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AUTH.NAME AUTHOR AFFILIATION
UHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Submits response to NRC questions re safety analysis for positive moderator temp coefficient operation. "Reload Safety Evaluation,Turkey Point Plant Unit 4,Cycle 6,Revision 1" encl.

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FLORIDA POWER & LIGHT COMPANY

January 20, 1982
L-82-24

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

Dear Mr. Eisenhut:

Re: Turkey Point Unit 3 & 4
Docket Nos. 50-250 and 50-251
Proposed License Amendment
Moderator Temperature Coefficient

On December 10, 1981 we sent you the subject proposed amendment with an attached safety evaluation (FPL letter L-81-517). Mr. Ron Frahm requested clarification concerning this submittal which was discussed by telephone on January 18, 1982. A formal response is submitted with this letter. Since the response refers to the Reload Safety Evaluation of Turkey point Unit 4 Cycle 6, a copy of it is also attached.

Very truly yours,

J A De Mastry
for

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/SKM/jc

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

Attachment

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FPL Response to
the NRC Questions on the Safety
Analysis for Positive Moderator
Temperature Coefficient Operation
for Turkey Point Units 3 & 4

Question: 1

The analysis of the Rod Ejection indicates a peak hot spot clad temperature of 2210-F in the write up while a value of 2367-F is indicated in Table III. Which is the correct value?

Response:

The value of 2367-F indicated in Table III is the correct value.

Question: 2

The analysis of the Rod Ejection concludes that the fuel and clad temperature limits specified in the FSAR are not exceeded. Have the limits specified in WCAP 7588 been verified?

Response:

In the present analysis, the fuel performance values are below the limits specified in WCAP 7588.

Question: 3

In the Locked Rotor analysis, the FSAR quotes a value of 2510-F for peak average pellet temperature. What is the corresponding value for the present analysis?

Response:

The peak average pellet temperature during the transient calculated for the present analysis is 2137-F.

Question: 4

For the Uncontrolled Rod Withdrawal, peak heat flux in the present analysis is much higher than the FSAR value. Explain the reasons for this behavior and the consequences.

Response:

The FSAR analysis is based on a reactivity insertion rate of 60X10-5 delta-k/sec while the present analysis is based on a reactivity insertion rate of 75X10-5 delta-k/sec. Also the doppler power defect and prompt neutron lifetime are different from the FSAR values (see Unit 4, Cycle 6 RSE). The following description of this accident analysis is provided to adequately address the question. The description refers to figures 2, 3 & 4 of the subject submittal.

Control Rod Withdrawal From a Subcritical Condition

I INTRODUCTION

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 14.1.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient. The power excursion causes a heatup of the moderator and fuel. The reactivity addition due to a positive moderator coefficient could result in increases in peak heat flux, peak fuel, and clad temperatures. The time the core is critical before a reactor trip is very short so that the coolant temperature does not increase significantly. Hence, the effect of a positive moderator coefficient is small.

II METHOD OF ANALYSIS

The analysis was performed in Unit 4 Cycle 6 RSE for a reactivity insertion rate of 75X10-5 delta-k/sec. This reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). A constant moderator temperature coefficient of +5 pcm/degrees-F was used in the analysis. The digital computer codes, initial power level, and reactor trip instrument delays and setpoint errors used in the analysis were the same as used in the FSAR, subsequent safety analyses, and WCAP-8284 Rev. 2, (Florida Power and Light - Turkey Point Units 3 and 4 - Precautions, limitations, and Setpoints).

III RESULTS AND CONCLUSIONS

The nuclear power, coolant temperature, heat flux, fuel average temperature, and clad temperature versus time for a 75 X 10-5 delta-k/sec insertion rate are shown in Figures 2 through 4. This insertion rate, coupled with a positive moderator temperature coefficient of +5 pcm/degrees-F, yields a peak heat flux which does not exceed the nominal value.

Taking into account the conservative assumptions with which the accident has been analyzed, it is concluded that in the unlikely event of a control rod withdrawal accident, the core and reactor coolant systems are not adversely affected since the thermal power reached is less than the nominal value and the core water temperature reached is 564 degrees-F compared to 577 degrees-F for the nominal conditions. These conditions (subcooled coolant, less than nominal heat flux) show that the minimum DNBR is well above 1.30. No damage could occur to the cladding due to low temperature (less than 650 degrees-F) if compared to the melting point (greater than 3200 degrees-F).