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SUBJECT: FORWARDS ADDL INFO CYCLE 3 RELOAD APPLICATION RE ANTICIPATED BURNUP FOR CLAD COLLAPSE FOR EACH FUEL BATCH,COMBINING OF FUEL PIN POWERS & AUGMENTATION FACTORS & INVESTIGATION OF CONTROL ELEMENT ASSEMBLY STICKING.

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FLORIDA POWER & LIGHT COMPANY

April 30, 1979
L-79-106

Office of Nuclear Reactor Regulation
Attention: Mr. R. W. Reid, Chief
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
<u>Cycle 3 RSE Information Request</u>

In early April we received a telecopied request for additional information concerning the Cycle 3 reload application. Our response to the information request is attached.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/MAS/ms

Attachment

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

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Question 1.1

Reload Application Page 8 (RAP-8). Provide the anticipated burnup for clad collapse for each fuel batch and the maximum anticipated burnup during Cycle 3 for each fuel batch.

Response 1.1

Batch	Burnup for Clad Collapse* (EFPH)	Maximum Anticipated Burnup During Cycle 3 (EFPH)
B	≥ 24,600	23,690
C	≥ 24,600	23,690
D	≥ 25,200	14,170
E	≥ 26,900	7,715

* Conservatively calculated. It is expected that more detailed calculation will yield higher exposure for prediction of clad creep collapse.

Question 1.2

Explain in detail the mechanics (including equations) of the statistical combining of fuel pin powers and augmentation factors.

Response 1.2

C-E has an NRC approved procedure for calculations power peaking augmentation factors caused by fuel densification.[1] In C-E's topical report on densifying fuel, single gap power peaking factors are defined by calculations using the geometry of Figure 1. The single gap peaking factors, from the five rings identified in Figure 1, are statistically convoluted with the flattest pin census calculated during the cycle and the densification characteristics of the fuel in the core to generate power peaking expectation values. For this purpose, the flattest pin census is identified as that pin census having the most pins in the .05 power peaking interval with its upper limit at the maximum power peak for the pin census. The densification characteristics of importance are the maximum densification the fuel can undergo, expressed as a fraction of the theoretical density of the fuel, and the clad creep coefficient. Together, these parameters define the size distribution of the gaps. The augmentation factor is then defined as the ratio of the power peaking at an expectation value of .355 to the maximum power peak of the pin census.

Cycle 3 of St. Lucie Unit 1 is being loaded with 21 B assemblies, 68 C assemblies, 60 D assemblies and 68 E assemblies. The fuel in the C, D and E assemblies is "stable fuel" which could undergo densification up to 1% of the theoretical density. Only the B fuel is "unstable fuel", capable of undergoing densification up to 3.5% of theoretical density.

The calculated peak power pin in the high power assemblies, including the assembly containing the core peak power pin, is always immediately adjacent to a CEA guide tube. Therefore, the peak pins to which power peaking augmentation factors must be applied are at least one full row of fuel pins away from an assembly boundary. Design calculations have also shown that the calculated radial power peaking in B fuel is always at least 15% lower than that calculated for the peak pin. Thus, applying a power peaking augmentation factor for densifying fuel, calculated by C-E's NRC approved procedure, to the peak pins which occur in nondensifying fuel assemblies would be a gross exaggeration of its impact.

Because of the relative position of the peak power pin and of the pins which can undergo "unstable fuel" densification, the impact of unstable fuel densification would affect only the augmented power peaking due to some of the fuel pins in rings 3, 4 and 5. Stable fuel densification would be applicable to the pins in rings 1 and 2 and the balance of the pins in rings 3, 4 and 5.

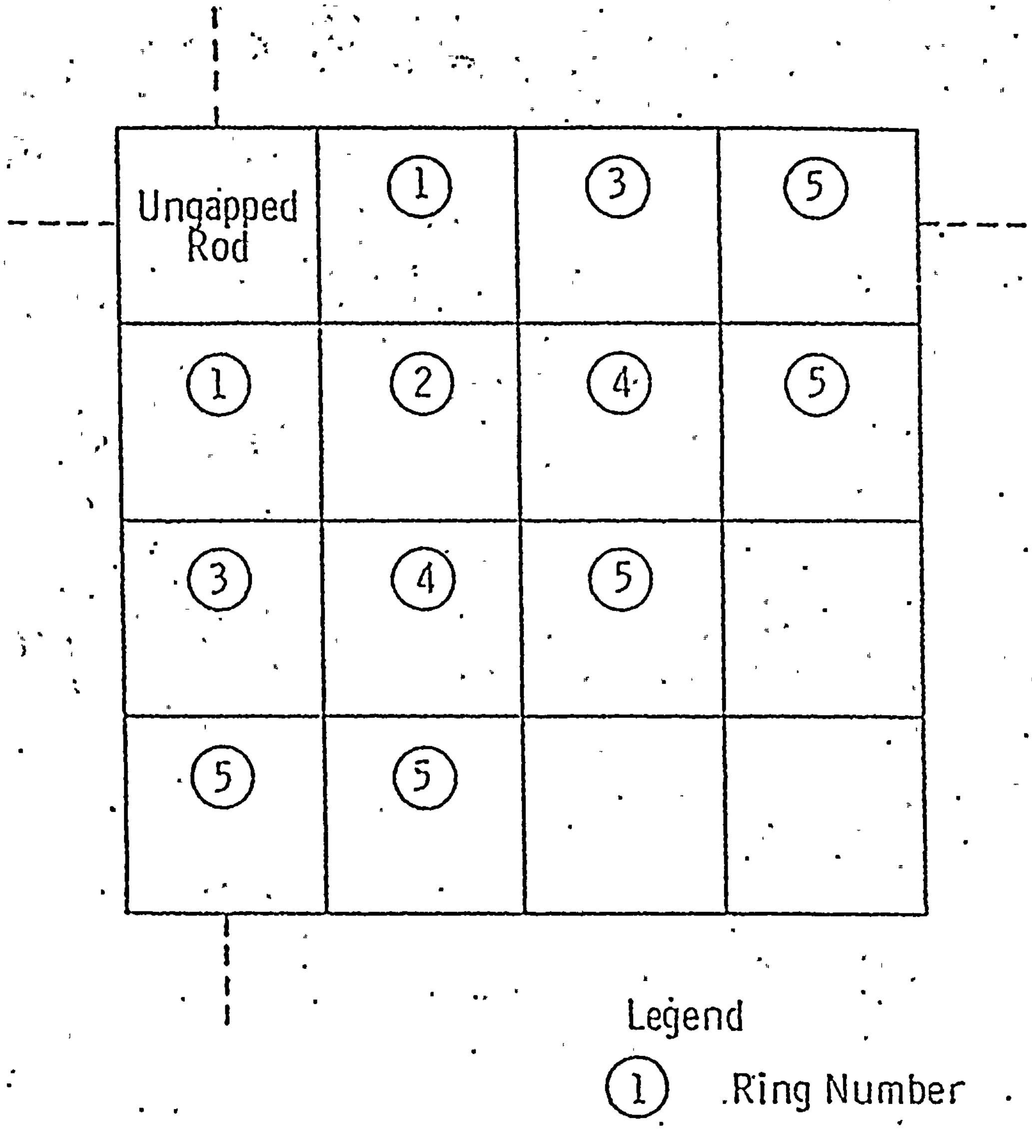
Because of the complexity of the geometry, we only calculate expectation values based upon single gap peaking factors for groups of whole rings. Accordingly, we calculated the power peaking expectation values for rings 1 and 3 for nondensifying fuel and for rings 3, 4 and 5 for densifying fuel for each of 10 axial regions of the core. The two sets of expectation values for each core height were then added together. The power peak at the expectation value of 0.355 was determined for each core height. The ratio of each of these augmented power peaks to the pin census power peak was then calculated. These ratios are the augmentation factors given in the Cycle 3 license submittal.

References

(1) "Fuel Evaluation Model", CENPD-139-P-A, July, 1974

FIGURE 1

Single Gap Peaking Factor Array And
Rod Location Assignment to Radial Groups
For a Typical 14 x 14 Assembly Core



Question 1.3

(Section 5.2.3.1 ROCS) At this time, ROCS is accepted for scoping calculations but not for safety calculations. It is not clear from Section 5.2.3.1 which calculations are performed with ROCS and which calculations are performed with fine mesh PDQ. Please supply a list of all (if any) safety related calculations that were performed with ROCS.

Response 1.3

The following parameters were calculated with the ROCS computer code:

- -Fuel Temperature Coefficients
- -Moderator Temperature Coefficients
- -Inverse Boron Worths
- -Critical Boron Concentrations
- -CEA Drop Distortion Factors and Reactivity Worths
- -Reactivity Scram Worths and Allowances
- -Reactivity Worth of Regulating CEA Banks

None of these parameters require detailed knowledge of pin peaking factors and in most cases are calculated more accurately by ROCS because of its ability to account for 3-D effects.

Question 1.4

RAP-25. Here a 3% credit is identified for the DNBR LCO and DNBR LSSS. Where is this credit taken?

Response 1.4

The 3% credit is a partial credit for a statistical combination of measurement uncertainties which is applied directly to increase the Tech Spec value of F_r^T. (see Reference 1).

Reference 1

Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 18 to Facility Operating License No. DPR-69 Baltimore Gas and Electric Company Calvert Cliffs Nuclear Power Plant, Unit No. 2 Docket No. 50-318", October 21, 1978.

Qestion 1.5

RAP-33 and Tech Spec page 3/4 2-9. The Safety Analysis assumes Fr = 1.59. The Tech Spec assumes Fr = 1.64. Explain the 3% difference in those two numbers. Explain how the 3% is taken into account in Table 7-2, and if it is not taken into account in Table 7-2; explain why not.

Response 1.5

The Tech Spec assumes an Fr^T of 1.64 which, by definition is the Total Integrated Radial Peaking Factor and includes an azimuthal tilt allowance T_z of 0.03. The value of 1.59 in Table 7-2 represents the Integrated Radial Peaking factor Fr without the tilt allowance. These two radials are related as follows:

$$Fr^T = Fr\ (1 + T_z)$$

Although not listed in Table 7-2, the aximuthal tilt allowance is factored into the Safety Analyses.

Question 1.6

Explain how the measurement uncertainty is taken into account in monitoring Fr and Fxy, i.e. is the uncertainty added to the measured value and the result compared with the Tech Spec value, or is the uncertainty already combined in the Tech Spec value so that the measured value is compared directly with the Tech Spec value. If the uncertainty is added to the measured value is this done automatically by INCA, or is it done by hand?

Response 1.6

The measurement uncertainties on Fr and Fxy are directly factored in the LCO and LSSS limits, such that the implied radial peaking factors are actually higher than the Tech Spec values, by an amount corresponding to the uncertainties. Therefore, since the uncertainties are pre-factored in these limits, the as-measured radial peakings can be compared directly to the Tech Spec values.

Question 1.7

Tech Spec Page 3/4-2-9. If the radial peaking has increased, logically the TM-LP setpoint should change. Why has the TM-LP setpoint not changed?

Response 1.7

The increase in the radial peaking was accommodated without requiring a change in the TM/LP setpoints for Cycle 3 because:

(a) We took a partial credit of 3% for statistically combining various measurement uncertainties as discussed in section 6.1 of the License Submittal and in Reference 1,

(b) There are less shims in Cycle 3. This increases the core heat transfer surface area and increases the margin to DNB (~2%) and,

(c) There were some conservatisms in the Cycle 2 TM/LP setpoints.

Reference 1
NRC, "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amend. No. 18 to Facility Operating License No. DPR-69 Baltimore Gas and Electric Company Calvert Cliffs Nuclear Power Plant, Unit No. 2 Docket No. 50-318", October 21, 1978.

Question 1.8

Tech Spec Pages 3/4 2-4, 3/4 2-6, 3/4 2-9 and 3/4 2-15. Explain the reduction of these two ASI LCO Tents from the Cycle 2 Tents. The increase in allowed Fr and Fxy combined with the reduced tents should provide initial conditions for AOO's no more deleterious than those provided by the Cycle 2 Tech Specs. This is necessary because you have not reanalyzed most of your AOO's. Demonstrate (with numerical examples wherever possible) that the Cycle 3 Tech Specs do, in fact, provide initial conditions no more deleterious than those provided by the Cycle 2 Tech Specs for all AOO's not reanalyzed.

Response 1.8

The reduction in the DNBR-LCO ASI Tent was not at the 100% power band but at lower power where the band was reduced for consistency with the LHR-LSSS. The increase in the integrated radial peaking was accomodated without change to the DNB-LCO ASI Tent for the same reasons given in answering question 1.7.

The reduction in the LHR-LCO and LSSS ASI Tents was to allow for more deleterious axial shapes at BOC for Cycle 3, an increase in azimuthal tilt allowance, and an increase in Fxy. These reductions in the ASI tents provide initial conditions for AOO's no more deleterious than those provided by the Cycle 2 Tech Specs.

Question 1.9

REPORTABLE OCCURRENCE 335-79-1 STUCK CEA:

The cause of the stuck CEA's should be investigated during the refueling outage. Indicate what investigations are planned to determine the cause of the stuck CEA.

The question of whether the sticking is related to a generic cause, such as guide tube wear, sleeving, or loose parts, should be addressed.

Response 1.9

The following refueling outage examinations have been performed:

(1). The 2 CEAs were withdrawn and reinserted twice with no indication of binding.

(2) The CEAs were thoroughly examined visually during the withdrawal/insertion operation. There was no indication of abnormal stress, wear, or binding.

(3) The withdrawal/insertion operation was performed with a load cell installed on each CEA. The load cell readings were normal.

(4) The top of each fuel assembly was visually examined. There was no indication of abnormal wear or binding.

(5) The top end of each guide tube sleeve was visually examined. Each sleeve was in place and there were no indications of deformation or damage.

The examinations conducted during the refueling outage have given no indication that Reportable Occurrence 335-79-1 was related to guide tube wear, sleeving, or loose parts. In addition, the CEAs had been verified operable following the occurrence. The CEAs and CEDM 43 were operationally tested satisfactorily. Hot and cold drop time testing was performed, with all drop times well within limits and consistent with previous results.