SEP 30 1975

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Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Senior Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

We are currently reviewing your reload application of August 22, 1975 for the H. B. Robinson Unit 2 Cycle 4 core. In order for us to complete our review it will be necessary for you to provide additional information in support of this application.

Specific questions and areas needing further attention are itemized in the enclosure to this letter.

In order to keep the reload application review timely, you are requested to provide this information within two weeks of receipt of this letter. Your reply should include 3 signed originals and 37 copies of the requested information.

Sincerely,

/S/

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing

Enclosure:
Request for
Additional Information

cc: See next page

A-3

dispatch 10/7 OFFICE ➤	ORB#4	ORB#4				
SURNAME ➤	DNBridges/dg	RWReid				
DATE ➤	9/ 26 /75	9/26/75				

cc:

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
Barr Building
910 17th Street, N. W.
Washington, D. C. 20006

Hartsville Memorial Library
Home and Fifth Avenue
Hartsville, South Carolina 29550

REQUEST FOR ADDITIONAL INFORMATION

The following question refers to Section 3.2 of Exxon Topical Report XN-75-14

1. Provide an analysis of the consequences of the RCCA drop presented in Section 3.2 of XN-75-14 if automatic turbine cutback does not occur. Also, what happens if the system is under manual control and control rods are withdrawn to maintain power?

The following questions refer to Section 4.0 of Exxon Topical Report XN-75-38.

2. What fuel surveillance program is planned for these initial Exxon Fuel Assemblies in HBR?

3. The effects of a combined seismic and LOCA accident are not addressed in the HBR reload submittal. Will this be included in the seismic analysis to be submitted? If not, state what commitments will be made with regard to analyzing the effects of this accident?

4. Provide a drawing of the spacer grid assembly design which in particular shows the details of the Zr-4 grid strips and Inconel spring strips. What other differences are there between the Exxon and Westinghouse spacer grid assembly designs?

5. What method of attachment between the spacer grids and the guide tubes is used?

6. Provide a more detailed description, or a report if available, of the DLITE code which is used to analyze circumferential strain as a function of burnup and to design the fuel pellet and fuel cladding gap.

7. What were the types of thermal cycles and the number of cycles of each type considered to evaluate the effects of cyclic stresses in the cladding. Were both normal operational and abnormal (upset) transients considered in this evaluation.

8. As presented in Table 4.5, what is the difference between mechanical wear and fretting corrosion? What was the cause of the 0.8 and 1.0 mil depth wear at two locations attributed to mechanical wear in these fretting corrosion tests?

9. Provide an analysis or thermal hydraulic test data which shows the design adequacy of the holddown forces provided by the four leaf springs located in the upper grid plate in preventing fuel assembly liftoff.

10. In Section 4.2.2 which is titled "Fuel Temperature Analysis" values of UO_2 thermal conductivity, the UO_2 thermal conductivity integral, the fuel densification model and gap closure model used in the Exxon fuel thermal performance model are presented. Why are XN-209 the Exxon densification report, in which these models are described in detail, and the NRC staff safety evaluation on this report not referenced?

11. Provide a reference for the Hanevik data used to develop the densification rate expressions.

12. Describe the testing and inspections to be performed to verify the design characteristics of the fuel components including cladding integrity, verification of fuel enrichment, burnable poison concentration, fuel pellet characteristics, radiographic inspections, destructive tests, fuel assembly dimensional checks and the program for inspection of new fuel assemblies to assure mechanical integrity after shipment.

13. For the control rod ejection accident analyzed for H. B. Robinson Unit 2 in Exxon Report XN-75-44, what were the maximum cladding strains calculated for the cases analyzed?

14. The consequences of refueling accident does not appear to be addressed in the reload submittal. A reference to a previous analysis or an analysis for this accident should be provided.

15. Provide a drawing which shows the method of attachment of the control rod guide tubes to the upper and lower grid plates and the method of attachment of the spacer grids to the control rod guide tubes. Describe any differences between the Exxon design and the Westinghouse assemblies currently in HBR in these areas.

The following question refers to Section 5.0 of Exxon Topical Report XN-75-38.

16. Is the material on page 5.9, and associated figures being offered as an alternate or adjunct to your present power distribution control and monitoring method? It is our understanding that you plan to continue use of the presently adopted Constant Axial Offset Control (CAOC) System. Please advise us of your plans for power distribution control and monitoring and your schedule for submittal of the necessary supporting analysis for the reload core. Under separage cover we are providing for your information a recently established NRC position on CAOC.

The following question refers to Enclosure A to Letter of August 3, 1975.

17. More information is required regarding the following tests:

1) Initial Criticality CPL-R-6.1
2) Design Check Test CPL-R-6.2
3) Boron Dilution CPL-R-6.3
4) Boron Addition CPL-R-6.4
5) Power Distribution Maps CPL-R-9.4.

In particular, the acceptance criteria for these tests should be specified and related to values of physics parameters used in your accident analysis.

The following questions and request for discussion refers to Exxon Report XN-74-5

18. Pg. 6, Eqns. 2.1.5 to 2.1.10

Describe the manner by which T_f is determined from the fuel model shown on Figure 2.1. Discuss the considerations for axially weighted Doppler feedback reactivity in this program, and the lack of fuel temperature dependence for the Doppler coefficient α_D.

19. Pg. 7, Eqn. 2.1.14

Provide an assessment of the approximation in reactor power transients when using this equation in place of equation 2.1.1 for a +30¢ input reactivity step, and a 5¢/sec ramp up to +60¢ including crossover to equation 2.1.1 at +40¢ in the ramp transient.

20. Pg. 15, Eqn. 2.2.18

Discuss the rationale for coolant temperature summation over only 9 nodes in the 10 node model of Figure 2.2 to obtain core average moderator temperature.

21. Pg. 16, Eqn. 2.2.20

Identify the initial condition recommended for determining h_o, and provide an assessment of the transient error in h resulting from neglect of Nusselt number variation for the more severe coolant temperature excursions computed with this program.

22. Pg. 22, 2nd Par.

Describe the manner in which the variable time delay FDELAY is determined in PTS-PWR.

23. Pg. 25, 1st Par.

Describe the three point differencing technique used to compute primary system fluid mass changes.

24. Pgs. 31, 33, and 34, Eqns. 3.1.1, 3.1.3, and 3.1.4

Describe the manner by which U_{SUBC}, U_{BOIL}, and U_{STEAM} are determined in these equations.

25. Pg. 33, 1st Eqn.

Provide justification for use of this expression for ΔTH.

26. Pg. 34, 1st Eqn.

Provide a quantitative assessment of the error introduced into the determination of TIP3 from the assumption of negligible heat transfer when steam generator secondary quality is less than 0.89. Discuss the conditions resulting in reversal of primary loop flow.

27. Pg. 35, 2nd Eqn.

Identify the parameter Alpslp in this equation.

28. Pg. 35, Last Eqn.

Describe the steam generator operating condition resulting in the equality given by this equation, and describe the derivation of this equality from equation 3.1.4.

29. Pg. 36, 1st & 2nd Eqns.

Discuss the steam generator operating conditions with Qlps34 = 0 for both directions of primary flow.

30. General Suggestions

All parameters shown in the equations have not been identified in the list of nomenclature. Review of the report would be considerably expedited by use of conventional engineering nomenclature in the equations.

The following comments and questions apply to EXXON Report XN-75-44:

31. The nuclear power transient was calculated with the XTRAN code which has not been reviewed, therefore, questions on the rod ejection analysis may be incomplete.

32. Describe the DNB correlation used.

33. Describe the calculational method used to predict DNB and the number of fuel rods experiencing clad failure.

34. Discuss the reason why the total peaking factors after ejection are significantly lower than those usually predicted for similar plants.

35. Discuss the pressure surge calculation and show the variation of reactor pressure with time.

36. Present representative values of the Doppler and moderator feedback coefficients used in the calculations.

37. Evlauate the conservatism of the models and codes used by comparison with experiemnt, as available, and with more sophisticated spatial kinetics codes. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant should be evaluated.