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RECIP.NAME RECIPIENT AFFILIATION
MILLER,J.R. Operating Reactors Branch 3

SUBJECT: Forwards response to 840830 ltr requesting addl info re facility Cycle 2 reload.Response to Item 13 C-E proprietary info.Response to Item 13 withheld (ref 10CFR2.790).

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FLORIDA POWER & LIGHT COMPANY

September 26, 1984
L-84-262

CONTAINS PROPRIETARY INFORMATION

Office of Nuclear Reactor Regulation
Attention: Mr. James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miller:

Re: St. Lucie Unit 2
Docket No. 50-389
Request for Additional
Information Cycle 2 Reload

Attached is Florida Power & Light Company's response to your letter of August 30, 1984, which contained a request for additional information. The response to item #13 is Combustion Engineering, Inc. proprietary information and, therefore, exempt from public disclosure in accordance with 10 CFR 2.790.

Very truly yours,

J. W. Williams, Jr.
Group Vice President
Nuclear Energy

JWW/CGO/PKG/js

Attachment

cc: J. P. O'Reilly, Region II
Harold F. Reis, Esquire
PNS-LI-84-337

PA01
1/1

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

J. W. Williams, Jr., being duly sworn, deposes and says:

That he is a Group Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

Item No. 13 of the attached is Combustion Engineering, Inc. proprietary information and, therefore, exempt from public disclosure in accordance with Section 2.790 of the NRC "Rules of Practice", Title 10, Code of Federal Regulations.

J. W. Williams, Jr.

Subscribed and sworn to before me this

26 day of September, 1984

NOTARY PUBLIC, in and for the County of Dade, State of Florida.

NOTARY PUBLIC STATE OF FLORIDA
MY COMMISSION EXP. FEB 14,1988
BONDED THRU GENERAL INS. UND.

My commission expires 2-14-88

AFFIDAVIT PURSUANT

TO 10 CFR 2.790

Combustion Engineering, Inc.)
State of Connecticut)
County of Hartford) SS.:

I, A. E. Scherer, depose and say that I am the Director, Nuclear Licensing, of Combustion Engineering, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations and in conjunction with the application of Florida Power and Light Company for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

St. Lucie 2, Cycle 2, Response to NRC Questions, Question #13.

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Combustion Engineering in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the

Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure is the sensitivity to DNBR of the CETOP-D and the Detailed TORC Computer Codes for various conditions of an operating reactor, which is owned and has been held in confidence by Combustion Engineering.

2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in a substantial competitive advantage to Combustion Engineering.

3. The information is of a type customarily held in confidence by Combustion Engineering and not customarily disclosed to the public. Combustion Engineering has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The details of the aforementioned system were provided to the Nuclear Regulatory Commission via letter DP-537 from F.M. Stern to Frank Schroeder dated December 2, 1974. This system was applied in determining that the subject document herein is proprietary.

4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of Combustion Engineering because:

a. A similar product is manufactured and sold by major pressurized water reactor competitors of Combustion Engineering.

b. Development of this information by C-E required hundreds of man-hours and tens of thousands of dollars. To the best of my knowledge and belief a competitor would have to undergo similar expense in generating equivalent information.

c. In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of computer codes with the sensitivity to DNBR of CETOP-D and Detailed TORC.

d. The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e. The information consists of the sensitivity to DNBR of the CETOP-D and the Detailed TORC Computer Codes for various conditions of an operating reactor, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Combustion Engineering, take marketing or other actions to improve their product's position or impair the position of Combustion Engineering's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

f. In pricing Combustion Engineering's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of Combustion Engineering's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

g. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on Combustion Engineering's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

A. E. Scherer
Director
Nuclear Licensing

Sworn to before me
this 19th day of September 1984

Lydia A. Smith
Notary Public
LYDIA A. SMITH, NOTARY PUBLIC
STATE OF CONNECTICUT No. 68542
COMMISSION EXPIRES MARCH 31, 1989

RESPONSE TO REQUEST FOR

ADDITIONAL INFORMATION ON

ST. LUCIE UNIT 2 CYCLE 2 RELOAD

Question 1: Verify that the maximum radial peaking factors expected during Cycle 2 (shown in Figures 2.4.3-2 through 2.4.3-5 without uncertainties) do not exceed the Technical Specifications limiting values or the values used in the safety analyses when uncertainties and other allowances are included.

Answer: The C-E safety analysis remains valid as long as the measured peaking factors are less than or equal to the Technical Specification values. Uncertainties in radial peaking factors are included in the protection and monitoring system setpoints to ensure that applicable safety analysis criteria are not exceeded even when the measured peaking factors equal the Technical Specification limit. The uncertainties on peaking factors employed are detailed in Reference 1-2 (CECOR topical).

There are three equivalent methods which may be used for the treatment of radial peaking factor uncertainties. The first method is to factor the uncertainties directly into the Technical Specification limits, i.e., to define the Technical Specification limit as the value used in the safety analyses less uncertainties. The radial peaking factors calculated with the in-core detector system, which would not be adjusted for uncertainties, would be compared directly to the Technical Specification limit. The peaking factors assumed in the transient analyses with this method are the Technical Specification values plus uncertainties.

The second method is to apply the uncertainties on the measured peaking factor values to compare against the Technical Specification limits which do not include uncertainties. The setpoint and transient analyses would then assume the Technical Specification values.

The third method, which is employed by C-E, is to apply these uncertainties in the setpoint analyses. Therefore, neither the Technical Specification limits, the measured peaking factor values, nor transient analyses have to include these uncertainties. A detailed description of how these uncertainties are applied can be found in Reference 1-1. In this method, trip setpoints and limiting conditions for operation (monitoring system setpoints) are first established so that all applicable safety analysis criteria are met assuming peaking factors at the Technical Specification limit (without uncertainties). Trip setpoints and limiting conditions for operation are then reduced by allowances for power distribution uncertainties, as well as for other uncertainties associated with the monitoring and protection system. This ensures that, considering all applicable uncertainties, initial conditions for all transients will be no more adverse than those assumed in the safety analysis, and that a protection system action will occur no later than that assumed in the safety analysis. The consequence of accidents and transients will then be no more adverse than those presented in the safety analysis.

In the C-E approach, the Technical Specification limits on radial peaking factors ($F^T_{xy} \leq 1.75$ and $F^T_r \leq 1.70$) are set such that they bound the values expected to occur throughout the entire cycle. These "expected" values are based on 3-D ROCS coarse-mesh and 2-D PDQ fine mesh core depletion calculations. Provisions are included in the Technical Specifications to reduce power and/or restrict operating space to restore the margins to safety, should the measured peaking factor exceed the Technical Specification limit. Even though the Technical Specification limits on F^T_{xy} and F^T_r could be set equal to those values determined by the 3-D ROCS and 2-D PDQ synthesis, C-E has always set aside some margin so that the probability of having to run the plant at reduced power, due to exceeded Technical Specification limits on peaking, becomes remote. Through past experience C-E has found that if 0.07 is added to the peak calculated values of F^T_{xy} and F^T_r to determine the Technical Specification limits, enough margin exists to preclude approaching the Technical Specification limits. The peak calculated F^T_{xy} and F^T_r for Cycle 2 were 1.64 and 1.60 respectively, therefore, the Cycle 2 Technical Specification limits on these parameters contain more than 0.07 margin. The extra margin was applied in an attempt to bound future cycles.

References:

1-1 CENPD-199-P, Rev. 1-P, "C-E Setpoint Methodology," March 1982.

1-2 CENPD-153, Rev. 1-P, "INCA/CECOR Power Peaking Uncertainty," May 1980.

Question 2: What are the HFP values of total CEA worth, stuck CEA worth, and CEA bite worth?

Answer: The calculation of the hot full power (HFP), end-of-cycle (EOC), steam line break (SLB) accident scram worth performed for St. Lucie Unit 2 Cycle 2 is shown below. The format is the same as appears for the HZP EOC SLB accident scram worth calculation shown in the St. Lucie Unit 2 Cycle 2 Reload Design Report.

1.	Worth of all CEAs Inserted (%$\Delta\rho$)	12.4
2.	Stuck CEA Allowance (%$\Delta\rho$)	2.6
3.	Worth of all CEAs Less Highest Worth CEA Stuck Out (%$\Delta\rho$)	9.8
4.	Full Power Insertion Limit CEA Bite (%$\Delta\rho$)	0.4
5.	Calculated Scram Worth (%$\Delta\rho$) (Item 3 minus Item 4)	9.4
6.	Physics Biases and Uncertainty (%$\Delta\rho$)	1.6
7.	Net Available Scram Worth (%$\Delta\rho$) (Item 5 minus Item 6)	7.8

Question #3: Explain in more detail the Cycle 2 changes and analysis which now allow a CEA misalignment to exist for up to 63 minutes for an initial $F_R^T \leq 1.55$ compared to only 30 minutes for an initial $F_R^T \leq 1.50$ in Cycle 1.

Answer: As discussed in Section 3.2.4.3 of the Reload Design Report, "CEA Drop Event", the subgroup CEA drop event is the limiting Anticipated Operational Occurrence (AOO) event requiring the maximum initial margin to be maintained by the LCO's. In the Cycle 2 analysis of CEA drop, the uncertainties of several input parameters are combined statistically to reduce the conservatisms inherent in the Cycle 1 deterministic approach.

The single CEA drop was examined for Cycle 2 using conservative values of CEA worth and power peaking increase in order to bound future cycle designs. The effects of post drop Xenon redistribution and the time to reach the subgroup drop limiting radial peaking value was calculated as a function of the predrop F_R^T. The figure that accompanies proposed Technical Specification 3/4.1.3 is a conservative interpretation of this data.

The maximum initial radial peaking factor F_R^T assumed for the single and subgroup CEA drop event is the Technical Specification limit of 1.70. For the CEA subgroup drop event, the maximum increase in F_R^T assumed for Cycle 2 is 19.0%, (Cycle 1 value: 19.4%). Table 2.4.5-1 of the Reload Design Report shows that the comparable increase in F_R^T for the single CEA drop event is 14.0%, (Cycle 1 value: 13.5%). Thus, the F_R^T can increase an additional 5% due to power redistribution following a single dropped CEA and still be conservatively bounded by the results of the subgroup CEA drop. Table 2.4.5-1 shows that after 15 minutes, the net increase in F_R^T for the Cycle 2 analysis (18%) remains below the limiting increase in F_R^T for the subgroup CEA drop (19%). Figure 3.1-1a in the proposed Technical Specification submitted shows a decrease in the allowable pre-drop radial to compensate for the increasing post-drop radial due to xenon redistribution with time (up to 63 minutes) to keep the net increase less than 1.70 plus 19%.

The Cycle 1 analysis used a similar approach to merely verify that after 25 minutes, the net increase in F_R^T is less than 19.4% above 1.60 when the pre-drop F_R^T is less than or equal to 1.50. No attempt was made to optimize the results of the Cycle 1 calculations.

In addition, the low leakage fuel management scheme, (In-Out) employed in Cycle 2 is more stable to power shifts than the Out-In fuel management pattern used in Cycle 1. This increased stability contributed to a slight reduction in the xenon redistribution of effects of a dropped CEA in Cycle 2 when compared to Cycle 1. This allows greater time with a misaligned CEA before exceeding the limiting analysis power redistribution.

Question #4: Since the acceptable minimum DNBR limit is used as a criterion in anticipated operational occurrences and postulated accidents, we request that the actual value (1.28) remain in the Technical Specification bases.

Answer: As reported in section 3.3 of Appendix I of the Reload Safety Report, the minimum DNBR limit is derived through a statistical combination of the system parameter probability distribution functions with the CE-1 critical heat flux (CHF) correlation. These system parameters are particular to the St. Lucie Unit 2 fuel assembly design.

Should economic or safety related improvements to the design be implemented, a re-evaluation of the combination of these parameters with the CHF correlation would be performed. Should this result in a change to the MDNBR limit, this would require a Technical Specification change, if the MDNBR value was included in the Technical Specification bases. However, if a similar design change was implemented using a deterministic MDNBR limit (1.20), this design change would not necessarily require a Technical Specification change, since the changed parameters would not have been included in the MDNBR limit.

The proposed change would avoid additional unnecessary Technical Specification changes that would be required in situations where they would not have been required previously. The methodology (SCU Statistical Combination of Uncertainties - CEN(F)-P 123) used to calculate the MDNBR limit for Cycle 2 has been previously approved by the NRC. Any changes in approved methods are reportable in accordance with 10CFR 50.59.

NRC Question 5

Question 5: Explain the reason for the two-component form of the uncertainties associated with the power distribution and with the ASI calibration in Table 2-1 of Appendix I.

Answer: The apparent two-component forms of the uncertainties associated with the power distribution and with the ASI calibration in Table 2-1 of Appendix I are inadvertently similar to that of the primary coolant pressure uncertainty of the same table. They are actually shorthand statements of the mean and $k_{95/95}$S deviations of single uncertainty distributions and were used in that manner.

NRC Question 6

Question 6: Have the non-LOCA events been reanalyzed with CESEC or with the NRC approved CECSEC III version?

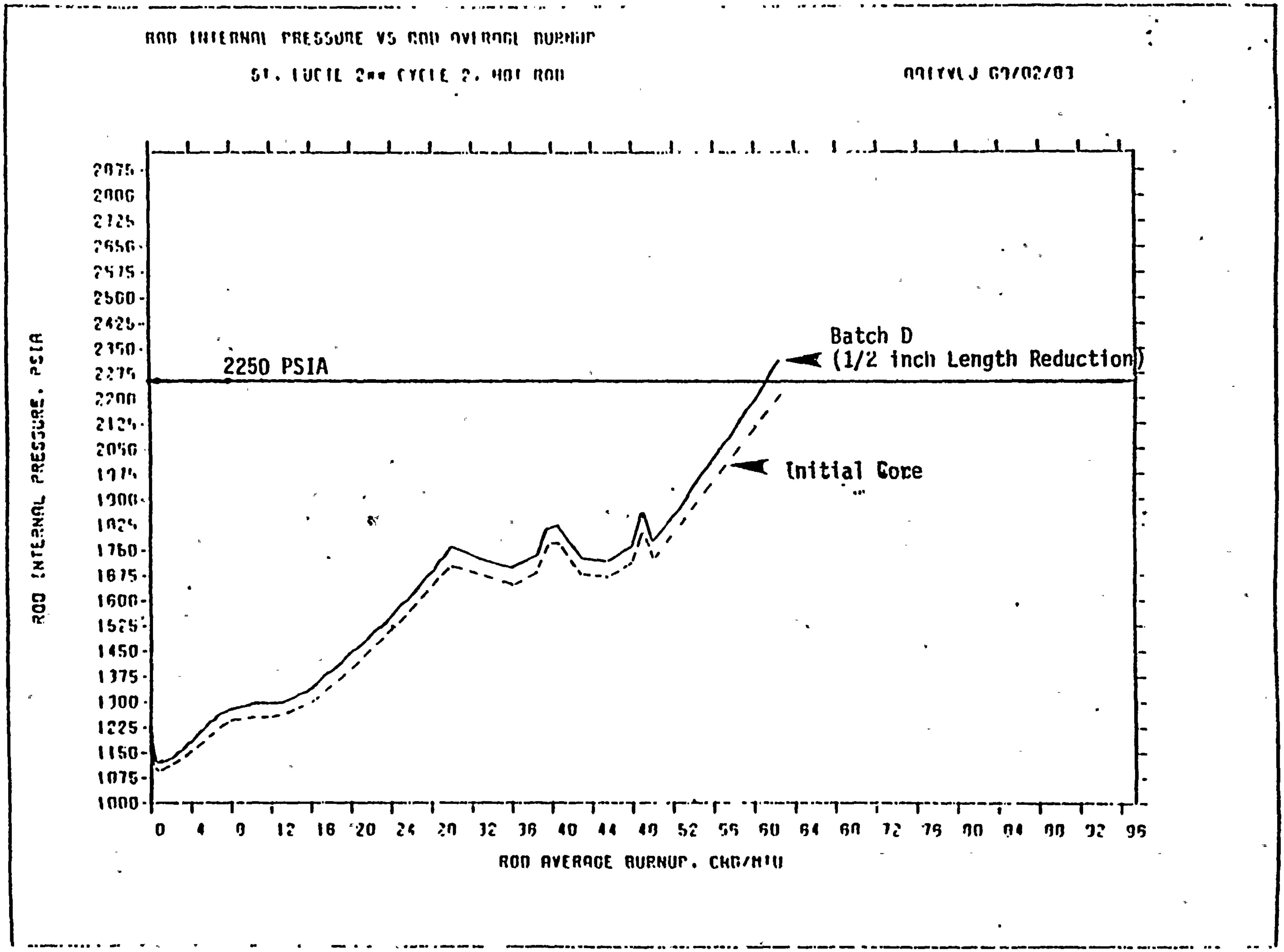
Answer: The non-LOCA events have been reanalyzed with CESEC III which is the NRC approved version of CESEC.

Question #7: Please describe the effect of the shorter plenum length on Batch D fuel rod internal pressure as compared to the SRP criteria.

Answer: Calculations have been performed using the most recent version of the FATES (FATES3) computer code to determine the internal rod pressure versus burnup for the limiting fuel rod. The results of these calculations are shown on the attached figure. These calculations were performed assuming a 500 mil reduction in the overall rod length and no change in the fuel pellet column length. These calculations were also performed using conservatively high radial peaking factors versus pin burnup.

The results of this work indicate that the internal rod pressure will remain below the system pressure (2250 psia) for burnups up to 60,000 MWD/MTU. This calculation was performed assuming a LOCA linear heat rate limit no greater than 13 kw/ft. Because the St. Lucie 2 Batch D rod plenum will be reduced by 300 mils, instead of 500 mils, there is sufficient assurance that the fuel rod internal pressure will not exceed the primary system pressure for all anticipated burnups. Compliance with the criteria discussed above will maintain the capability of the fuel rods to meet existing licensing requirements on maximum rod pressure as covered by Section 4.2 of the Unit 2 FSAR.

Figure 3


ROD INTERNAL PRESSURE VS ROD AVERAGE BURNUP
ST. LUCIE 2** CYCLE 2, HOT ROD
ROD INTERNAL PRESSURE, PSIA
ROD AVERAGE BURNUP
2250 PSIA
Batch D
(1/2 inch Length Reduction)
Initial Core

Question #8: Would the increase in guide tube length shorten the clearance between the upper core plate and the upper end fitting, thereby compressing the spring unnecessarily?

Answer: Among the modifications made to the Batch D fuel bundles, two involve increasing the guide tube length by 400 mils and changing the guide tube material from cold worked zircaloy to fully annealed zircaloy. The increase in annealed guide tube length will result in additional spring compressive forces of approximately 100 pounds per fuel assembly and a larger engagement of the posts within the upper core alignment plate at BOL conditions. The additional spring holdown forces at BOL and the increased post engagement within the upper alignment plate are within the range of the documented initial core parameters with regard to forces, stresses and axial displacements and as such, will have no negative impact on design or functional requirements.

The selection of annealed guide tube material was primarily based on the small axial growth characteristics per unit neutron fluence compared to cold worked material. This limited axial growth for annealed material will minimize the change in spring compressive forces on the fuel assembly with increasing fluence. Therefore, although the longer annealed guide tube may begin operation with a higher spring loading, the lower growth rate for annealed guidetubes should assure that the differences in spring loadings between the two designs will decrease with increasing burnup.

Question #9: The St. Lucie 2 license condition on axial growth states that "Prior to startup following the first refueling outage, the licensee shall provide an analysis and/or make hardware modifications to assure that the shoulder gap clearance between fuel rods and fuel assembly end fittings is adequate." The axial growth for Batches B and C fuel was analyzed using the growth model in CENPD-198. CE has stated that CENPD-198 for the 16 x 16 fuel design in ANO-2 (which is identical in design to the St. Lucie 2 fuel) is non-conservative, but has not yet revised the growth model. Therefore, further justification is required for why Batches B and C fuel can be used for Cycle-2 operation without hardware modifications and/or applicable analysis (other than CENPD-198) as indicated in the license condition.

Answer: The Cycle 2 reload consists of 73 Batch B assemblies and 64 Batch C assemblies. All Batch C assemblies and 16 Batch B assemblies have been shimmed. The 0.45 inch shim has increased the initial shoulder gap clearance from 0.997 in. to 1.447 in. Based on ANO-2 measured shoulder gap closure -(not CENPD-198 growth models)- in conjunction with predicted fluences to evaluate shoulder gap, it is concluded that this increase in shoulder gap of approximately 45% is sufficient to assure at least 95% confidence of adequate shoulder gap clearance during Cycle 2 operation for the shimmed assemblies.

The Cycle 2 reload included 57 Batch B unshimmed assemblies with an initial shoulder gap of 0.997 in. During the Cycle 1-2 outage, verification of an adequate shoulder gap for a second cycle of operation for these assemblies will take place by conducting shoulder gap measurements in conjunction with supporting analysis. Based on ANO-2 measured shoulder gap closure and the predicted fluences for Cycles 1 and 2, shoulder gap closure predictions will be made for these assemblies. EOC-1 measurements of the Batch B assemblies in question will be used to show that the ANO-2 growth correlations are conservative for St. Luice 2 fuel applications. Also, if measurement results warrant, St. Luice Unit 2 specific growth correlations will be developed using the St. Lucie 2 16 x 16 shoulder gap measurements. Those assemblies which fail to show adequate shoulder gap for the Cycle 2 operation will be shimmed at the site.

It should be noted that the ANO-2 16x16 fuel assembly design and the St. Lucie 2 16x16 fuel assembly design are not identical. Although shoulder gap closure predictions for St. Lucie 2 fuel are based on ANO-2 measurements, the pertinent design differences have been conservatively taken into account. The pertinent design differences (shorter active fuel rod length, cold worked guide tubes and a decreased hold down load) should contribute to a smaller shoulder gap closure at St. Lucie 2 compared to ANO-2. The measurement program at St. Lucie 2 is expected to confirm this.

A formal report addressing this question will be submitted to the NRC prior to Cycle 2 startup, as required by the St. Lucie Unit 2 license condition on axial growth.

NRC Question 10

Q: Was a new bias factor for the TM/LP setpoint obtained from the CEA withdrawal analysis or from the inadvertent opening of a PORV analysis? What new value was obtained and how were the Technical Specifications modified to include this value?

A: The limiting bias factor for the TM/LP trip setpoint was obtained from the CEA withdrawal analysis. The value of this bias factor is 70 Psi. This value is included in the " γ " term of the P_{var}^{trip} equation (shown below), and results in an increased P_{var} trip setpoint of 70 Psi. The higher P_{var} trip setpoint results in earlier action of TM/LP trip.

$$P_{var}^{trip} = (\alpha)\ (Q_{DNB}) + (\beta)\ (T_{in}) + (\gamma)$$

The P_{var} equation appears in the Technical Specification in Figures 2.2-3 and 2.2-4 and includes the Cycle 2 bias factor of 70 psi.

NRC Question 11

Q: Do any fuel pins experience DNB during the steam generator tube rupture event? If so how many? Has Tech Spec limit for tube leakage in the unaffected SG been included for the offsite dose calculations?

A: The DNBR SAFDL is not violated during the steam generator tube rupture event; therefore, no fuel pins are predicted to experience DNB.

The Tech. Spec. limit for tube leakage in the unaffected steam generator has been included in the offsite dose calculations.

NRC Question 12

Q: Please explain why a Doppler coefficient multiplier of 0.85 is used in the loss of load to one steam generator whereas the most negative moderator temperature coefficient is used.

A: A doppler coefficient multiplier of 0.85 is used in the Loss of Load to One Steam Generator to minimize negative reactivity insertion due to the doppler feedback effect as a result of the very slight power rise prior to trip. It should be noted, however, that this power rise and the resultant doppler feedback are very insignificant. Therefore, the choice of doppler coefficient multiplier has no impact on the results of the analysis.

The most negative moderator temperature coefficient is used for this event since it causes power to shift to the cold side of the core, therefore maximizing the post event F_r.