Florida Power
CORPORATION

March 22, 1989
3F0389-18

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

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Technical Specification Change Request No. 166
Core Operating Limits Report

Dear Sir:

Florida Power Corporation (FPC) hereby submits Technical Specification Change Request No. (TSCRN) 166, requesting amendment to Appendix A of Operating License No. DPR-72. Proposed replacement pages for Appendix A and associated bases are provided.

This submittal requests that cycle-dependent core operating limits be removed from Technical Specifications (TS) and relocated to a Core Operating Limits Report. This report will document the specific values of parameter limits resulting from the licensee's calculations including any mid-cycle revisions to such parameter values.

Generic letter 88-16, which was issued by the NRC on October 4, 1988, provides a means for removing the numeric values of cycle specific parameters from the technical specifications. The generic letter requires that the cycle specific limits be determined using an NRC-approved methodology.

B&W Fuel Company (BWFC) calculates the axial power imbalance ranges, control rod insertion ranges, axial power shaping rod insertion ranges, and quadrant power tilt limits with the methodology described in topical report BAW-10122A, Rev. 1, "Normal Operating Controls." This report received NRC approval in Safety Evaluation Reports (SERs) dated September 14, 1979 and April 20, 1984. The SERs conclude that the procedures and techniques described in BAW-10122A, Rev. 1 are acceptable for establishing limiting conditions for operation for the parameters discussed above.

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BWFC calculates the moderator temperature coefficient with the methodology described in topical reports BAW-10118A, "Core Calculational Techniques and Procedures" and BAW-10152A, "NOODLE, A Multi-Dimensional Two-Group Reactor Simulator." Topical report BAW-10118A received NRC approval in the Safety Evaluation Report dated September 25, 1979. Topical report BAW-10152A received NRC approval in the SER dated April 24, 1985. The SERs conclude that the procedures and techniques described in BAW-10118A and BAW-10152A are acceptable for establishing limiting conditions for operation for the moderator temperature coefficient.

The methodology for calculating the programmed rod positions is described in BWFC topical report, BAW-10118A, "Core Calculational Techniques and Procedures", and in the attached methodology (attachment 1) provided by BWFC. Topical report BAW-10118A received NRC approval as noted above. The refuel boron concentration limits are calculated by BWFC using the methodology described in the attached BWFC report (attachment 1). It is requested that these methodologies be approved with this change request as they will be utilized in determining the parameters to be used in the Core Operating Limits Report for Rod Program and Refuel Boron Concentration.

The NRC issued a Safety Evaluation Report on January 26, 1989 to Duke Power Company for their Core Operating Limits Report change request. Oconee Nuclear Station was the lead plant for the Babcock and Wilcox owners group regarding this submittal.

FPC requests this amendment become effective 30 days after issuance in order to allow for procedure changes and training.

Very truly yours,

Gary L. Boldt

Gary L. Boldt, Vice President
Nuclear Production

GLB:dlh
Attachment

xc: Regional Administrator, Region II

Senior Resident Inspector

STATE OF FLORIDA

COUNTY OF CITRUS

G.L. Boldt states that he is the Vice President, Nuclear Production for Florida Power Corporation; 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.

Gary L. Boldt

G.L. Boldt, Vice President
Nuclear Production

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 22nd day of March, 1989.

Joan Lisa Bute

Notary Public

Notary Public, State of Florida at Large
My Commission Expires: June 21, 1991

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER	)	
	)	DOCKET NO. 50-302
FLORIDA POWER CORPORATION	)	

CERTIFICATE OF SERVICE

G.L. Boldt deposes and says that the following has been served on the Designated State Representative and Chief Executive of Citrus County, Florida, by deposit in the United States mail, addressed as follows:

Chairman,
Board of County Commissioners
of Citrus County
Citrus County Courthouse
Inverness, FL 32650

Administrator
Radiological Health Services
Department of Health and
Rehabilitative Services
1323 Winewood Blvd.
Tallahassee, FL 32301

A copy of Technical Specification Change Request No. 166, Revision 0, requesting Amendment to Appendix A of Operating Licensing No. DPR-72.

FLORIDA POWER CORPORATION

Gary L. Boldt
G.L. Boldt, Vice President
Nuclear Production

SWORN TO AND SUBSCRIBED BEFORE ME THIS 22nd DAY OF March, 1989.

Joan Lisa Buffé
Notary Public

Notary Public, State of Florida at Large
My Commission Expires: June 21, 1991

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72
REQUEST NO. 166
CORE OPERATING LIMITS REPORT

LICENSE DOCUMENT INVOLVED: Technical Specifications

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DESCRIPTION OF REQUEST:

This submittal requests that cycle-dependent core operating limits be removed from Technical Specifications (TS) and relocated to a Core Operating Limits Report. The term, CORE OPERATING LIMITS REPORT, will be added to the Definitions Section of TS. A new administrative reporting requirement will be added to existing reporting requirements of TS. Individual specifications and applicable bases will be revised to state that the values of cycle-specific parameters shall be maintained within the limits identified in Core Operating Limits Report.

REASON FOR REQUEST:

A number of Technical Specifications (TS) address limits associated with reactor physics parameters that generally change with each reload requiring the processing of changes to TS to update these limits each fuel cycle. These limits are developed using an NRC-approved methodology, therefore, the license amendment process is an unnecessary burden on the licensee and the NRC. An alternative to including the values of these cycle-specific parameters in individual specifications is responsive to industry and NRC efforts on improvements in TS and is provided in this change request.

EVALUATION OF REQUEST:

It is essential to safety that the plant is operated within the bounds of cycle-specific parameter limits and that a requirement to maintain the plant within the appropriate bounds must be retained in the TS. However, the specific values of these limits may be modified by licensees, without affecting nuclear safety, provided that these changes are determined using an NRC-approved methodology and consistent with all applicable limits of the plant safety analysis that are addressed in the Final Safety Analysis Report (FSAR). A Core Operating Limits Report will be submitted to NRC with the values of these limits. This will allow continued trending of this information, even though prior NRC approval of the changes to these limits would not be required. The Core Operating Limits Report will document the specific values of parameter limits resulting from the licensee's calculations including any mid-cycle revisions to such parameter values. The methodology for determining cycle-specific parameter limits (except for Rod Program and Refuel Boron Concentration) is documented in NRC-approved Topical Reports BAW-10122A, Rev. 1, BAW-10118A, Rev. 0, and BAW 10152A, Rev. 0. As a consequence, the NRC review of proposed changes to TS for these limits is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology and consistent with all applicable limits of the safety analysis. Attachment 1 describes the methodology for determining parameter limits for Rod Program and Refuel Boron Concentration. It is requested that these methodologies be approved with this change request as they will be utilized in determining the parameters to be used in the Core Operating Limits Report for Rod Program and Refuel Boron Concentration.

The cycle specific core operating limits calculated in accordance with the approved methodologies will be included in the report, Core Operating Limits Report, and provided to the NRC upon issuance as required by the proposed Technical Specification 6.9.1.7. Controlled copies of the Core Operating Limits Report will be maintained at Crystal River Unit-3 (CR-3) and will be revised as required by the future CR-3 cycles. Attachment 2 provides a sample copy of a Core Operating Limits Report for information purposes only.

SHOLLY EVALUATION OF REQUEST:

Florida Power Corporation (FPC) proposes the removal of cycle specific core operating limits from Technical Specifications does not involve a significant hazard consideration. The removal of cycle dependent variables from Technical Specifications has no impact upon plant operation or safety. The Technical Specifications will continue to require operation within the core operational limits for each cycle reload calculated by the approved reload design methodologies. Appropriate actions to be taken if limits are violated will also remain in Technical Specifications.

FPC concludes this change will not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated because the removal of cycle specific core operating limits from the Crystal River Unit 3 (CR-3) Technical Specifications has no influence or impact on the probability of a Design Basis Accident (DBA) occurrence.

   The cycle specific core operating limits will be relocated to a CORE OPERATING LIMITS REPORT. The requirements to operate CR-3 within the limits will continue to be maintained in Technical Specifications.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because the removal of the cycle specific variables has no influence, impact, nor does it contribute to the probability or consequences of an accident. The cycle specific variables are calculated using NRC approved methodologies, and Technical Specifications will continue to require the operation within the core operating limits.

3. Involve a significant reduction in the margin of safety because the margin of safety presently provided by current Technical Specifications remains unchanged. This proposed amendment still requires operation within the core limits as obtained from the NRC approved methodologies, and appropriate actions to be taken when, or if limits are violated, remain unchanged.

ATTACHMENT 1

METHODOLOGY

- ROD PROGRAM
- REFUEL BORON CONCENTRATION

METHODOLOGY

ROD PROGRAM

The Core Operating Limits Report has a figure which provides the Rod Program for each cycle. The Rod Program Figure shows the location of each control assembly in the core and identifies to which rod group assembly is assigned. The Technical Specification states, "Each control rod assembly (safety, regulating and APSR) shall be programmed to operate in the core location and rod group specified in the CORE OPERATING LIMITS REPORT." The following discussion describes the procedures and methods of analysis used to determine the rods assigned to each group.

Each control rod assembly in the core can be electrically connected to any of the eight control rod groups. The only limitation is that each group must have between 4 and 12 rod assemblies. The groups are numbered 1 through 8 and are divided into these categories. Groups 1 through 4 are the safety rods, Groups 5 through 7 are the regulating rods, and Group 8 is the axial power shaping rods (APSR's). The following description of the methods and procedures for the analysis addresses the APSR's first and then the regulating rods and finally the safety rods.

The APSR's are unique in their physical characteristics from the other control rods. They have a shorter poison region and the poison may be a different composition. The latching mechanism for the APSR's is also different from the other control rods in that these rods cannot be unlatched when the control rods are scrammed. The location of the APSR's in the core is symmetric by quadrant. These locations (L12 and N10 in the lower right quadrant) were determined to have a minimal impact on radial power peaking while achieving the greatest amount of overall control on core offset (or imbalance). The locations of the APSR's have not been changed since the first cycle and are not expected to be changed.

The regulating rods, Groups 5 through 7, are used to control the core power level. These rod groups are electrically coupled to be sequentially withdrawn (5,6,7), with an overlap of 25% $\pm$5% and sequentially inserted (7,6,5). The location of the rods in each of the regulating groups is determined beginning with Group 7. Group 7 is usually composed of 8 rod assemblies. The location of the Group 7 rod is selected to be symmetrical in the eighth core and have a minimal impact on the radial power distribution. The locations are also restricted to positions other than those adjacent to the Group 8 rods, if possible. The Groups 6 and 5 rods are also selected to have eighth core symmetry and have a minimal impact on the radial power distribution. In addition, the Groups 6 and 5 rods are positioned to ensure an ejected rod will not violate the safety criteria. (See BAW-10118A, "Core Calculational Techniques And Procedures," J.J. Romano, December, 1979, for a discussion of ejected rod worth analysis.)

Design analyses with Groups 5 through 8 determine the limiting rod positions for the operation of each cycle. These analyses evaluate the operation of the core at various power levels throughout the cycle and show how normal operating controls on the rod groups can be set to ensure safe operation. (See BAW-10122A, Rev. 1, "Normal Operating Controls," G.E. Hanson, April, 1984, for a discussion of rod operation analyses which ensure operating margin with respect to power peaking, shutdown reactivity and ejected rod worths).

The safety rods, Groups 1 through 4, are used to ensure sufficient scram reactivity for a safe shutdown of the reactor core. These rod groups are fully withdrawn prior to the core going critical. They are withdrawn one group at a time but not necessarily in order. The core remains shutdown with Groups 1 through 4 out of the core, therefore there is no peaking requirement on the selection of the rod groups for Groups 1 through 4. However, there are two conditions which influence the location of the Group 4 rods and the Group 1 rods.

The Group 4 rods may be briefly inserted into the core while the core is in the Startup Mode (2) during physics testing. Thus the location of the Group 4 rod assemblies is chosen such that they are eighth core symmetric. They are also positioned with respect to the location of Groups 5 through 7 to have a worth of approximately 1.0% ΔK/K.

The Group 1 rods may be withdrawn any time the core is in either the Hot Standby, Hot Shutdown, or Cold Shutdown Modes by increasing the boron concentration above that required by the shutdown margin with all control rods in.

The reason for withdrawing Group 1 when going through a heat-up to the Startup Mode (2) is to provide an extra margin of safety by having some worth for a scram should one be necessary. Thus the location of the Group 1 rods is chosen to be symmetrical and to have a worth on the order of 1.0% in reactivity. The results of the Group 1 worth are also compared to the stuck rod worth such that the reactivity difference can be included in the requirements for the shutdown boron concentration if necessary.

The location of the remaining safety groups, 2 and 3, are selected to be in the remaining locations, but may not be eighth core symmetrical. If the Group 1 rods are towards the interior, the Group 2 rods will be more towards the periphery and the Group 3 rods will be more towards the interior. If the Group 1 rods are towards the periphery, then the location of the Group 2 and 3 rods will be reversed. The Group 2 rods will be more towards the interior, and the Group 3 rods towards the periphery.

The methods and procedures used to analyze the reactivity and power peaking effects of the control rods are discussed in BAW-10118A and BAW-10122A as noted above. In order for these analyses to be valid, the location of the rod groups in the core must be the same as those in the Program Figure. The verification that the electrical connections of the rod groups do indeed correspond to the Rod Program Figure is specified in the Surveillance Requirements of Technical Specification 3.1.3.7. This surveillance requirement provides the link between the design analyses and core operation.

METHODOLOGY

REFUEL BORON CONCENTRATION

The Core Operating Limits Report contains the specification for the refuel boron concentration. The procedures and methods of analysis to determine the refuel boron concentration are described in the following paragraph.

The calculations are performed with the approved computer codes, (1) PDQ (BAW-10117A (P), "Babcock & Wilcox's Version of PDQ07--User's Manual, "H.A. Hassan, et. al., June, 1976), (2) NOODLE (BAW-10152A, "A Multi-Dimensional Two-Group Reactor Simulator," C.W. Mays, et. al., June, 1985), and (3) FLAME (BAW-10124A, "A Three-Dimensional NODAL Code for Calculating Core Reactivity and Power Distributions," C.W. Mays, May, 1976). Technical Specifications set the requirements on the value of the effective neutron multiplication factor (Keff) for the core. To this value and additional 1.0% in reactivity is added to account for uncertainties.

The formula to determine the refuel boron concentrations is:

Refuel Boron (ppm) = Base Case Boron (ppm)

+ Inverse Boron Worth (ppm /%$\Delta\rho$)

x (Base Case reactivity (ρ) + $\Delta\rho$(Shutdown Keff + 1.0%)

+ $\Delta\rho$ (Model Corrections) + $\Delta\rho$ (Uncertainty))

+ Δ ppm (Uncertainty)

The calculational procedure to determine the values of the terms in the above equation begins with the base case. The base case calculation has 140 F and 14.7 psia, BOL cold, conditions, no control rods inserted, and uses an estimated BOL cold boron concentration. The results provide the base case reactivity. To the base case reactivity three differential reactivities are added. The first is the differential reactivity from the critical condition to the shutdown Keff including a 1.0% reactivity increase for modeling uncertainties. The second differential reactivity corrects the base case for modeling biases. These biases are a result of the methods or approximations in the base case, such a two-dimensional calculation. The third differential reactivity is an adjustment for uncertainties in the core conditions. Conditions such as the BOL core burnup which is estimated while the previous cycle is operating. Any estimates or uncertainties include a reactivity increase to ensure conservative analyses.

The sum of the base case reactivity plus the second and third differential reactivities produce a total reactivity which is near zero. This small total reactivity is converted into a boron concentration by multiplying it by the inverse boron worth. The inverse boron worth is calculated from two NOODLE cases run at two different boron concentrations, bracketing the boron concentration required for refueling. The change in the boron concentration due to the total reactivity correction is added to the base case boron concentration to establish the refuel boron concentration.

The refuel boron concentration may be increased by an additional incremental amount to account for uncertainties in the methods and procedures and provide conservatism. The uncertainties and conservatisms would arise from the same type of conditions noted above for the third differential reactivity term (previous cycle length estimates, etc.).