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TO: Mr. John Stolz

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

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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 River Unit 3

Do Not Remove

ACKNOWLEDGED

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SAFETY

FOR ACTION/INFORMATION

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ASSIGNED AD:
BRANCH CHIEF: (2) Stolz
PROJECT MANAGER: Engle
LIC. ASST.: Milton

ASSIGNED AD:
BRANCH CHIEF: Regan
PROJECT MANAGER: BAJWA
LIC. ASST.:

INTERNAL DISTRIBUTION

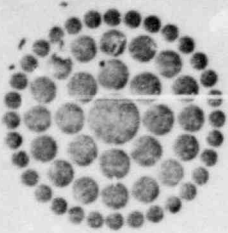
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<input checked="" type="checkbox"/> OELD		<input checked="" type="checkbox"/> LAINAS	
<input checked="" type="checkbox"/> GOSSICK & STAFF	ENGINEERING	<input checked="" type="checkbox"/> IPPOLITO	<input checked="" type="checkbox"/> ENVIRO TECH.
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			<input checked="" type="checkbox"/> SITE TECH.
<input checked="" type="checkbox"/> PROJECT MANAGEMENT	REACTOR SAFETY	<input checked="" type="checkbox"/> OPERATING TECH.	<input checked="" type="checkbox"/> GAMMILL
<input checked="" type="checkbox"/> BOYD	ROSS	<input checked="" type="checkbox"/> EISENHUT	<input checked="" type="checkbox"/> STEPP
<input checked="" type="checkbox"/> P. COLLINS	NOVAK	<input checked="" type="checkbox"/> SHAO	<input checked="" type="checkbox"/> HULMAN
<input checked="" type="checkbox"/> HOUSTON	ROSZTOCZY	<input checked="" type="checkbox"/> BAER	
<input checked="" type="checkbox"/> PETERSON	CHECK	<input checked="" type="checkbox"/> BUTLER	<input checked="" type="checkbox"/> SITE ANALYSIS
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	RUTBERG		<input checked="" type="checkbox"/> KREGER

EXTERNAL DISTRIBUTION

LPR Crystal River, Fla. AT LAB: PNWL-2 BROOKHAVEN NAT LAB
 TIC: REG. VIE ULRIKSON (CENL)
 NSIC: LA PDR
 ASLB: CONSULTANTS
 ACRS 16 CYS SENT CAT "B" Blume 8008120748

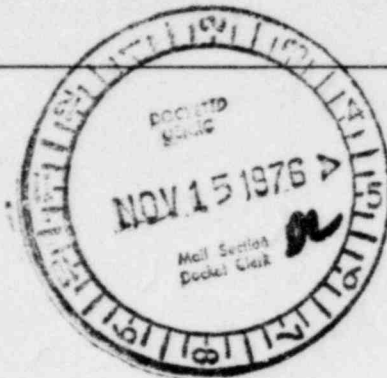
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Regulatory Docket File

**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 soluble 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.

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Mr. John Stolz

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November 12, 1976

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. F. Rodgers
Asst. Vice President

JTR/iw
Attachment.

Table 14-8
Summary of Moderator Dilution Accident Analysis

Regulatory Document 1119

A. Dilution at Power

<u>Condition</u>	<u>Dilution Water Flow, gpm</u>	<u>Reactivity Rate, ($\Delta k/k$)/s</u>	<u>Average Reactor Coolant System Temp Change, °F/s</u>
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

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

C. Dilution at Shutdown

<u>Initial Boron Concentration</u>	<u>Initial Total Subcriticality</u>	<u>Final Boron Concentration</u>	<u>Final Total Subcriticality</u>
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, ($\Delta k/k/°F$)	-4.0×10^{-4}
Doppler Coefficient, ($\Delta k/k/°F$)	-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