Florida Power
CORPORATION

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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Technical Specification Change Request No. 152
Response to Questions Concerning Cycle 7 Reload

Dear Sir:

Enclosed please find a response to the questions in your June 19, 1987 letter. We were somewhat concerned that your request appears to contain new applicable staff positions concerning the content of Technical Specification Bases and application of SRP 15.4.6. Nevertheless we have provided the requested information and made appropriate changes.

If you would like a meeting or teleconference to discuss this further, please do not hesitate to contact us.

Sincerely,

Ken Wilson for

E.C. Simpson, Director
Nuclear Operations Site Support

PGH/dhd

xc: Dr. J. Nelson Grace
Regional Administrator, Region II

Mr. T.F. Stetka
Senior Resident Inspector

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RESPONSE TO QUESTIONS CONCERNING CRYSTAL RIVER UNIT 3 CYCLE 7 RELOAD

1. Request: Justify that the increased bypass flow of 8.8% used in the thermal-hydraulic design evaluation for the full Mark BZ core provides sufficient margin to offset the core penalty required for the Cycle 7 mixed core of Mark BZ and Mark B fuel assemblies.

1. Response: Cycle 7 is the first Mark BZ transition cycle for Crystal River 3 and has a feed batch of 80 Mark BZ fuel assemblies with a 7.6% core bypass flow. The worst case minimum DNBR is modeled using the LYNXT computer code (Reference 1). A single-pass, thermal-hydraulic model is used in LYNXT so only one type of fuel assembly is represented. The Mark BZ assembly with an 8.8% bypass flow resulted in a worst case minimum DNBR of 1.59 for the one pump coastdown. This is significantly above the limiting minimum DNBR of 1.18, so a core transition penalty was not necessary.

To assure that the Mark BZ assembly with an 8.8% bypass flow bounds the Cycle 7 mixed core (7.6% bypass flow), additional analyses were performed. Using LYNX1 and LYNX2 (References 2 and 3), a mixed core containing both Mark B and Mark BZ fuel assemblies with a core bypass flow fraction of 7.8% and a homogenous Mark BZ core with a bypass flow fraction of 8.8% were modeled. The calculated minimum DNBR for the full Mark BZ core with the 8.8% bypass flow fraction was 1.81 (BWC). The mixed core with the 7.8% bypass flow fraction resulted in a 1.88 (BWC) calculated minimum DNBR. Thus the full Mark BZ core with an 8.8% bypass flow conservatively models the mixed core of Cycle 7.

2. Request: The license has proposed to eliminate any reference to the DNB correlation used or the limiting DNBR value in the Bases for Tech Spec 2.2.1 and 2.1.2. Although the staff has previously accepted the removal of the DNBR limiting value, we require that the particular correlation(s) used for the most recent core continue to be referenced in the Bases.

2. Response: Attached are replacement pages for the Bases for Tech Spec 2.1.1 and 2.1.2 which specify the correlations used for Cycle 7.

3. Request: Provide details on the reevaluation of the boron dilution event indicating why a reevaluation was required and justifying that the resulting time available for the operator to take corrective action meets the requirements of SRP 15.4.6.

3. Response: The boron dilution event of FSAR Section 14.1.2.4.1, "Terminated Dilution Through the Makeup and Purification System", was reanalyzed because the longer cycle, 24 months, changed core kinetics parameters and BOL boron concentration. Attached please find Table 14-10 and 14-14 describing the new accident parameters and analysis results. Operator action to terminate the analyzed events is not required, so SRP 15.4.6 criteria is satisfied.

The boron dilution event of FSAR Section 14.1.2.4.2, "Unterminated Dilution Through the Decay Heat Removal System", was not reanalyzed for Cycle 7. A modification to the Building Spray Additive system, NaOH tank, was installed that makes this event no longer credible. Because the event is incredible, it was not considered for reanalysis. This event will be deleted from the FSAR after the development of the Analysis Basis Document and FSAR Section 14 rewrite.

4. Request: The Cycle 7 control rods will differ from those used in previous cycles. Provide a reference where the use of this type of control rod has been previously approved by the NRC. Since no mention of these new rods is made in the Cycle 7 reload report, verify that the Cycle 7 values of rod worths, shutdown margin, etc., reflect the use of these new control rods.

4. Response: Eight extended life control rod assemblies will be located in core positions C7 and C9 and their symmetric counterparts of the Cycle 7 core. This type of control rod assembly was placed in service at Arkansas Nuclear One, Unit 1 in Cycle 7 (Reference 4). The Crystal River Unit 3 Cycle 7 values of rod worths, shutdown margin, etc., reflect to use of the extended life control rods.

5. Request: Provide references for the NRC's previous acceptance of the revised power imbalance detector correlation to justify the reduction from 3.5% to 2.5%.

5. Response: The purpose of the New Power Imbalance Detector Correlation is to avoid unnecessary reactor trips caused by over conservatism in the power imbalance measured by the excore detectors. In the past, the excore detectors have been conservatively calibrated to the more accurate incore detectors by requiring the excore/incore offset slope to be > 1.35, with a recalibration criterion of ± 3.5% offset. By requiring more restrictive recalibration and slope correlation test criteria, we have provided for a more accurate excore indication of imbalance, and hence avoidance of unnecessary reactor trips.

The new power imbalance detector correlation has been implemented at Arkansas Nuclear One, Unit 1 and Three Mile Island, Unit 1. Because these units do not have the minimum allowable recalibration criterion in their Tech Specs as does Crystal River Unit 3, the new correlation is only reflected in Section 9, Physics Testing of their reload reports. However, the change in the recalibration criterion from 3.5% to 2.5% should have been reflected in their plant procedures. Use of the new correlation requires that the excore/incore offset slope be greater than .96 (with a target of 1.00) vice the previous 1.15 limit, per Section 9.3.2 of the reload report. References 5 and 6 are the NRC SER's which accept the proposed Physics testing requirements for the cycles implementing the new correlation.

The more restrictive criteria for excore to incore calibration in the Tech Specs, and more restrictive criterion in the PIDC Physics Testing acceptance criteria preserve and required measurement system error allowance, and combine to provide a more accurate excore measurement. This avoids unnecessary trips due to overconservative excore imbalance, as could happen with the 1.15 slope.

REFERENCES:

1. BAW-10156-A, LYNXT: Core Transient Thermal Hydraulic Program, Babcock and Wilcox, Lynchburg, Virginia, February 1986.

2. BAW-10129-A, LYNX1: Reactor Fuel Assembly Thermal Hydraulic Analysis code, Babcock and Wilcox, Lynchburg, Virginia, July 1985.

3. BAW-10130-A, LYNX2: Subchannel Thermal Hydraulic Analysis Code, Babcock and Wilcox, Lynchburg, Virginia, July 1985.

4. BAW-1840, Arkansas Nuclear One, Unit 1, Cycle 7 Reload Report, Babcock and Wilcox, Lynchburg, Virginia, August, 1984.

5. Amendment No. 105: Arkansas Nuclear One, Unit 1, Cycle 8 Tech Specs, Guy S. Vissing to Mr. Gene Campbell, November 24, 1986

6. Amendment No. 126: Three Mile Island Nuclear Station, Unit 1, Cycle 6 Tech Specs, John O. Thoma to Mr. Henry D. Hukill, March 20, 1987.