Florida Power
CORPORATION

May 31, 1979

3-A-7

Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Sugject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Cycle 2 Reload

Dear Sir:

In your letter of May 23, 1979, you requested additional information in order to continue your review of our Cycle 2 reload for Crystal River Unit 3. The attached answers 1-4 address your Cycle 2 Questions and the attached answers 5-10 address your Physics Startup Test Questions for Cycle 2.

If you require any further information, please contact this office.

Sincerely,

FLORIDA POWER CORPORATION

W. P. Stewart

W. P. Stewart
Manager, Nuclear Operations

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STATE OF FLORIDA

COUNTY OF PINELLAS

W. P. Stewart states that he is the Manager, Nuclear Operations, of 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.

W. P. Stewart

W. P. Stewart

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 31st day of May, 1979.

Charles R Pope

Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: July 25, 1980

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RESPONSES TO NRC QUESTION ON
CRYSTAL RIVER 3 - CYCLE 2

Question:

1. In table (5-1) of Reference 1, design cycle 1 length should be corrected to 450 EFPD.

Answer: As explained in footnote (a) to table 5-1, the cycle 2 parameters listed are based on a cycle 1 length of 450 EFPD; the cycle 1 parameters (which are shown only for comparison) are the original cycle 1 design parameters - and that design was for 430 EFPD. This design was revised following the removal of BPRA/ORA to a design life of 510 EFPD, and design parameters for 510 EFPD can be found in Reference 2. When, for schedular reasons, the plant was shutdown at 450 EFPD, no new cycle 1 parameters were calculated.

Question:

2. In tables (5-1) and (7-1) of Reference 1, the following information is presented:

Table 5-1

I	Boron Worth - HFP, ppm/%Δk/k		
II	BOC	100 (cycle 1)	106 (cycle 2)
III	EOC	103 (cycle 1)	94 (cycle 2)
IV	Critical Boron - HFP, ppm		
V	BOC	1210 (cycle 1)	991 (cycle 2)

Table 7-1

I	Boron Worth - HFP, ppm/%Δk/k		
II		101 (cycle 1)	105 (cycle 2)
III	Initial Boron - HFP, ppm		
IV		795 (cycle 1)	1084 (cycle 2)

A. Please explain the difference between line II in both tables (5-1) and (7-1) above.

B. Explain the increased worth of the boron poison at EOC2 in line III of table (5-1).

C. Explain the difference between line V in table (5-1) and line IV in table (7-1).

Answer:

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A. The inverse boron worth (IBW) for BOC1 presented in table 5-1 is the IBW for the original cycle 1 design [3] (see footnote A in table 5-1). The IBW for BOC1 presented in table 7-1 is for the cycle 1 restart at 268 EFPD following removal of the lumped burnable

poison [2] (as identified by reference 3 in table 7-1). The IBW for BOC2 given in table 5-1 was calculated with all full length control rods withdrawn; the IBW for BOC2 given in table 7-1 was calculated with control rod bank 7 inserted.

B. Boron worth is a complex function of the neutron spectrum, fuel and fission product isotopic concentrations, and soluble boron concentration. The increase in soluble boron worth with burnup for cycle 2 of Crystal River III is typical of B&W's other reload cycles. Among factors contributing to the increase in boron worth with burnup are the following:

1. The boron concentration in the coolant decreases with burnup due to fuel depletion. Boron worth (%Δk/k/ppm) increases as the boron concentration decreases because of a reduction in self-shielding by the boron.

2. The neutron spectrum in the core becomes more highly thermalized with burnup, increasing the worth of boron which is primarily a thermal absorber.

3. The average macroscopic absorption cross section of the core decreases with burnup, resulting in less competition for neutrons and tending to increase the boron worth.

C. The BOC1 critical boron concentration given in table 5-1 is for the original cycle 1 design. The value given in table 7-1 is for the cycle 1 restart at 268 EFPD. Both BOC1 critical boron concentrations were calculated with control rod bank 7 fully inserted, as was the BOC2 critical boron concentration given in table 5-1. The BOC2 initial (critical) boron concentration in table 7-1 was calculated with bank 7 withdrawn.

Question:

3. In March, 1978, after BPRA and ORA removal, four partially burned Mark B2 fuel assemblies were loaded in the core to replace one damaged fuel assembly and three quarterly symmetric fuel assemblies (Reference 2). Those four fuel assemblies were not mentioned in the reload report.

A. Will the four fuel assemblies be unloaded from the core as part of batch 1, at EOC1?

B. If the answer to A above is no, has their exposure history been appropriately considered for cycle 2 operation?

C. If the four fuel assemblies will stay through cycle 2, do the Mark B2 fuel assemblies have a higher flow resistance than Mark B3 and Mark B4 And if they do, has that been considered in the DNBR analyses?

Answer: The four Mark B2 fuel assemblies will be discharged at the end of cycle 1. This can be confirmed by cross checking Figure 3-1 of Reference 1 with Figure 3-3 of Reference 2.

Question:

4. For the pellet resinter test please confirm the use of the model accepted by the NRC.

Answer: The pellet resinter test will be performed in accordance with BαW topical report BAW-10083PA, Rev. 1, "Babcock α Wilcox Model for Predicting In-Reactor Densification - Revision 1", July 1977 which was approved by the NRC on May 16, 1977.

Question:

5. Critical boron concentration

The criteria of ± 100 ppm is an acceptable value for an acceptance criteria with remedial action involving total solution of the problem before going above 5% power. A review criteria of more like ± 50 ppm corresponds to your started remedial actions. Please supply a review criteria, the basis for it and associated remedial action. Also, please supply the remedial actions associated with the ± 100 ppm criteria.

Answer: If the critical boron concentration is greater than 100 ppm from the predicted value, the plant conditions would be checked, additional boron samples would be analyzed and model inputs would be verified. The problem would then be resolved to the satisfaction of all concerned before power escalation would continue.

It is impossible with the data presently available to justify the basis for a review criteria of less than ± 100 ppm since this acceptance criteria is based on expected model accuracy rather than any specific safety limit.

Question:

6. Temperature Reactivity Coefficient

Please supply the basis for the $\pm .4 \times 10^{-4} \Delta k/k/°F$ value.

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Answer: The true acceptance criteria here is that the Moderator Temperature Coefficient (MTC) not exceed the Technical Specification limit of $+0.9 \times 10^{-4} \Delta k/k/°F$. The $\pm 0.4 \times 10^{-4} \Delta k/k/°F$ acceptance criteria of the Temperature Coefficient (TC) results from engineering judgment based on the measured TC being able to exceed its predicted value ($+0.14 \times 10^{-4} \Delta k/k/°F$) by this amount without the MTC exceeding the Technical Specification limit.

Question:

7. Control Rod Group Reactivity Worth

Criteria 1 with proposed remedial actions is acceptable.

Criteria 2 with proposed remedial actions is acceptable

for $\frac{P - M}{M} < -10\%$.

For $\frac{P - M}{M} > 10\%$ the remedial actions must be measurement

of additional bank(s) in order to verify shutdown margin. Please state that this will be the remedial action for $\frac{P - M}{M} > 10\%$

Answer: If $\frac{P - M}{M} > 10\%$, control rod group 4 will be measured by deboration.

Question:

8. Ejected Control Rod Reactivity Worth

Either all symmetric rods must be measured and compared for indication of tilt or there must be a quadrant symmetry test below 5% power. Please describe how this will be accomplished, the criteria and remedial actions.

Answer: We proposed that a quadrant symmetry check be made during the Core Power Distribution at 40% RTP. During escalation to 40%, beginning at 15%, the technical specification limits on quadrant tilt will be in effect, and will be monitored by the tilt alarms. If the technical specification limit is approached as power is escalated above 20% RTP, escalation would be halted until equilibrium xenon is attained and further escalation would depend on evaluation of available data. Power escalation would not be continued until a power distribution was analyzed and all parameters found to be within the technical specification values.

At power levels below 40% RTP, core safety would be compromised only by the grossest possible assymetries; fuel loading verification and the zero power test program in general assure that assymetries of this magnitude do not exist. Test for minor assymetries would be inaccurate, and unnecessary, below 40% RTP.

Question:

9. Core Power Distribution Verification

Please supply the acceptance and/or review criteria for the comparison of measured and predicted power distribution on an assembly by assembly basis. Also please state the appropriate remedial actions.

Answer: Since the Safety Analysis is performed for a single hot channel/pellet peak, the acceptance criteria between measured and predicted power distributions are based upon maximum values. This has been taken into account by developing criteria to assure that the measured maximum radial and total peaking do not exceed the calculated maximum anywhere in the core.

Inasmuch as the core power distribution (flux distribution) affects other nuclear parameters, these values are routinely measured during Physics Testing. During the normal Physics Test Program and Power Escalation Sequence safety related nuclear parameters, such as control rod worth and reactivity coefficient, are measured. These parameters must meet the acceptance criteria specified by the Safety Analysis prior to further power escalation.

This response is consistent with responses previously supplied by B&W to questions on topical report BAW-10119P, "Power Peaking Nuclear Reliability Factors", which was approved by the NRC on November 29, 1978.

Question:

10. Procedure For Use When Acceptance Criteria Are Not Met

Please state that the resolution of not meeting an acceptance or review criteria will be made by the on-site Safety Review Committee and that why failure to meet review or acceptance criteria does not pose a safety question will be in the Safety Committee meeting minutes which are reviewable by I&E.

Answer: Technical Specification 6.9.1.2 requires that the Startup Report "shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values

with design predictions and specifications." In addition, failure to meet any acceptance criteria would be brought to the attention of the Plant Review Committee and the problem resolved prior to escalation in power level. All actions taken by the committee are documented in the minutes which are reviewable by I&E.

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References: (1) Crystal River Unit 3, Cycle 2 Reload Report BAW-1521, February 1979.

(2) Crystal River Unit 3, Licensing Considerations for Continued Cycle 1 Operation Without Burnable Poison Rod Assemblies and Orifice Rod Assemblies, BAW-1490, Rev. 1, July 1978.

(3) Crystal River Unit 3, Final Safety Analysis Report Docket 50-302, Florida Power Corp.

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