NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-302

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TO: Mr. John Stolz

FROM: Flordia Power Corp.
St. Petersburg, Fla. 33733
J.T. Rodgers

DATE OF DOCUMENT 11-12-76

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DESCRIPTION Ltr trans the following:

ENCLOSURE Proposed FSAR Change-Amdt #50 re Table 14-8 entitled Summary of Moderator Dilution Accident Analysis....

(1 cy encl rec'd)

PLANT NAME: Crystal Rivaer Unit 3

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FOR ACTION/INFORMATION

SAFETY		ENVIRO	
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Florida Power
CORPORATION

November 12, 1976

Mr. John Stolz
Branch Chief
Light Water Reactors Branch I
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit #3
Docket No. 50-302

Dear Mr. Stolz:

On October 26, 1976, Florida Power Corporation (FPC) submitted a proposed FSAR revision to Table 14-8 on Page 14-61, which summarized under Item C, Dilution at Shutdown, the results for several initial boron concentration conditions examined by Babcock & Wilcox.

The results identified in Table 14-8 were to be used as the basis for lowering the refueling boron concentration contained in CR#3 Standardized Technical Specification 3.9.1.b from the present value of 2270 ppm.

Since our October 26, 1976 submittal, a new soluable boron level condition, corresponding to cold, K_{eff} = 1.00, no CRA in, was defined. Therefore, a reassessment of the refueling boron concentration was performed by Babcock & Wilcox.

The results of this assessment show that the core will remain subcritical if one makeup tank of demineralized water is emptied into the reactor vessel (Moderator Dilution Accident) initially at K_{eff} = .95 with 1875 ppm boron concentration.

Attached is Table 14-8 which has been revised under Item C, Dilution of Shutdown, to include the results of this latest B&W reassessment of the refueling boron concentration. This revised FSAR page will be included in the next amendment to the CR#3 Final Safety Analysis Report.

If you require additional discussion or clarification of our submittal, please do not hesitate to contact this office.

Very truly yours,

J. T. Rodgers
Asst. Vice President

JTR/iw
Attachment.

Table 14-8
Summary of Moderator Dilution Accident Analysis

A. Dilution at Power

Condition	Dilution Water Flow, gpm	Reactivity Rate, (Δk/k)/s	Average Reactor Coolant System Temp Change, °F/s
Normal	70	$+1.72 \times 10^{-6}$	0.005
Low RCS Pressure	100	$+2.45 \times 10^{-6}$	0.008
Maximum Considered	500	$+1.227 \times 10^{-5}$	0.034

B. Dilution to Trip

Dilution Water Flow, gpm	Peak Thermal Power, Percent of Rated	Peak Pressure psia	Time to Trip
70	103.2	2440	111
500	106.3	2452	48

C. Dilution at Shutdown

Initial Boron Concentration	Initial Total Subcriticality	Final Boron Concentration	Final Total Subcriticality
1875 ppm	-5%	1633 ppm	-1.78%
2270 ppm	-10.26%	1977 ppm	-6.36%

The results are based on the following assumptions:
(a) cold (70°), BOL, all rods out of the core
(b) emptying 1 full makeup tank volume of 600 $ft.^3$

Table 14-9
Pump Startup Accident Parameters

Moderator Coefficient, (Δ k/k/°F	-4.0×10^{-4}
Doppler Coefficient, (Δ k/k/°F	-1.3×10^{-5}

Table 14-10
Summary of Pump Startup Accident Analysis

Maximum Neutron Power (at 7 s), %	79
Maximum Thermal Power (at 8 s), %	65
Maximum Pressure Rise (at 12 s), psi	150