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SUBJECT: FORWARDS RESPONSES TO RELOAD SAFETY EVALUATION QUESTIONS.

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FLORIDA POWER & LIGHT COMPANY

May 23, 1979
L-79-136

Office of Nuclear Reactor Regulation
Attention: Mr. R. W. Reid, Director
Operating Reactors Branch #4
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Reid:

Re: St. Lucie Unit #1
Docket No. 50-335
RSE Questions

Attached are responses to several questions generated by the NRC staff relative to the St. Lucie Unit 1 Cycle 3 Reload Safety Evaluation (RSE). This submittal (Questions 2.15, 2.18, and 2.21 through 2.28) completes our response to all questions in this area received to date from the NRC.

Please note that a revised response to Question 1.11 is also attached. Our earlier response to this question inadvertently omitted information we intended to transmit.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/DKJ/cph

Attachment

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

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Question 1.11

In the response to Question 1.6 you state that the implied peaking factors F_r and F_{xy} are higher than the Tech Spec values by the uncertainty in these factors. If this is the case, then the Safety Analysis value should be greater than the Tech Spec value by this uncertainty. Explain why the Tech Spec and Safety Analysis values are equal in your reload application.

Response 1.11

The measurement uncertainties are accounted for in the safety analyses.

The measurement uncertainty factor for F_r is listed in Table 6-1 of the Reload Application (Statistical Component of F_r^N @ 95/95 Confidence Level) along with other calculational factors which are also used in the safety analyses. These factors are not repeated in the tables presented in Section 7.0 of the Reload Application, since these tables only list core parameters and initial conditions for the various transients, consistent with the format used in the Reference Cycle Application and the FSAR. Even though Table 6-1 lists the uncertainty factor on F_r as 1.0513 (CE calculated value), a 1.06 uncertainty factor on F_r was conservatively used along with the other factors listed in Table 6-1.

As indicated in Table 7-2, the initial peak linear heat rate assumed for NON-LOCA safety analyses is 16.KW/FT. The peak linear heat rate for LOCA is 14.8 KW/FT (Chapter 8, page 67). In order to account for the uncertainties on F_q, the Tech Spec ex-core monitoring band on ASI has been based on the LOCA peak linear heat rate limit of 14.8 KW/FT and also incorporates the 7% uncertainty on F_q as described in Section 9.1 of CENPD-199. When monitoring on In-core Instrumentation, the F_q uncertainties are applied as indicated in Section 4.2.1.4 of the Tech Spec. Because the initial peak linear heat rate used in the safety analyses is higher and therefore more deleterious than allowed per Tech Spec by at least the uncertainty factor, the uncertainties on F_q are accounted for in the safety analyses.

Question 2.15

The Cycle 2 parameters were supposedly computed with fine mesh 2D PDQ. How were you able to compute different F_r and F_{xy} with only a two dimensional model?

Response 2.15

Our estimates of the integrated radial peaking factors are obtained by weighting planar radial peaking factors calculated with fine mesh 2-D PDQ for different planar regions of the core. The values used in the safety analyses are always higher than the estimated integrated radials.

Question 2.18

(Cycle 3 Application Page 36) The steady state LHR to fuel centerline melt is given as 21.0 kw/ft. Is this a best estimate value, or is it conservative relative to the best estimate. If it is conservative, what is the best estimate value?

Response 2.18

The power given for fuel centerline melt of 21.0 kw/ft is conservative.

Power to fuel centerline melt is calculated by the FATES fuel performance code as described in CENPD-139-P-A. The FATES code was developed for the purpose of providing conservatively high fuel temperatures for a given power level for use in safety calculations, where conservatism is essential. Best estimate values of power to melt are not required and therefore they are not addressed.

Question 2.21 (Cycle 3 Application Pages 34 and 42)

On page 34, the TAB for CEA Withdrawal Event is 52 psia and on page 42 the TAB for RCS Depressurization Event is 30 psia. What are the previously calculated values for these two parameters?

Response 2.21

The pressure bias factors of 52 psia for CEA Withdrawal event and 30 psia for the RCS Depressurization event were calculated for Cycle 2. Since the values are conservative for Cycle 3, these values were unchanged from Cycle 2 to Cycle 3.

Question 2.22 (Cycle 3 Application Page 34)

For the CEA Withdrawal Event it is indicated that the approach of the LHR to KW/FT limit is considered. Is this a consideration for both the HFP and HZP cases? What criteria are used to determine that the LHR is acceptable? State the core parameter values used to determine that the LHR SAFDL (No centerline melt) is not violated.

Response 2.22

For the CEA Withdrawal event, the approach to the fuel centerline melt was considered for both the HFP and the HZP cases. In Table 7.2 of the licensing submittal, 21 KW/FT was conservatively established as the steady state linear heat rate for centerline melt. For the HFP case, the event is terminated by the Variable High Power trip or the local power density trip prior to reaching 21 KW/FT. For the HZP case, since reactor power rises rapidly for a very short period of time before the power transient is terminated, the integral of the heat generation rate is the controlling parameter. Therefore, for the HZP case the total energy generated and the corresponding temperature rise at the hottest spot of the core was calculated for the duration of the transient to demonstrate that the fuel centerline melt was not violated. Core parameter values used for the CEA Withdrawal event are listed in Table 7.2 of the Cycle 3 license submittal.

Question 2.23 (Cycle 3 Application Page 16)

Here the predicted boron worthsare given as 90 PPM/% $\Delta\rho$ for BOC and 80 PPM/% $\Delta\rho$ for EOC. In Cycle 2 these items were 88 and 77 respectively. Does the boron worth play a significant role in the Safety Analyses. (For example, the Steam Line Break) If so, what values of boron worth were used in the safety analyses for Cycles 2 and 3?

Response 2.23

The inverse boron worth has no impact on the design basis events (DBE's) in the safety analysis, except for steam line break and boron dilution. For the steam line break event, EOC conditions produce the limiting case. For both Cycle 2 and Cycle 3, a high EOC inverse boron worth of 82 ppm/%$\Delta\rho$ was assumed. This is conservative in comparison to the EOC values of 80 ppm/%$\Delta\rho$ for Cycle 3 and the 77 ppm/%$\Delta\rho$ for Cycle 2. For boron dilution it is conservative to use the minimum inverse boron worth. Since those values assumed for the Cycle 2 analysis (Table 7.2-1 of Cycle 2 submittal) bound those values for Cycle 3, the boron dilution event was not reanalyzed.

Question 2.24 (Cycle 3 Application Page 55)

Here the HFP F_z is given as 1.39 and a HZP F_z is given as 1.59. Are these best estimates or conservative values? If they are conservative relative to the best estimates, what are the best estimates values?

Response 2.24

These are conservative values assumed in the safety analysis for the CEA Ejection event and they are based on axial power shape at the limiting I_p allowed by the LCO's. Only at full power can a best estimate F_z be determined, since one can define the operational mode. The best estimate for F_z at full power, ARO, is 1.12. The actual F_z at HZP is a function of many variables such as the mode of operation, time at zero power, past operating history and xenon distribution. Therefore, it is not possible to quote one single best estimate value. However, the F_z will always be less than 1.59.

Question 2.25 (Cycle 3 Application, Page 16)

Here the neutron generation time ℓ^* is given as 33xE-6 sec at EOC and 28XE-6 sec at BOC. ℓ^* should play a role in the CEA Ejection Event, but it does not appear in the list of parameters for this event. What value of ℓ^* was used in the Safety Analysis?

Response 2.25

Based on parametric studies where ℓ^* was varied over the range of BOC and EOC values, there is a 0.05 cal/gm difference in total energy deposited for the CEA Ejection event. For Cycle 3 the nominal value of 30×10^{-6} sec was assumed in the safety analysis.

Question 2.26 (Cycle 3 Application Page 55)

Here a HZP Azimuthal Power Tilt of 1.10 is used in the Safety Analysis. What is the best estimate of the HZP Azimuthal Power Tilt?

Response 2.26

Our best estimate HZP azimuthal power tilt is 1.03 as compared to 1.10 used in the safety analyses.

Question 2.27 (Cycle 3 Application Page 33)

The Three Pump Plenum Factor used in the Safety Analysis is 1.09. What is the best estimate value for this parameter?

Response 2.27

The use of the plenum factors in the thermal-hydraulic code, COSMO, is discussed on Page 7-12 of CENPD-161-P and in Section 3.2.4.3 of CENPD-199. The 1.09 three pump plenum factor is a conservative value used in the safety calculations where conservatism is essential. Best estimates are not required and therefore not addressed.

Question 2.28

The malfunction of one-steam generator events are not re-analyzed in either Cycle 2 or Cycle 3 Reload Application. Thus, the only analysis must be in the FSAR. We are unable to find the malfunction of one-steam generator events in the FSAR Chapter 15 Table of Contents. Indicate where these events are discussed in the FSAR.

Response 2.28

The Asymmetric Steam Generator transients (Loss of Load to One Steam Generator, Excess Load to One Steam Generator, Loss of Feedwater and Excess Feedwater to One Steam Generator) were not in the set of design basis events analyzed for the FSAR. Subsequent to completing the FSAR, but prior to generating the Cycle 1 setpoints, these events were incorporated into the list of the Design Basis Events as documented in CENPD-199-P. Analyses were done at that time to demonstrate that the required overpower margins for these events were less than for other AOO's, such as the Loss of Flow. The input parameters for this original analysis bound those for Cycle 3 as indicated in Table 1. This is the basis for the statement in the Cycle 3 license submittal that these events were not reanalyzed.

TABLE 1

Key Parameters Assumed in the Analysis
of the Malfunction of One Steam Generator

Parameter	Units	Cycle 1	Cycle 3
Initial Core Power	MW_t	2621	2621
Initial Core Inlet Temperature	$^\circ F$	544	544
Initial Reactor Coolant System Pressure	psia	2200	2200
Moderator Temperature Coefficient	$\times 10^{-4} \Delta\rho/^\circ F$	-2.5	-2.5
Doppler Coefficient Multiplier		0.85	0.85