHTP CHF Correlation.

No changes are being proposed to the existing pressure-temperature Departure from Nucleate Boiling (DNB) Safety Limits in ITS Figure 2.1.1-1. The existing limits are based on a previous deterministic methodology, not on the Statistical Core Design (SCD) methods. Recalculation of these limits using SCD shows that the original methodology contains significant conservatism. The existing limits remain bounding within the range of acceptable operation as defined by the RCS reactor protection system high pressure, low pressure, and high outlet temperature trips and are applicable for the establishment of the Variable Low Pressure Trip (VLPT). Florida Power Corporation (FPC) has no need to reduce this conservatism and, therefore, the ITS safety limit curve is not being changed.

5.0 REGULATORY ANALYSIS

The proposed ITS change is based on NRC approval of BAW-10241(P) and BAW-10179P, Revision 5, submittals by FRA-ANP on December 18, 2002.

<u>5.1 No Significant Hazards Consideration Determination</u>

FPC has evaluated the proposed License Amendment Request (LAR), which consists of the identified Improved Technical Specification (ITS) change, against the criteria of 10 CFR 50.92(c). The ITS change allows the use of the BHTP Correlation for departure from nucleate boiling (DNB) calculations of reload cores containing the Mark-B/HTP fuel design.

FPC has concluded that this proposed LAR does not involve a significant hazards consideration.

The following is a discussion of how each of the criteria is satisfied.

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed safety limit value ensures that fuel integrity will be maintained during normal operations and anticipated operational occurrences (AOOs), and that the design requirements will continue to be met. The proposed methodology for the BHTP departure from nucleate boiling (DNB) correlation will be generically reviewed and approved by the NRC prior to its use by Crystal River Unit 3 (CR-3) in mixed core reload analyses. The core operating limits will be developed in accordance with the new methodology and any limitations established by the NRC in its safety evaluation of the new methodology. The proposed safety limit value does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously evaluated. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. Therefore, the safety limit value for the BHTP correlation will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed safety limit value does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The BHTP correlation is not an accident/event initiator. No new initiating events or transients result from the use of the BHTP correlation and the related safety limit changes. Therefore, the safety limit value for the BHTP correlation will not involve the possibility of a new or different kind of accident from any previously evaluated.

(3) Involve a significant reduction in a margin of safety.

The proposed safety limit value has been established in accordance with the methodology for the BHTP correlation, to ensure that the applicable margin of safety is maintained (i.e., there is at least 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience departure from nucleate boiling (DNB)). The proposed methodology for the BHTP DNB correlation will be generically reviewed and approved by the NRC prior to its use by CR-3. The other reactor core safety limits will continue to be met by analyzing the reload for the mixed core using NRC approved methods, and incorporation of resultant operating limits into the Core Operating Limits Report (COLR). Therefore, the safety limit value for the BHTP correlation will not involve a significant reduction in a margin of safety.

5.2 Regulatory Safety Analysis

BAW-10241(P), demonstrates that the BHTP correlation is suitable for use in calculating the DNBR for Framatome Mark-B/HTP fuel. The core reload analysis will be done using BAW-10179P, Revision 5, which will incorporate the BHTP DNB correlation.

6.0 ENVIRONMENTAL EVALUATION

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

Florida Power Corporation (FPC) has reviewed this license amendment request and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

The proposed license amendment does not involve a significant hazards consideration as described previously in the no significant hazards evaluation for this License Amendment Request (LAR).

Offsite release concentrations and doses will continue to be maintained within the limits of 10 CFR 20 and 10 CFR 50, Appendix I, in accordance with the requirements of the CR-3 Offsite Dose Calculation Manual (ODCM). The ODCM contains offsite dose calculation methodologies, the radioactive effluent controls program, and radiological environmental monitoring activities. The ODCM contains the methodologies and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, the methodologies and parameters used in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and the controls for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable in accordance with 10 CFR 50.36a. The proposed changes will not result in changes in the operation or design of the gaseous, liquid or solid waste systems, and will not create any new or different radiological release pathways.

Therefore, the proposed license amendment will not result in a significant change in the types or increase in the amounts of any effluents that may be released off-site.

The proposed changes will not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 10 CFR 20. Radiation levels in the plant will not be changed due to the new DNBR safety limit value. Individual worker exposures will be maintained within acceptable limits by the CR-3 as-low-as-reasonably-achievable (ALARA) program. Therefore, the proposed license amendment will not result in a significant increase to the individual or cumulative occupational radiation exposure.

Non-Radiological Evaluation

With regard to non-radiological impacts, the proposed license amendment involves no significant increase in the amounts or changes in the types of any non-radiological effluents that may be released offsite.

7.0 REFERENCES

1. EMF-92-153(P)(A) and EMF-92-153(P)(A) Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," March 1994.

2. XN-NF-75-21(P)(A), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Operation," Exxon Nuclear Company, January 1986.

3. BAW-10241(P), Revision 0, "BHTP DNB Correlation Applied with LYNXT," December 2002.

4. BAW-10156-A, Revision 1, "LYNXT – Core Transient Thermal-Hydraulic Program," Babcock & Wilcox, Lynchburg, Virginia, August 1993.

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

8.0 PRECEDENTS

The use of the HTP correlation for DNB calculations has been approved for other fuel designs in EMF-92-153(P)(A), "HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel," Safety Evaluation dated December 28, 1993. The HTP DNB correlation was developed with critical heat flux test local conditions predicted by the XCOBRA-IIIC computer code. The HTP correlation is in use at a number of other plants including Progress Energy's H. B. Robinson and Shearon Harris Plants. The BHTP correlation utilizes the same critical heat flux database and correlation functional form as the HTP correlation, but uses the NRC approved LYNXT computer code (Reference 4).

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #277, REVISION 0
BHTP Departure From Nucleate Boiling Correlation

Proposed Revised Improved Technical Specification Pages
Strikeout Version

~~Strikeout text~~	Indicates deleted text
Shadowed text	Indicates added text

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5 E-3) X (Burnup MWD/MTU)°F. Operation within this limit is ensured by compliance with the AXIAL POWER IMBALANCE protective limits preserved by the Reactor Protection System setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained greater than the limits of 1.3 for the BAW-2 correlation, ~~and~~ 1.18 for the BWC correlation **and 1.132 for the BHTP correlation.** Operation within this limit is ensured by compliance with SL 2.1.1.3 and with the AXIAL POWER IMBALANCE protective limits preserved by the RPS setpoints in LCO 3.3.1, as specified in the COLR.

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the SL shown in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

The following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1, SL 2.1.1.2 or SL 2.1.1.3 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT C

LICENSE AMENDMENT REQUEST #277, REVISION 0
BHTP Departure From Nucleate Boiling Correlation

Proposed Revised Improved Technical Specifications Pages

Revision Bar Format

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5 E-3) X (Burnup MWD/MTU)°F. Operation within this limit is ensured by compliance with the AXIAL POWER IMBALANCE protective limits preserved by the Reactor Protection System setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained greater than the limits of 1.3 for the BAW-2 correlation, 1.18 for the BWC correlation and 1.132 for the BHTP correlation. Operation within this limit is ensured by compliance with SL 2.1.1.3 and with the AXIAL POWER IMBALANCE protective limits preserved by the RPS setpoints in LCO 3.3.1, as specified in the COLR.

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the SL shown in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

The following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1, SL 2.1.1.2 or SL 2.1.1.3 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.