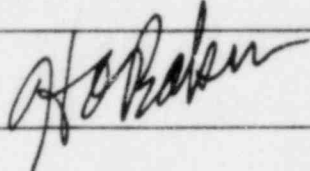


THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To R. J. Trost, Project Management

From H. A. Baker, Project Engineering



BDS 663.5

Cust. Florida Power Corp.

File No.
or Ref. FPC- Reload

Subj. CR-3, Cycle 3 Reload Report Amendment

Date
May 30, 1980

This letter to cover one customer and one subject only.

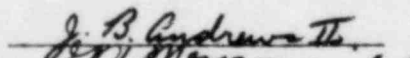
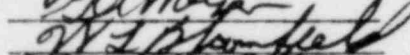
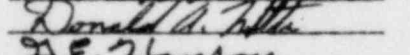
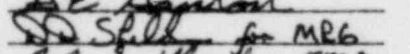
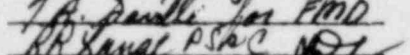
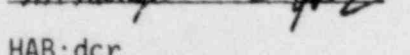


During the recent refueling at Crystal River 3, a broken holddown spring was discovered in assembly NJ018E, a batch 4 assembly that had been in core location N-14 during cycle 2. This assembly and three symmetric assemblies have been replaced with batch 5 fuel. Thus, fifty-two batch 2 assemblies and four batch 4 assemblies have been discharged and fifty-six batch 5 assemblies will be loaded for cycle 3. The necessary changes to the Crystal River Unit 3 Cycle 3 Reload Report BAW-1607, Rev. 1, April, 1980 are attached.

Figures 3-1, 3-2, Table 5-2, and Figure 5-1, have been updated to reflect the changes in the core loading. With the exception of the stuck rod worth at BOC and EOC, as shown in the revised Table 5-2, all values quoted in Table 5-1 of Reference 1 change by $\leq 1\%$. It was not deemed necessary to make these insignificant changes.

The safety, control and power peaking analyses performed previously, with and without pump monitors, at power levels of 2452 MWt and 2544 MWt and forwarded to FPC by letter on May 6, 1980 have been evaluated and remain valid. No changes to the technical specifications are required by the modified core loading.

In addition, two corrections have been included as the result of discussions with Florida Power. In paragraph 7.12, page 7-8, the cycle 3 predicted value of maximum rod worth was corrected to $0.49\% \Delta k/k$ from the $0.59\% \Delta k/k$ value. In Table 7-1, on page 7-13, the Dropped Rod Worth was corrected from 0.65 to 0.40.

Due to the small number of pages effected by this change, we have elected to revise only the necessary pages and not rev the entire report. The revised pages have been noted. Those on distribution of this memo should incorporate the attached pages in their copy of the reload report.

	Safety Analysis
	Thermo-Hydraulic Engineering
	ECCS
	RADIATION ANALYSIS
	Nuclear Operations Analysis
	Fuel Management and Development
	
	

HAB:dcr

The current and potential transformers are not seismically qualified. However, separation of the cables carrying redundant transformer outputs to the RCPPM cabinets is provided in accordance with the separation criteria stated above. The current and potential transformers are not seismically qualified because they are not required to safely shutdown the reactor. The loss of the current or potential transformers would result in a "pump inoperable" signal to the RPS. Upon receipt of two such signals, whatever the cause, the RPS trips the reactor.

3.2. Core Description

The CR-3 reactor core is described in detail in Chapter 3 of the Final Safety Analysis Report for the unit.¹ The cycle 3 core consists of 177 fuel assemblies (FAs), each of which is a 15-by-15 array containing 208 fuel rods; 16 control rod guide tubes; and one incore instrument guide tube. The fuel assemblies in batches 2, 3, and 5 have an average nominal fuel loading of 463.6 kg of uranium, whereas the batch 4 assemblies maintain an average nominal fuel loading of 468.6 kg of uranium. The cladding is cold-worked Zircaloy-4 with an OD of 0.430 inch and a wall thickness of 0.0265 inch. The fuel consists of dished-end, cylindrical pellets of uranium dioxide (see Table 4-2 for data).

Figure 3-1 is the core loading diagram for cycle 3 of Crystal River 3. The initial enrichments of batches 2, 3, and 4 were 2.54, 2.83, and 2.64 wt % uranium-235, respectively. Fifty-two batch 2 and four batch 4 assemblies will be discharged at the end of cycle 2. The batch 5 design enrichment is 2.62 wt % uranium-235. Batches 3 and 4 and the remaining batch 2 assemblies will be shuffled to new locations. The batch 5 assemblies will occupy the periphery of the core. Figure 3-2 is an eighth-core map showing the burnup of each assembly at the beginning of cycle 3 and its initial enrichment.

Core reactivity will be controlled by 61 full-length Ag-In-Cd control rod assemblies (CRAs) and soluble boron shim. In addition to the full-length CRAs, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The cycle 3 locations of the 69 control rods and the group designations are unchanged from cycle 2 and are shown in Figure 3-3. Control rod group 7 will be withdrawn at 250 ± 10 EFPD of operation.

Figure 3-1. Core Loading Diagram for Crysta River 3, Cycle 3

															XX	Cycle 2 Location	
															Y	Batch Number	
A						5	5	5	5	5							
B					5	5	5	F7 3	C9 3	F9 3	5	5	5				
C			5	5	D7 3	N3 4	L1 4	P8 2	L15 4	N13 4	D9 3	5	5				
D	5	5	C7 3	O3 4	M2 4	D5 3	R8 4	D11 3	M14 4	C13 4	G13 3	5	5				
E	5	G4 3	B12 4	F6 2	K5 3	K1 4	L8 3	K15 4	K11 3	F10 2	B4 4	G12 3	5				
F	5	5	C12 4	B11 4	E9 3	E5 3	D6 3	B10 4	D10 3	E11 3	E7 3	B5 4	C4 4	5	5		
G	5	G6 3	A10 4	E4 3	A9 4	F4 3	B6 4	D8 3	F14 4	F12 3	A7 4	E12 3	A6 4	G10 3	5		
H	5	G3 3	H14 2	H15 4	H10 3	F2 4	H4 3	H8 2	H12 3	L14 4	H6 3	H1 4	H2 2	K13 3	5		
K	5	K6 3	R10 4	M4 3	R9 4	L4 3	L2 4	N8 3	P10 4	L12 3	R7 4	M12 3	R6 4	K10 3	5		
L	5	5	O12 4	P11 4	M9 3	M5 3	N6 3	P6 4	N10 3	M11 3	M7 3	P5 4	O4 4	5	5		
M			5	K4 3	P12 4	L6 2	G5 3	G1 4	F8 3	G15 4	G11 3	L10 2	P4 4	K12 3	5		
N			5	5	K3 3	C3 4	E2 4	N5 3	A8 4	N11 3	E14 4	C13 4	O9 3	5	5		
O				5	5	N7 3	D3 4	F1 4	B8 2	F15 4	D13 4	N9 3	5	5			
P					5	5	5	L7 3	O7 3	L9 3	5	5	5				
R						5	5	5	5	5							
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15		

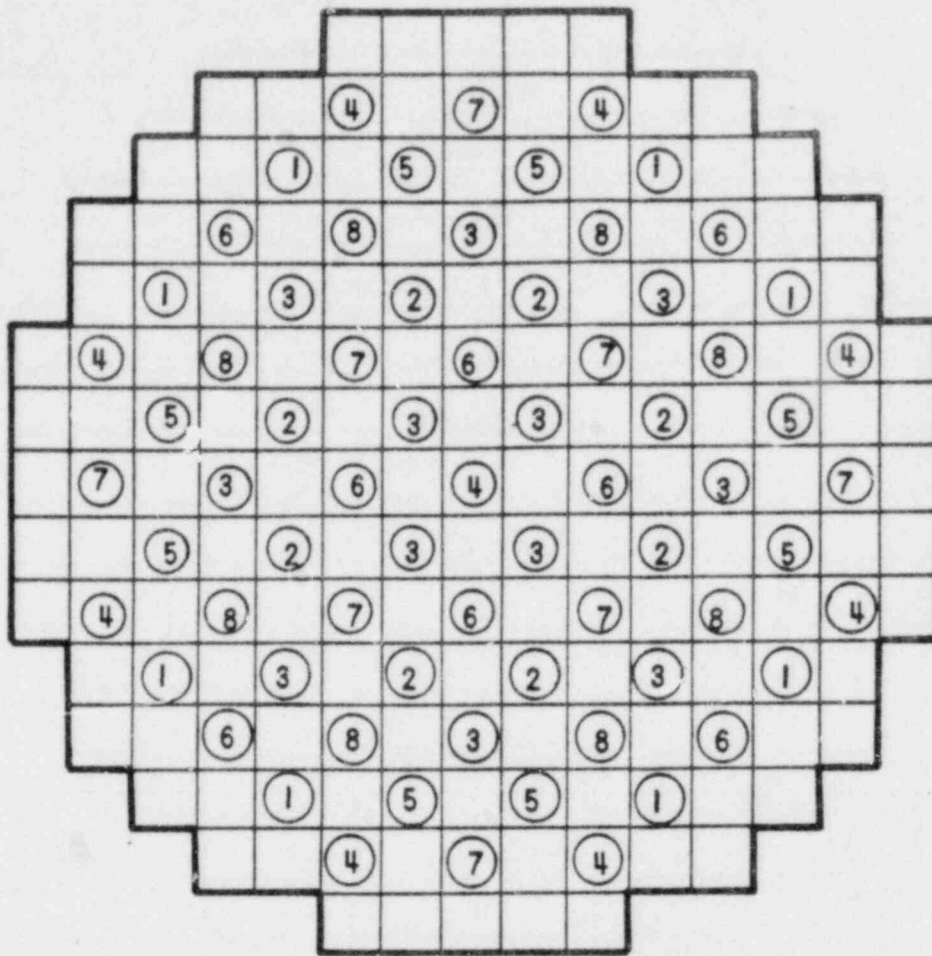
Figure 3-2. Enrichment and Burnup Distribution for
Crystal River 3, Cycle 3

	8	9	10	11	12	13	14	15
H	2.54 17,015	2.83 16,512	2.64 5,691	2.83 19,762	2.64 3,085	2.54 16,741	2.83 15,949	2.62 0
K		2.64 5,690	2.83 12,950	2.64 3,452	2.83 17,364	2.64 3,206	2.83 15,308	2.62 0
L			2.83 14,095	2.83 13,590	2.64 4,923	2.64 6,051	2.62 0	2.62 0
M				2.54 17,460	*2.64 3,639	2.83 17,466	2.62 0	
N				*2.64 4,231	2.83 15,950	2.62 0	2.62 0	
O						2.64 0		
P								
R								

x.xx	Initial enrichment
xx,xxx	BOC burnup, MWd/mtU

*Signifies asymmetric (1/4 core) shuffled assemblies.

Figure 3-3. Control Rod Locations



GROUP	NUMBER OF RODS	FUNCTION
1	8	SAFETY
2	8	SAFETY
3	12	SAFETY
4	9	SAFETY
5	8	CONTROL
6	8	CONTROL
7	8	CONTROL
8	8	APSRs
TOTAL	69	

Table 5-2. Shutdown Margin Calculation for Crystal River 3, Cycle 3

	<u>BOC, %Δk/k</u>	<u>EOC^(a), %Δk/k</u>
<u>Available Rod Worth</u>		
Total rod worth, HZP ^(b)	9.25	9.21
Worth reduction due to burnup of poison material	-0.37	-0.42
Maximum stuck rod worth, HZP	<u>-1.82</u>	<u>-1.79</u>
Net worth	7.06	7.00
Less 10% uncertainty	<u>-0.71</u>	<u>-0.70</u>
Total available worth	6.35	6.30
<u>Required Rod Worth</u>		
Power deficit, HFP to HZP	1.30	2.08
Max allowable inserted rod worth	1.06	1.36
Flux redistribution	<u>0.53</u>	<u>1.02</u>
Total required worth	2.89	4.46
<u>Shutdown Margin</u>		
Total available minus total required	3.46	1.84

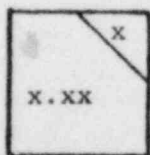
Note: Required shutdown margin is 1.00% Δk/k.

(a) For shutdown margin calculations, this is defined as ~250 EFPD, the latest time in core life in which the transient bank is nearly full-in.

(b) HZP: hot zero power, HFP: hot full power.

Figure 5-1. BOC (4 EFPD), Cycle 3 Two-Dimensional Relative Power Distribution - HFP, Equilibrium Xenon, Banks 7 and 8 Inserted

	8	9	10	11	12	13	14	15
H	1.08	1.14	1.30	1.17	1.34	0.94	0.46	0.50
K		1.27	1.14	1.31	1.16	1.22	0.82	0.57
L			0.67	1.03	1.15	1.24	1.08	0.54
M				1.01	*1.28	1.04	0.89	
N				*1.25	1.11	1.10	0.62	
O						0.73		
P								
R								



Inserted rod group No.

Relative power density

*Denotes only impact of 1/4-core symmetry greater than 0.01.

The FSAR has identified a double-ended rupture of the steam line between the steam generator and steam stop valve as the worst-case situation at end-of-life conditions.

The key parameter for the core response is the moderator temperature coefficient which in the FSAR was assumed to be $-3.0 \times 10^{-4} \Delta k/k/^{\circ}F$. The cycle 3 predicted value of moderator temperature coefficient is $-2.63 \times 10^{-4} \Delta k/k/^{\circ}F$. This value is bounded by that used in the FSAR analysis; hence, the results in the FSAR represent the worst situation.

The FSAR used an initial power level of 2568 MWt for these accidents. This is more conservative than running the accident at 102% of 2544 MWt and tripping the reactor at 110% versus the current 112% setpoint since more energy is added to the system for the FSAR analysis.

7.10. Steam Generator Tube Failure

A rupture or leak in a steam generator tube allows reactor coolant and associated activity to pass to the secondary system. The FSAR analysis is based on complete severance of a steam generator tube. The primary concern for this incident is the potential radiological release. The environmental dose assessment is presented in section 7.18.

7.11. Fuel Handling Accident

The mechanical damage type of accident is considered the maximum potential source of activity release during fuel handling activity. The primary concern is over radiological releases. The environmental dose assessment is presented in section 7.18.

7.12. Rod Ejection Accident

For reactivity to be added to the core at a more rapid rate than by uncontrolled rod withdrawal, physical failure of a pressure barrier component in the CRDA must occur. Such a failure could cause a pressure differential to act on a CRA and rapidly eject the assembly from the core. This incident represents the most rapid reactivity insertion that can be reasonably postulated. The values used in the FSAR and densification report at BOL conditions of $-1.17 \times 10^{-5} \Delta k/k/^{\circ}F$ Doppler coefficient, $0.0 \Delta k/k/^{\circ}F$ moderator temperature coefficient, and ejected rod worth of $0.65\% \Delta k/k$ represented the maximum possible transient.

The use of a 0.65% $\Delta k/k$ maximum rod worth is conservative in comparison to the cycle 3 predicted value of 0.49% $\Delta k/k$. Furthermore, the cycle 3 predicted values of $-1.52 \times 10^{-5} \Delta k/k/^\circ F$ Doppler and $-0.30 \times 10^{-5} \Delta k/k/^\circ F$ moderator temperature coefficient are both more negative than used in the FSAR analysis. | 1

The FSAR used an initial rated power level of 2568 MWt for this accident. This is more conservative than initializing the accident at 102% of 2544 MWt and tripping the reactor at 110% versus the current 112% setpoint since more energy is added to the system for the FSAR analysis. For the accident which trip on high pressure, the effect of higher initial power level (i.e., 102% of 2544 MWt) is to cause the pressure trip to occur slightly sooner. Since the FSAR input bound the cycle 3 predicted values, the results in the FSAR and densification report are applicable to this reload.

7.13. Maximum Hypothetical Accident

There is no postulated mechanism whereby this accident can occur since this would require a multitude of failures in the engineered safeguards. The hypothetical accident is based solely on a gross release of radioactivity to the reactor building. The environmental dose assessment is presented in section 7.18.

7.14. Waste Gas Tank Rupture

The waste gas tank was assumed to contain the gaseous activity evolved from degassing all the reactor coolant following operation with 1% defective fuel. Rupture of the tank would result in the release of its radioactive contents to the plant ventilation system and to the atmosphere through the unit vent. The environmental dose assessment is presented in section 7.18.

7.15. LOCA Analysis

Generic LOCA analyses for B&W 177-FA lowered-loop NSSs have been performed using the Final Acceptance Criteria ECCS Evaluation Model. The large-break analysis is presented in a topical report¹³, and is further substantiated in a letter report¹⁴. The small break analysis is presented in a letter report¹⁵. These analyses used the limiting values of key parameters for all plants in the category. Furthermore, the average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the LOCA limits analysis¹³ are conservative compared to those calculated for this reload. Thus, these

Table 7-1. Comparison of Key Parameters for Accident Analysis

<u>Parameter</u>	<u>FSAR¹, densif'n value⁷</u>	<u>Cycle 1¹¹</u>	<u>Cycle 3 value</u>
BOL Doppler coeff, $10^{-5} \Delta k/k/^{\circ}F$	-1.17	-1.47 (268 EFPD)	-1.52
EOL Doppler coeff, $10^{-5} \Delta k/k/^{\circ}F$	-1.30	-1.66 (510 EFPD)	-1.61
BOL moderator coeff, $10^{-4} \Delta k/k/^{\circ}F$	0 ^(a)	-0.75 (268 EFPD)	-0.30
EOL moderator coeff, $10^{-4} \Delta k/k/^{\circ}F$	-4.0 ^(b)	-2.42 (510 EFPD)	-2.63
All-rod bank worth at BOL, HZP, % $\Delta k/k$	12.9	9.12 (268 EFPD)	9.37
Boron reactivity worth (HFP), ppm/1% $\Delta k/k$	100	101	108
Max ejected rod worth (HFP), % $\Delta k/k$	0.65	0.55	0.49
Dropped rod worth (HFP), % $\Delta k/k$	0.40	0.20	0.20
Initial boron conc'n (HFP), ppm	1150	795	1185

(a) $+0.50 \times 10^{-4} \Delta k/k/^{\circ}F$ was used for the moderator dilution accident.

(b) $-3.0 \times 10^{-4} \Delta k/k/^{\circ}F$ was used for the steam line failure analysis and dropped rod accident analysis.

Table 7-2. Bounding Values for Allowable
LOCA Peak Linear Heat Rates

<u>Core elevation, ft</u>	<u>Allowable peak LHR, kW/ft</u>
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

Table 7-3. Input Parameters to Loss-of-Coolant-Flow Transients

	<u>Cycle 3 value</u>	<u>Value used in analysis</u>
Initial flow rate, % of 352,000 gpm	>109.5	106.5
Flow rate Vs time	>Fig. 14-17, FSAR Fig. 14-19a, FSAR	Fig. 14-17, FSAR (4PCD) Fig. 14-19a, FSAR (LR)
Initial power level, MW	2544	102% of 2568
Doppler coeff, $\Delta k/k/^\circ F$	-1.52×10^{-5}	-1.27×10^{-5}
Moderator temp coeff, $\Delta k/k/^\circ F$	-0.30×10^{-4}	0
F Δ H	1.47	1.71

Table 7-4. Summary of Minimum DNBR Results for Limiting Loss-of-Coolant-Flow Transients

<u>Transient</u>	<u>Cycle 1</u>			
	<u>FSAR¹</u> <u>(W-3)</u>	<u>Densif'n</u> <u>report</u> <u>(W-3)</u>	<u>Cycle 2</u> <u>(B&W-2)</u>	<u>Cycle 3</u> <u>(B&W-2)</u>
One-pump coastdown (flux/flow trip)	NR ^(a)	NR	1.75	1.75
Four-pump coastdown (flux/flow trip, cycle 1; pump monitor trip, cycles 2 and 3)	1.45	1.39	2.10	2.10

(a) NR: not reported.