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EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards minor corrections to 810723 analysis for main steam line break event.Conclusions remain valid.

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FLORIDA POWER & LIGHT COMPANY

October 20, 1981
L-81-457

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington DC 20555

Dear Mr. Eisenhut:

Re: St. Lucie Unit 1
Docket No. 50-335
Main Steam Line Break Analysis

The purpose of this letter is to forward minor corrections to the analysis for the St. Lucie Unit 1 Main Steam Line Break event (Reference 1).

Vendor quality assurance checks on the results previously forwarded revealed that a non-conservative calculational value of 0.85 was used for the Doppler Coefficient multiplier and that Beginning-of-Cycle Doppler feedback data had been used rather than End-of-Cycle values as had been indicated in Reference 1.

Reanalysis has been performed using a more conservative multiplier value of 1.15 and End-of-Cycle Doppler feedback data. The magnitude of both the predicted return to power and the maximum core reactivity peaks increased but remained inside established safety criteria. Therefore, the conclusions presented in Reference 1 remain valid.

The attached tables correct the list of parameters and the sequence of events previously forwarded. These tables supersede Table 7.3.2-3 and 7.3.2-4 of Reference 1. Values changed by reanalysis are indicated by vertical lines.

Very truly yours,

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/WEW/jc

cc: Mr. J. P. O'Reilly, Region II
Harold F. Reis, Esquire

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REFERENCES

(1) FPL letter L-81-306, "Proposed License Amendment to Facility Operating License DPR-67; Shutdown Margin, Main Steam Isolation System Changes, and Control Element Assembly Sleeving", dated July 23, 1981.

TABLE 7.3.2-3

KEY PARAMETERS ASSUMED IN THE STEAM LINE RUPTURE ANALYSIS
2-LOOP, NO LOAD

Parameter	Units	Reference Cycle*	Cycle 5
Initial Core Power	MWt	1.0	1.0
Initial Core Inlet Temperature	°F	532	532
Initial RCS Pressure	psia	2300	2300
Initial Steam Generator Pressure	psia	900	900
Minimum CEA Worth Available at Trip	%$\Delta\rho$	-4.3	-5.0
Doppler Multiplier		1.15	1.15
Moderator Cooldown Curve	%$\Delta\rho$ vs. density	See Figure 7.3.2-10	See Figure 7.3.2-10
Inverse Boron Worth	PPM/%$\Delta\rho$	100	80
Effective Full Power MTC	$10^{4}\Delta\rho$/°F	-2.5	-2.2
β Fraction (including uncertainty)		.0060	.0060

*Cycle 4 stretch power submittal

TABLE 7.3.2-4

SEQUENCE OF EVENTS FOR STEAM LINE RUPTURE EVENT WITH AUTOMATIC INITIATION OF AUXILIARY FEEDWATER AND MANUAL TRIP OF REACTOR COOLANT PUMPS 2-LOOP, NO LOAD

Time (sec)	Event	Setpoint or Value
0.0	Steam Line Rupture Occurs	---
2.5	Low Steam Generator Pressure Trip Signal Occurs	578.0 psia
2.5	Main Steam Isolation Signal Generated	578.0 psia
3.4	Main Steam Isolation Valves Begin to Close	---
3.4	Trip Breakers Open	---
3.9	CEAs Begin to Drop Into Core	---
9.3	Main Steam Isolation Valves Completely Closed	---
10.4	Pressurizer Empties	---
12.4	Safety Injection Actuation Signal	1578.0 psia
12.4	Reactor Coolant Pumps Manually Tripped	---
42.4	High Pressure Safety Injection Pumps Start	---
64.0	Main Feedwater Isolation	---
180.0	Affected Steam Generator Blows Dry	---
180.1	Auxiliary Feedwater Flow Initiated to Ruptured Steam Generator	183 lbm/sec
305.0	Peak Reactivity	+.290% $\Delta\rho$
524.0	Return to Power	2.5% of rated power
600.0	Operator Isolates Ruptured Steam Generator and Terminates Auxiliary Feedwater Flow	---