NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-251

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: MR G LEAR

FROM: FLORIDA POWER & LIGHT CO
MIAMI, FLA
R E UHRIG

DATE OF DOCUMENT 4-21-76

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DESCRIPTION

LTR RE OUR 4-7-76 LTR....FURNISHING ADD'L RELOAD INFORMATION TO QUESTIONS PREVIOUSLY REQUESTED.............

PLANT NAME: TURKEY POINT 4

ENCLOSURE

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ACKNOWLEDGED

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	ASSIGNED AD :			ASSIGNED AD :
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X	CASE		KNIGHT		OPERATING REACTORS		GAMMILL
	HANAUER		SIHWEIL		STELLO		STEPP
	HARLESS		PAWLICKI				HULMAN
					OPERATING TECH		
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X	BOYD	X	ROSS	✓	SHAO		VOLLMER
	P. COLLINS	X	NOVAK (3)	✓	BAER		BUNCH
	HOUSTON	X	ROSZTOCZY	✓	SCHWENCER	✓	J. COLLINS
	PETERSON		CHECK	✓	GRIMES		KREGER
	MELTZ						
	HELTEMES		AT & I		SITE SAFETY & ENVIRO		
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CONTROL NUMBER 4171

NRC FORM 195 (2-76)

P. O. BOX 013100, MIAMI, FLORIDA 33101

FPL

FLORIDA POWER & LIGHT COMPANY

April 21, 1976
L-76-169

Director of Nuclear Reactor Regulation
Attn: Mr. George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Lear:

Re: Turkey Point Unit 4
Docket No. 50-251
Additional Reload Information

Your letter of April 7, 1976 requested additional information concerning Florida Power and Light Company's reload submittal for core Cycle 3 of Unit 4. The following input, numbered to correspond to your questions, is provided.

1. Question

For each transient and accident including the loss-of-coolant-accident discussed in Chapter 14 of the FSAR, list all the nuclear, thermal and fuel design parameters which have impact on the event and compare these values with the values used to analyze these events for Cycle 3. If the event was not re-analyzed for Cycle 3 justify this by showing that the power distribution and the parameters listed above for Cycle 3 are included within the envelope of the FSAR parameters.

In particular, for the case of the loss-of-coolant-accident address the effect of changes in fuel rod pressure on the fuel rod behavior during a loss-of-coolant-accident.

Answer

All of the accidents discussed in Chapter 14 of the FSAR which could be affected by the reload have all been evaluated for Cycle 3. However, the Reload Safety Evaluation contains only a discussion of the differences rather than a complete accident review. This is consistent with Item 4 of a letter from K. R. Goller (NRC) to C. Eicheldinger (Westinghouse) dated October 29, 1975.

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"4. A proposed reload may involve changes in only a few of the areas identified in the guidance. The submittal therefore need address only that information necessary to demonstrate all aspects have been appropriately considered. We do not intend that every reload submittal necessarily contain a complete discussion of all areas indicated in the guidance. Only that information necessary to support the needed technical specification change or to resolve the unreviewed safety question is necessary."

If you have specific concerns relative to an accident, we would be happy to address them.

The effect of small (i.e., 65 psi) changes in hot fuel rod internal pressure, with respect to the loss-of-coolant accident (LOCA) for Unit 4 is negligible. The change in pressure will alter the pellet-clad gap conductance and the fuel rod swelling/burst calculations to a very small degree and the reduction in pressure (compared to Cycle 2) will probably result in a reduction in peak clad temperature.

2. <u>Question</u>

How many steam generator tubes are plugged in Turkey Point 4? Discuss effect of plugged steam generator tubes on the results of the loss-of-coolant analysis for Unit 4.

<u>Answer</u>

The impact of plugging steam generator tubes on ECCS performance has been assessed. Current analysis indicates that plugging 1% of the total number of steam generator tubes results in an ECCS evaluation peak clad temperature increase of approximately 10°F. Considering that the present ECCS analyses conservatively calculate steam generator performance during a LOCA, the current LOCA results and the current level of tube plugging in Unit 4 (≈3.4% of total), a safety concern does not exist.

3. <u>Question</u>

Table 2 of the Cycle 3 reload submittal gives the limits on the moderator temperature coefficients for Cycle 3. Please give the expected range of values of this coefficient and the uncertainty associated with these numbers for operation during Cycle 3. Explain what conditions were assumed for this calculation. What part do core measurements play in determining the expected values of these coefficients?

Answer

A direct comparison of calculated coefficients for a particular core condition (power level, burnup, rodded configuration) with the design limit values of Table 2 could be misleading since in many instances the most conservative combination of reactivity coefficients is used in the transient analysis even though the extreme coefficients assumed may not simultaneously occur at the conditions of lifetime, power level, temperature, and boron concentration assumed in the analysis.

The temperature coefficient is required by the technical specifications to be negative for all operating conditions except physics tests. If the startup physics tests show that it is necessary to insert control rods to obtain a negative coefficient, a control rod withdrawal or a maximum boron concentration limit is imposed to insure the coefficient is negative throughout power escalation.

For Turkey Point Unit 4, Cycle 3 the calculated HZP-ARO moderator temperature coefficient is negative. It is not expected that a control rod withdrawal or boron concentration limit will be needed.

4. Question

(a) Explain the bases for the rod insertion limits, that is, what factors are considered in their selection?

(b) Show that the rod insertion limits are acceptable in terms of the response to part (a).

(c) Explain what factors are involved in making the rod insertion limits different for 3 loop and 2 loop operation.

Answer

(a) In the course of design, an insertion limit is selected which is estimated to meet the insertion limit criteria. The following criteria are then checked in the design process:

1. The shutdown margin is maintained by calculating the inserted reactivity (reactivity allowance) for the control rods at the insertion limit.

2. For rod positions allowed in normal operation, the enthalpy rise hot channel factor, $F_{\Delta H}$, must be maintained within limits.

3. The consequences of an ejected control rod assembly from the allowed insertion must be within the accepted limits.

4. The trip reactivity assumed in the accident analysis must be available.

5. Statically misaligning a control assembly will not violate the thermal design basis with respect to DNBR.

6. The uncontrolled withdrawal of a control assembly bank will not result in a peak power density that exceeds the center line melting criterion.

The bases for rod insertion limits were also discussed for Cycle 3 of Unit 3 in FPL letter L-75-523 of October 29, 1975 from R. E. Uhrig to George Lear.

(b) If any of the above are not met, the insertion limit must be adjusted accordingly. The limiting constraint was maintaining the shutdown margin at end of life Cycle 3. The design requirements for Cycle 3 were met by confirming that the above criteria are satisfied for the Cycle 3 insertion limits.

(c) The control rod insertion limits for both two and three loop operation are selected to meet the above criteria. The core conditions for the two conditions are different. Two loop operation has a reduced core flow and a higher inlet temperature. (Note also that two loop operation has a reduced allowable power level.) Considering these differences when establishing the control rod insertion limit for two loop operation results in limits which are more restrictive than those for three loop operation. In particular, the limiting criterion for two loop operation becomes criterion 2 rather than criterion 1 (see answer (a) above).

5. <u>Question</u>

Supply the core radial power distribution at different times during the cycle. The limiting power distribution should be included.

<u>Answer</u>

The LOCA limit is set by maintaining F_Q at or below the 2.32 envelope of the Technical Specification 3.2 and as stated in the Reload Safety Evaluation Section 3.1.

It should be emphasized that this envelope is a conservative representation of the bounding values of local power density.

Expected values are considerably smaller and, in fact, less conservative bounding values may be justified with additional analysis or surveillance requirements. For example, both Beginning of Life and End of Life conditions are used in the analysis without consideration of radial power distribution flattening with burnup, i.e., both Beginning of Life and End of Life points presume the same radial peaking factor. Inclusion of the burnup flattening effect would reduce the local power densities corresponding to End of Life conditions which may be limiting at higher core elevations. The most limiting 2D radial power distribution in the cycle exists at Beginning of Life.

6. Question

Describe the procedure used to change the status of valves 862A, B and 864A, B following a LOCA.

Answer

Emergency Operating Procedure 20003 (LOSS OF REACTOR COOLANT) contains instructions on changing the status of valves 862A & B and 864A & B. The procedure specifies that, when the Refueling Water Storage Tank (RWST) low level alarm occurs, the number of operating pumps is reduced to no more than 2 Safety Injection (SI) pumps, 1 Residual Heat Removal (RHR) pump, and 1 Containment Spray (CS) pump. Then the Motor Control Center (MCC) breakers for Motor Operated Valves (MOV) 862A & B and 864A & B are unlocked and closed. When the RWST low-low level alarm occurs, the operating SI, RHR, and CS pumps are stopped. Then the RWST for the affected Unit is isolated by closing MOV's 862A & B and 864A & B.

7. Question

Please identify any differences in fuel rod or fuel bundle design between the fuel assemblies being added for Cycle 3 and the fuel assemblies previously irradiated in Turkey Point 4 during Cycle 2.

Answer

Region 5 fuel is the same as Region 4 fuel rod design except for the enrichment and initial prepressurization level. There are no significant differences in fuel bundle design. (Refer to Section 2.1 of the Reload Safety Evaluation.)

Very truly yours,

for [signature]
Robert E. Uhrig
Vice President

REU/jn

cc: Jack R. Newman, Esq.