C-E POWER SYSTEMS

COMBUSTION ENGINEERING, INC.

RESPONSE TO NRC QUESTIONS ON THE CALVERT CLIFFS UNIT I CYCLE 4 RELOAD SUBMITTAL APPENDIX 1-NP, APRIL 25, 1979

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Question 1: It is stated in the reload application that the mechanical design for Batch F reload fuel is essentially identical to that of Batch E fuel. Identify the differences in mechanical design between these two fuel batches.

Response: Described below are the differences between the two batches:

a. Upper End Fitting Assembly - The holddown plate in the upper fitting has been thickened slightly. Since this reduces the holddown spring working length, the free length of the springs has been reduced by the same amount. Therefore, the holddown force has remained constant.

b. Lower End Fitting - The cross-bracing which connects the lower end fitting posts has been thickened and raised 1/8" from the lowermost surface of the fuel assembly.

c. [ ] - Sixteen Batch F assemblies have [ ] identical [ ] to the Batch E fuel. Another sixteen assemblies [ ]

Question 2: State the clad creep-collapse burnup limit and the highest assembly burnup at the end of Cycle 4.

Response: The maximum exposure for any Cycle 4 fuel assembly is 35,400 EFPH for the Batch B assembly in the core center location. The minimum collapse time predicted for any standard fuel rod within this assembly is 38,500 EFPH.

Question 3: The Batch B test assembly is under a Group 5 CEA. Why is it not necessary to sleeve the CEA guide tubes in this fuel assembly?

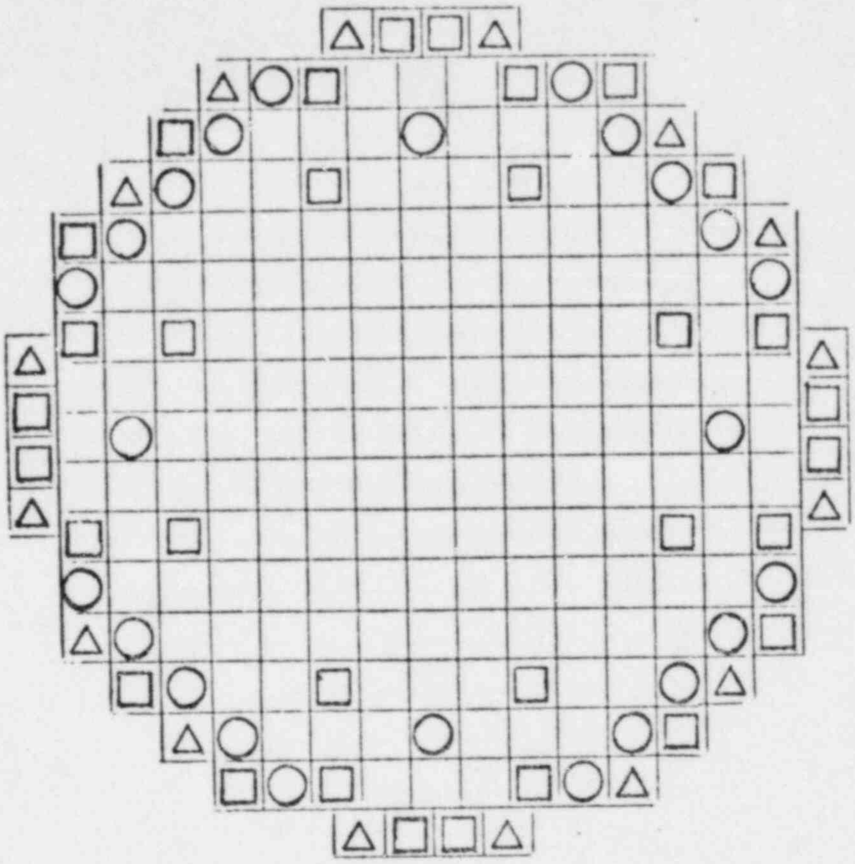
Response: The center core position is typically a low-wear location for fuel assemblies. However, as confirmation for the specific case of the Batch B test assembly, its guide tubes will be inspected during the shutdown and corrective action will be taken if the degree of wear is unacceptable.

Question 4: Of the 56 Batch F fuel assemblies not installed under CEA's identify (core location) which of these assemblies are sleeved and which have the [ ]?

Response: The distribution of the Batch F fuel assemblies is shown in Figure 1. In sumary, this figure shows:

- 24 Batch F assemblies under CEAs all of which are sleeved
- 16 Batch F assemblies with [ ]
- 32 Batch F assemblies with [ ] no sleeves

Please note that there are 48 Batch F fuel assemblies not installed under CEAs, rather than 56 as noted in the question.



△ Modified Design
□ Standard Design , No Sleeves
○ Standard Design , With Sleeves

Distribution of Batch F Fuel Assemblies

Figure 1

Question 5: Page 9 of the reload application indicates there are 15 demonstration rods in the Scout assembly. This does not agree with Reference 7 which states there are 20 demonstration rods. Correct as necessary.

Response: In the reload application, 15 demonstration rods refer to the presence of 15 fuel rods with other than standard Batch F fuel. Five additional test rods contain standard Batch F fuel pellets which have been characterized prior to loading. The total is twenty test fuel rods which is the figure presented in Reference 7.

Question 6: Does each [ ] rod have a uniform enrichment? If not, describe how the [ ]

Response: All [ ] within each [ ] have the same uniform enrichment. Enrichment will vary with type of pellet within each [ ] Standard Batch F pellets, [ ] (see Reference ~~7~~ of cycle 4 reload submittal).

11.47

Question 7: Provide an assembly drawing of the [ ] and detail drawing of the end closure and [connector.]

Response: The following proprietary C-E sketches are provided to satisfy this requirement

Sketch 9452-401-1, "SCOUT 1 Test [ ]

Sketch 9452-401-2, "SCOUT 1 Test [ ]

Question 8: Identify where the fuel assembly grids contact the [ ] with respect to the [ ]

Response: The fuel assembly grids contact the non-fueled regions of the [segmented rods] over the central 1 3/8" region of the [non-fueled sections]. The [ ] are [ ] in length.

Question 9: State in absolute terms the fill pressure for the normal Batch F rods, [ ]

Response Normal Batch F Rods - [ ]

[ ] - [ ]

[ ]-[ ]

SKETCH 9452-401-1

CONSTRUCTION DRAWING

SKETCH 9452-401-2

CONSTRUCTION DRAWING,
DETAILS

Question 10: What do you predict the end of life fuel rod pressure to be for both types of Scout demonstration rods?

Response: The predicted end-of-life internal pressure for all types of SCOUT test rods is less than the 2250 psia system pressure. Maximum burnup in the test rods will be 45,000 Mwd/Mtu.

Question 11: What surveillance or testing will be conducted on the C-E/EPRI fuel assembly during this outage?

Response: The following examinations are planned on the C-E/EPRI assembly during this outage:

A. Assembly examinations
   1. Visual Examination
   2. Assembly length
   3. Assembly bow
   4. Peripheral fuel rod and poison rod shoulder gaps
   5. Channel width measurements
B. Disassembly
C. Single rod examinations (up to 36 rods)
   1. Visual examination
   2. Rod length
   3. Eddy current test
   4. Profilometry
D. Re-Assembly
E. Six rods from BT03 to be sent to hot cell.

Question 12: Is the prototype CEA used in a sleeved fuel bundle or in a fuel bundle with [ ]?

Response: The prototype CEA is slated for installation in fuel assembly BT03, which is presently (4/79) unsleeved and which contains the original [ ]. Sleeving of this assembly is dependent upon site inspection following Cycle 3. No restriction exists as to the use of the CEA in a sleeved or unsleeved fuel assembly.

Question 13: How does the change in material and design of the prototype CEA effect the amplitude and frequency of the [ ] vibration of the poison pins? Will this impact on CEA guide tube wear?

Response: Only minor differences in cladding material density, weight, stiffness, and cross sectional area exist between the prototype cold-worked 316 stainless CEA rods and the conventional Inconel rods. Accordingly, changes in vibration and wear characteristics are insignificant.

Question 14: Describe how the value for the torque on the nuts that attach the CEA poison rods to the CEA spider was determined. Is this torque sufficient to account for differential thermal expansion, if any, that could cause the poison pins to vibrate in the CEA spider?

Response: The torque applied during assembly of the nuts is selected to provide a preload which exceeds the cyclic loads generated during stepping of the Control Element Drive Mechanism and which does not produce stresses in excess of allowable stresses. Differential expansion is in a direction which causes the joint to tighten at reactor operating temperatures, and has been accounted for in the joint design.

Question 15: Describe the surveillance planned on both the prototype CEA and the Scout demonstration fuel bundle.

Response: The first examination on the Scout demonstration bundle is scheduled for spring, 1980. Plans for that inspection will be developed closer to that date so that results from the Cycle 3 examination of the C-E/EPRI fuel assembly can be used to influence the direction of the examination. When examination plans are developed, they can be forwarded to the NRC for their information. Surveillance plans for the prototype CEA will be developed as the next shut-down date approaches. Visual examinations and eddy current testing are types of examinations that will be considered.

Question 16: Provide the Cycle 3 termination point when available.

Response: The Cycle 3 termination point is 9465 MWD/T.

Question 17: Identify the safety parameters calculated with ROCS and indicate which are measured during start-up testing, and which are measured over the course of the cycle.

Response: 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
- Changes in 3-D core power distributions that result from inlet temperature maldistributions (asymmetric steam generator transient)

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.

Measurements which confirm the adequacy of the Fuel Temperature Coefficients, Moderator Temperature Coefficient, Critical Boron Concentration and Reactivity Worth of the Regulating CEA Banks are performed during the startup testing.

The Moderator Temperature Coefficient is measured during the course of the cycle in accordance with the Technical Specification surveillance requirements.

The critical boron concentration is measured during the course of the cycle in order to make a determination relative to any reactivity anomaly in accordance with Technical Specification surveillance requirements.

Question 18: For each of the safety parameters calculated with ROCS provide the expected deviation of these as opposed to values that would have been obtained if the calculations were performed with PDQ and the bases for the expected deviations.

Response: Please refer to the list of safety parameters listed for question number 17.

The expected deviation between ROCS and PDQ are generally as follows:

Fuel Temperature Coefficients
ROCS and PDQ give essentially the same results for fuel temperature coefficients. This conclusion is based on comparisons of ROCS and PDQ calculations over several cycles.

Moderator Temperature Coefficients
ROCS tends to predict MTC values which are more negative then PDQ by approximately $-0.1X10^{-4}$ $\Delta\rho/°F$. This value has been determined by comparisons with measurements and is treated as a calculational bias.

Inverse Boron Worths
ROCS and PDQ give essentially idential results in the calculation of inverse boron worths.

Critical Boron Concentrations
3-D ROCS gives critical boron concentrations that are within 10 to 15 PPM of 2-D PDQ results.

CEA Drop Distortion Factors and Reactivity Worths
ROCS generally predicts CEA drop distortion factors within 2% and CEA worths within .02% $\Delta\rho$ of PDQ. These differences are based on a survey of ROCS and PDQ CEA drop calculations for a variety of CE first cycle and reload cores.

Reactivity Scram Worths and Allowances
2-D ROCS tends to give the same net reactivity scram worths as comparable PDQ calculations. 3-D ROCS tends to give higher scram worths than 2-D PDQ or 2-D ROCS for cases where the worst stuck CEA is a Type 11 or Type 12

(peripheral) CEA. This is due more to 3-D effects than to ROCS/PDQ differences. This observation is based on comparisons between ROCS and PDQ and comparison of both codes against measured CEA group worths.

CEA allowances are conservative estimates of CEA related effects such as scram worth which is unavailable due to CEA insertion to PDIL limits, etc. These bounding values are checked by a number of 2-D and 3-D calculations which are performed by the techniques described in the answer to question No. 20 below.

<u>Reactivity Worth of Regulating CEA Banks</u>
When ROCS is used in conjunction with PDQ in the manner described in the answer to question No. 20 below, the results generally agree to within 10% of measurement on individual groups and within 5% on accumulated groups.

<u>Asymmetric Inlet Temperature Maldistributions</u>
Comparisons of ROCS and PDQ in 2-D suggest that 3-D calculations of the effects of inlet temperature maldistributions are essentially the same as would be obtained from PDQ calculations. These calculations are used to obtain differences between symmetric and asymmetric power distributions. The results consist of comparisons between two 3-D ROCS calculations.

Question 19: For those safety parameters calculated with ROCS which are burn-up dependent state whether ROCS is used to calculate the fuel depletion.

Response: For those safety parameters calculated with ROCS which are burn-up dependent, three-dimensional ROCS calculations are used to calculate the fuel depletion.

Question 20: Explain the methodology by which two and three dimensional ROCS calculations were used in conjunction with two dimensional PDQ calculations to obtain the core parameters in Table 5-3.

Response: Table 5-3 is referred to in Section 5.3 of the Reload Submittal as an example of the type of core parameters that can be obtained using two and three dimensional ROCS in conjunction with two dimensional (2-D) PDQ.

In general "best estimate" core parameters are obtained from 3-D ROCS calculations. For those parameters which are relatively insensitive to power distribution such as MTC, FTC, critical boron worth, no correction is made. For those parameters such a CEA group worths which are sensitive to power distribution, a correction is made by multiplying the 3-D ROCS result by the ratio of a 2-D PDQ to a 2-D ROCS.
For example:

If Wi = worth of group i

then

$$Wi = Wi(\text{3-D ROCS}) \times \frac{Wi\ (\text{2-D PDQ})}{Wi\ (\text{2-D ROCS})}$$

This is viewed as a fine mesh correction to a coarse mesh calculation. The same equation, written as

$$Wi = \frac{Wi\ (\text{3-D ROCS})}{Wi\ (\text{2-D ROCS})} \times Wi\ (\text{2-D PDQ})$$

may be viewed as a 3-D correction (via ROCS) to a 2-D fine mesh PDQ.

The CEA worth data shown in Table 5-3 of this submittal was obtained from the set of 2-D fine mesh PDQ calculations that were used to generate the power distributions shown in Figures 5-1 through 5-5. This CEA worth information is presented in the licensing submittal for information purposes and is not used in the safety analysis.

For this application, no 2-D/3-D corrections were made to these worths. Experience has shown that these corrections would cause the group worths to increase or decrease by no more than several hundreths of a percent $\Delta\rho$.

Question 21: Show that the rod worth and the maximum percent increase in pin peak for the Dropped CEA event that was used in the safety analysis bounds the Cycle 4 calculated values.

Response: Table 5-5 of the Cycle 4 Reload Submittal contains the calculated values of the rod worth and the maximum percent increase in pin peak for the Dropped CEA event. The most severe Dropped CEA shown in this Table is the Type 11 into the All Rods Out configuration at Beginning of Cycle. This event has a worth of .13% $\Delta\rho$ and a percent increase in pin peak of 13.3%.

Table 7.2-1 of Reference 1 shows that the Cycle 3 Dropped CEA Safety Analysis used a CEA worth of 0.07% $\Delta\rho$ and a percent increase in pin peak of 16.0%. Since a lower worth and higher distortion produce more severe consequences, the Cycle 4 values are bounded by the Cycle 3 analysis.

Reference 1. Letter, J. W. Gore, Jr. to E. G. Case, "Third Cycle License Application", dated December 1, 1977,

Question 22

On Page 46 and Page 62 of the reload application, there appears to be a typo on the setpoint for the low flow pump trip. Correct as appropriate.

Answer

On Pages 46 and 62, the low flow trip setpoint should be corrected to 93% of 4-pump flow.

Question 23

In the analysis of the CEA Ejection event describe how TWIGL is used in conjunction with CHIC-KIN to define the k-factor?

During the CEA ejection event the k-factor accounts for the flattening of the static radial peaks. The k-factor is defined as the ratio of the maximum radial peak calculated with feedback effects (effective radial peaks) to the corresponding static radial peak calculated with no feedback effects. A study was performed using TWIGL, to determine the ratio of the effective radial peaks to the static radial peak for a wide range of ejected worths. A straight line relationship was obtained such that no data point was above it. This straight line defines the k-factor and is used in the analysis as a reduction factor on the static radial peaking factor. The k-factor in conjunction with the static radial peaking factor and the average core energy rise obtained from CHIC-KIN is used in calculating the average energy rise in the hottest fuel pellet.

Question 24

For the Seized Rotor Event, Table 7.5-2 identifies a dump valve opening. Identify this valve and state the effect of the operation of this valve on the transient?

Answer

On reactor trip, a turbine trip is automatically generated. This causes the atmospheric dump valve to automatically open. The opening of the atmospheric dump valve is used in the ensuing transient to remove decay heat and to aid in bringing the reactor temperatures to cold shutdown conditions. But note that the operator can manually close the atmospheric dump valve and use the steam bypass system and the condenser to remove decay heat, if necessary.

25. The small break LOCA analysis for CC-I submitted on March 13, 1979 uses a gap conductance of 1882 BTU/HR-FT^2°F and a fuel average temperature of 2286°F whereas Page 70 of the reload application presents a gap conductance of 1426 BTU/HR-FT^2°F and fuel average temperature of 2151°F. Explain the difference in values and their effect of the calculation results for PCT.

Response: The differences in gap conductance and fuel average temperature in the two analyses are a result of different peak linear heat generation rates (PLHGR) used in these analyses. The large break reload analysis was performed at a PLHGR of 14.2 kw/ft, whereas the small break analysis was performed at 16.0 kw/ft. Both analyses were performed using the same limiting fuel batch and time in life. The large break analysis was performed at the time in life of minimum gap conductance (maximum initial stored energy) which maximizes PCT. The small break LOCA analysis, although insensitive to initial fuel stored energy, employed the same time in life fuel performance data. The small break LOCA is sensitive to decay heat and hence a higher assumed PLHGR in the small break analysis results in a conservatively derived PCT.

26. Present the derivative rules used and their basis for predicting the effect of a variation of gap conductance and stored energy on PCT.

Response: No derivative rules were used in predicting the variation in gap conductance and stored energy. The gap conductance and associated fuel stored energy was obtained from burnup dependent calculations using the FATES[1] and STRIKIN-II[2] codes for Cycle 4. As discussed in the Calvert Cliffs Unit 1, Cycle 4 submittal, the Cycle 4 fuel parameters were bounded by those for Cycle 3.

References

1. CENPD-139, "CE Fuel Evaluation Model", July 1974 (Proprietary).

2. CENPD-135, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program, April 1974 (Proprietary).

   CENPD-135, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)", February 1975 (Proprietary).

   CENPD-135, Supplement 4, "STRIKIN-II, A Cylindrica[illegible] [illegible]metry Fuel Rod Heat Transfer Program", August 1976, (Propr[illegible]

   CENPD-135, Supplement 5, "STRIKIN-II, A Cylinc[illegible] [illegible]metry Fuel Rod Heat Transfer Program", April, 1977 (Proprietary).

Question

27. Describe the surveillance procedures and acceptance criteria you intend to use during this refueling outage to evaluate CEA guide tube wear. Provide us with the measurement data when it becomes available and an evaluation of the acceptability of fuel assemblies to be reused during Cycle 4.

Response

The inner diameter of thirty guide tube wear sleeves located in core positions which previously had high guide tube wear will be inspected with a [ ] eddy current test probe.

Eddy current testing results at Millstone II and St. Lucie I, on a total of 110 sleeves of similar design, indicate that after one full cycle of reactor operation that sleeve wear is virtually nonexistent. The examinations of sleeves at Calvert Cliffs I are being conducted to verify the performance of the sleeves. The verification standard is based on wear not exceeding a [ ] reduction in sleeve cross-sectional area and a volume loss of [ ] of the minimum volume required to [ ] the sleeve. This standard is equivalent to a [ ] inch wear depth tapered over [ ] of inner sleeve diameter arc. The eddy current test probe is capable of identifying a minimum wear depth of [ ] inches tapered over [ ] of inner sleeve diameter arc.

Question 28

Describe the basis for changing the constants in the P_{var} trip equation and the map of A1 vs ASI. Is this change solely due to the increased rod drop time?

Answer

The constant in the P_{var}^{trip} equation and the A1 vs. ASI graph were changed in order to realize additional margin through the removal of "fitting" conservatism in the previous constants. The increased rod drop time was a second order effect on the γ bias term (see answer to Question 30).

Question 29

Explain the necessity for the changes on Page B 2-6 T/S regarding steam generator water level and its effect on RCS pressure?

Answer

The safety analysis criterion for peak RCS pressure is that the upset pressure limit of 2750 psia (not the design pressure of 2500 psia) not be exceeded during any Anticipated Operational Occurrence (AOO). The proposed change simply reflects the correct safety analysis criterion on peak RCS pressure. This change has no impact on the calculated values of the RCS pressure during any AOO.

Question 30

Explain how the change in instrumentation response times, as indicated in Table 3.3-2 of T/S, is accounted for in setting plant operating limits.

Answer

The instrumentation response time (RTD time constant) is used in determining the a, T coefficient curve and the γ bias term. The a, T coefficients are used in the dynamic portion of the ΔT Power calculator. The bias term is applied directly in generating the γ coefficient of the P_{var} trip equation. The ΔT Power is in turn one of the inputs to the TM/LP trip. The change in the RTD time constant from 5 seconds to 8 seconds is reflected in the new a, T curve and in the change in the γ bias term from 30 to 62 psia.