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FLORIDA POWER & LIGHT COMPANY

March 5, 1980
L-80-74

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

Dear Mr. Eisenhut:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251

This letter provides additional information in support of Florida Power & Light Company's request for an amendment to Operating Licenses DPR-31 and DPR-41 dated February 13, 1980 (L-51-80).

After correcting input errors reported in Reportable Occurrence 250-79-33 (Letter PRN-LI-79-414 of November 15, 1979) and correcting a potential nonconservatism reported in Reportable Occurrence 250-79-33 (Letter PRN-LI-79-423 of November 23, 1979), our NSSS vendor, Westinghouse, performed an ECCS analysis for Turkey Point Units 3 and 4 with the February 1978 Appendix K evaluation model for the worst break (C_D=0.4), assuming a steam generator tube plugging level of 22%. This ECCS analysis yielded an allowable F_q of 1.89.

A second ECCS analysis for the same conditions but with the removal of a 65°F fuel temperature conservatism led to an F_q of 1.99, thus increasing the allowable F_q limit by 0.10. This analysis was transmitted to you with our letter L-51-80 dated February 13, 1980. We understand that this particular model change is still under review and will be acted on shortly.

In the attachment and in our letter L-80-34 of January 23, 1980, it is shown that for Turkey Point the F_q penalty associated with a revised clad burst and flow blockage model is 0.05. However, there is an F_q credit of 0.15 available for technical changes derived from upper head injection investigations.

To summarize, the allowable F_q limit for Turkey Point with $\leq$22% tube plugging is as follows:

A039
S 3/3

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February 1978 Appendix K model	1.89
Clad burst and flow blockage penalty	-0.05
Upper head injection changes credit	+0.15
Allowable F_q limit	1.99
Credit for 65°F fuel temperature conservatism	+0.10
Allowable F_q limit	2.09

We request, on the basis of these analyses, that you approve our submittal of February 13, 1980 for a change in Technical Specifications to an F_q limit of 1.99. When your review of the removal of the 65°F fuel temperature conservatism has been completed, the F_q limit for Turkey Point with 22% tube plugging should be increased to 2.09.

Very truly yours,

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/GDW/ah

Attachment

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

The Nuclear Regulatory Commission (NRC) issued a letter dated November 9, 1979 to operators of light water reactors regarding fuel rod models used in Loss of Coolant Accident (LOCA) ECCS evaluation models. That letter describes a meeting called by the NRC on November 1, 1979 to present draft report NUREG 0630, "Cladding Swelling and Rupture Models for LOCA Analysis." At the meeting, representatives of NSSS vendors and fuel suppliers were asked to show how plants licensed using their LOCA/ECCS evaluation model continued to conform to 10 CFR Part 50-46 in view of the new fuel rod models presented in draft NUREG 0630. Westinghouse representatives presented information on the fuel rod models used in analyses for plants licensed with the Westinghouse ECCS evaluation model and discussed the potential impact of fuel rod model changes on results of those analyses. That information was formally documented in letter NS-TMA-2147, dated November 2, 1979, and formed the basis for the Westinghouse conclusion that the information was presented in draft NUREG 0630 did not constitute a safety problem for Westinghouse plants and that all plants conformed with NRC regulations. In the November 9, 1979 letter, the NRC requested that operators of light water reactors provide, within sixty (60) days, information which will enable the staff to determine, in light of the fuel rod model concerns, whether or not further action is necessary.

As a result of compiling information for letter NS-TMA-2147, Westinghouse recognized a potential discrepancy in the calculation of fuel rod burst for cases having clad heatup rates (prior to rupture) significantly lower than 25 degrees F per second. This issue was reported to the NRC staff, by telephone, on November 9, 1979, and although independent of the NRC fuel rod model concern, the combined effect of this issue and the effect of the NRC fuel rod models had to be studied. Details of the work done on this issue were presented to the NRC on November 13, 1969 and documented in letter NS-TMA-2163 dated November 16, 1979. That work included development of a procedure to determine the clad heatup rate prior to burst and a reevaluation of operating Westinghouse plants with consideration of a modified Westinghouse fuel rod burst model. As part of this reevaluation, the Westinghouse position on NUREG-0630 was reviewed and it was still concluded that the information presented in draft NUREG-0630 did not constitute a safety problem for plants licensed with the Westinghouse ECCS evaluation model.

On December 6, 1979, NRC and Westinghouse personnel discussed the information thus far presented. At the conclusion of that discussion, the NRC staff requested Westinghouse to provide further detail on the potential impact of modifications to each of the fuel rod models used in the LOCA analysis and to outline analytical model improvements in other parts of the analysis and the potential benefit associated with those improvements. This additional information was compiled from various LOCA analysis results and documented in letter NS-TMA-2174 dated December 7, 1979.

Another meeting was held in Bethesda on December 20, 1979 where NRC and Westinghouse personnel established: 1) The currently accepted procedure for assessing the potential impact on LOCA analysis results of using the

fuel rod models presente[illegible]in draft NUREG-0630 and 2) Acc[illegible]able benefits resulting from analytical model improvements that would justify continued plant operation for the interim until differences between the fuel rod models of concern are resolved.

Part of the Westinghouse effort provided to assist in the resolution of these LOCA fuel rod model differences is documented in letter NS-TMA-2175, dated December 10, 1979, which contains Westinghouse comments on draft NUREG-0630. As stated in that letter, Westinghouse believes the current Westinghouse models to be conservative and to be in compliance with Appendix K.

A. Evaluation of the potential impact of using fuel rod models presented in draft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for Turkey Point Units 3&4 with 22% SGTP, 3% reduced TDF and 65°F conservatism removed.
This evaluation is based on the limiting break LOCA analysis identified as follows:

BREAK TYPE - DOUBLE ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT CD=0.4

WESTINGHOUSE ECCS EVALUATION MODEL VERSION February, 1978

CORE PEAKING FACTOR 1.99

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 2160 °F = PCT_B

ELEVATION - 6.0 Feet.

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 1983 °F = PCT_N

ELEVATION - 7.75 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION 4.29 Percent
MAXIMUM CLAD STRAIN AT THIS ELEVATION - 8.92 Percent

Maximum temperature for this non-burst node occurs when the core reflood rate is greater than 1.0 inch per second and reflood heat transfer is based on the FLECHT calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - N/A Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - 0.0 Percent

1. BURST NODE

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in terms of the change in the peaking factor limit (FQ) required to maintain a peak cald temperature (PCT) of 2200°F and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above 2200°F, individual effects (such as ΔPCT due to changes in several fuel rod models) indicated here may not accurately apply over large ranges,

but a simultaneous change in FQ which causes the PCT to remain in the neighborhood of 2200.°F justifies use of this evaluation procedure.

From NS-TMA-2174:
For the Burst Node of the clad:

- 0.01 $\Delta FQ \rightarrow \sim$ 150°F BURST NODE ΔPCT
- Use of the NRC burst model and the revised Westinghouse burst model of 0.027
- The maximum estimated impact of using the NRC strain model is a required FQ reduction of 0.03.

Therefore, the maximum penalty for the Hot Rod burst node is:

$$\Delta PCT_1 = (.027 + .03)\ (150°F/.01) = 855°F$$

Margin to the 2200.°F limit is:

$$\Delta PCT_2 = 2200.°F - PCT_B = \underline{40°F}$$

The FQ reduction required to maintain the 2200°F clad temperature limit is:

$$\Delta FQ_B = (\Delta PCT_1 - \Delta PCT_2)\ \left(\frac{.01\ \Delta FQ}{150°F}\right)$$

$$= (\underline{855} - \underline{40})\ \left(\frac{.01}{150}\right)$$

$$= \underline{0.054}\ \text{(but not less than zero).}$$

2. <u>NON-BURST NODE</u>

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspects of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which burst is calculated. Three sets of LOCA analysis results were studied to established an acceptable sensitivity to apply generically in this evalution. The possible PCT increase resulting from a change in strain (in the Hot Rod) is +20.°F per percent decrease in strain at the maximum clad temperature

locations. Since the clad strain calculated during the reactor coolant system blowdown phase of the accident is not changed by the use of NRC fuel rod models, the maximum decrease in clad strain that must be considered here is the difference between the "maximum clad strain" and the "clad strain at the end of RCS blowdown" indicated above.

Therefore:

$$\Delta PCT_3 = \left(\frac{20^oF}{.01 \text{ strain}}\right) (\text{MAX STRAIN} - \text{BLOWDOWN STRAIN})$$

$$= \left(\frac{20}{.01}\right) (\underline{0.08}92-\underline{0.04}29)$$

$$= \underline{\quad 92.6 \quad}$$

The second aspect of the analysis that can increase PCT is the flow blockage calculated. Since the greatest value of blockage indicated by the NRC blockage model is 75 percent, the maximum PCT increase can be estimated by assuming that the current level of blockage in the analysis (indicated above) is raised to 75 percent and then applying an appropriate sensitivity formula shown in NS-TMA-2174.

Therefore,

$$\Delta PCT_4 = 1.25^oF\ (50 - \text{PERCENT CURRENT BLOCKAGE}) + 2.36^oF\ (75-50)$$

$$= 1.25\ (50 - \underline{0.0}\ ) + 2.36\ (75-50)$$

$$= \underline{121.5}\ ^oF$$

If PCT_N occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 92.6 + 0 = 92.6°F$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200^oF - PCT_N = 217\ °F$$

The FQ reduction required to maintain this 2200°F clad temperature limit is (from NS-TMA-2174)

$$\Delta FQ_N = (\Delta PCT_5 - \Delta PCT_6)\ \left(\frac{.01\Delta FQ}{10^oF\ \Delta PCT}\right) = -.1244$$

ΔFQ_N = ___.0___ but not less than zero.

The peaking factor reduction required to maintain the 2200 °F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_H,

or; $\Delta FQ_{PENALTY}$ = 0.054

B. The effect on LOCA analysis results of using improved analytical and modeling techniques (which are currently approved for use in the Upper Head Injection plant LOCA analyses) in the reactor coolant system blowdown calculation (SATAN computer code) has been quantified via an analysis which has recently been submitted to the NRC for review. Recognizing that review of that analysis is not yet complete and that the benefits associated with those model improvements can change for other plant designs, the NRC has established a credit that is acceptable for this interim period to help offset penalties resulting from application of the NRC fuel rod models. That credit for two, three and four loop plants is an increase in the LOCA peaking factor limit of 0.12, 0.15 and 0.20 respectively.

C. The peaking factor limit adjustment required to justify plant operation for this interim period is determined as the appropriate ΔFQ credit identified in section (B) above, minus the $\Delta FQ_{PENALTY}$ calculated in section (A) above (but not greater than zero).

FQ ADJUSTMENT = 0.15 - 0.054