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Docket No. 50-237

NOV 11 1977

Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

We are currently reviewing your application dated September 12, 1977, regarding Reload No. 3 for your Dresden Unit No. 2. In order for us to continue our review, we require the information requested in Enclosure 1. These questions are, with slight changes, the same as those telecopied to you on November 1 and 2, 1977.

Your prompt and complete response will help expedite our review.

Sincerely,

/s/

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

OFFICE ➤	DOR:ORB-2	DOR:ORB-2				
SURNAME ➤	RBevan:esp	DKDavis				
DATE ➤	11/9/77	11/10/77				

NOV 11 1977

cc w/enclosure:
Mr. John W. Rowe
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Mr. B. B. Stephenson
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

Anthony Z. Roisman, Esquire
Sheldon, Harmon & Roisman
1025 15th Street, N. W., 5th Floor
Washington, D. C. 20005

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

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REQUEST FOR ADDITIONAL INFORMATION

DRESDEN STATION UNIT NO. 2

DOCKET NO. 50-237

1. Specify the multiplying factors that have been applied to the nominal values of Doppler, void, and scram reactivity coefficients to obtain the values of these input parameters as given in Table 6-1 of NEDO-24034, Rev. 1, Supp. 4.

2. The results provided for the fuel loading error have included only the case of a misoriented fresh 8D250 bundle. Specify the LHGR and MCPR that would result from the worst case misloading of a fresh 8D250 bundle into a 7x7 bundle site.

3. With regard to the control rod drop accident analysis:

   (a) Provide the plant specific values of β and P_L and also provide any quantitative information to indicate how conservatism in these parameters might compensate for the cross over in scram reactivity curves beyond 3.75 seconds.

   (b) By how much does the value of scram reactivity inserted at 3.75 seconds differ from 0.0024 ΔK?

   (c) Specify the largest margin by which the bounding scram reactivity curve exceeds the plant specific curve beyond 3.75 seconds.

   (d) The description of the Banked Position Withdrawal Sequence in NEDO-21231 indicates that control rod drop accidents involving longer drops and larger worths than specified in NEDO-24034, Rev. 1, Supp. 4, are possible. Explain how it has been determined that the worst case rod drop for Dresden 2 Cycle 6 involves a drop from position 00 to 08 and insertion of 0.0024 ΔK as contrasted with the larger worth rod drops of NEDO-21231.

4. Based on NEDO-20360 it is expected that reload submittals will contain graphs showing instrument responses during rod withdrawal transients, including the effect of failed instruments. Either provide such graphical data, or provide a written description of how the effect of failed instruments was taken into account.

5. NEDO-20964, the GE topical report on void and Doppler reactivity feedback, indicates that the most severe transients occur at EOC values of void coefficient and scram reactivity. Since you plan

on derating the plant before EOC, you must show that the operating limits have considered the worst combination of power level and reactivity coefficients.

For the purpose of establishing that the correct operating limit MCPR for 8x8 fuel has been identified, specify how you will assure that the actual values of the void and scram reactivities and transient ΔCPR at the time of derate and through EOC 6 will remain bounded by the values given in NEDO-24034, Rev. 1, Supp. 4. Take into account any uncertainties in predicting rod configuration, void distributions and burnup at the time of derate through EOC 6.

6. Indicate your intentions for completion of the requested operational assurance tests as outlined in the appendices A and B.

7. Regarding the discussion of ASME vessel pressure code compliance in NEDO-24034, Rev. 1, Supp. 4:

   (a) Explain the statement that the safety/relief valves are assumed to be inactive.

   (b) Give the pressure relief capacity of the plant (in percent NBR steam flow of the plant) and also give the pressure relieving capacity assuming the most limiting safety valve fails to open.

   (c) Confirm the applicability of the sensitivity study of peak vessel pressure to valve operability described in the December 13, 1975 letter from I. Stuart (GE) to V. Stello (NRC) to Dresden Unit 2.

8. For the ASME Vessel Pressure Code Compliance analysis, the initial operating pressure is assumed to be 1005 psig in the vessel dome. The present Dresden Unit 2 Technical Specifications do not contain a requirement which limits the operating reactor (dome) pressure to that which was assumed in the overpressurization analysis. Moreover, sensitivity studies have not been performed which show the effect of initial operating pressure on the peak transient pressure attained during this limiting overpressure event. Therefore, either: 1) provide a sensitivity study which shows that increasing the initial operating pressure (up to the maximum pressure permitted by the high pressure trip point, i.e., 1060 psig) will have a negligible effect on the peak transient pressure or, 2) propose Technical Specification

changes which will assure that the reactor operating pressure will not exceed the initial pressure which was assumed in the Cycle 6 ASME vessel pressure code compliance analysis for Dresden Unit 2.

9. The Technical Specification ratio of the design total peaking factor to the maximum total peaking factor establishes both rod blocks and high flux setpoints. The maximum total peaking factor is calculated during the cycle by the process computer and compared to the design total peaking factor. Explain how changes in the relative numbers of 7x7 and 8x8 assemblies are accommodated in consistently normalizing the design total peaking factor and the maximum total peaking factor.

10. Qualitatively explain why the incorporation of a different pressure rule assumption from the previous analyses results in a greater reflooding time increase for QC12/D23 compared to other BWR/3s (as stated on the bottom of Page 6-6 of LOCA submittal).

11. Provide a qualitative explanation of the hot mode uncovery time versus break size (Figure 8 of LOCA submittal). The explanation should include a description of how the interrelation of the phenomena involved (new pressure rule, bypass area, etc.) combine for various break sizes to cause the calculated results. A thorough explanation on the lead plant (QC12/D23) may obviate the need for extensive questions on each "non-lead" plant, some of which have limiting breaks smaller than the DBA. It is hoped that the requested explanation will allow acceptance of such results on those other plants (e.g., Monticello) without extensive explanation and/or further calculations.

APPENDIX A

Operational Assurance and Tests

1. Startup physics tests selected from Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors", should be performed. The purpose of the startup test program is to (1) verify that the core was correctly loaded and there are no anomalies present which could cause problems later in the cycle, and (2) verify that the calculational model that has been used will correctly predict core behavior during this cycle. A test abstract summarizing the test objective, test method, and acceptance criteria should be presented.

   To validate the analytical models utilized for predicting plant responses to anticipated transients and postulated accidents, these tests should establish that measured responses are in accordance with predicted responses. The predicted responses should be developed using real or expected values of items such as beginning-of-cycle core reactivity coefficients, flow rates, pressures, temperatures, and the actual status of the plant and not those values or plant conditions assumed for conservative evaluations of postulated accidents. Acceptance criteria that are proposed should assure that the response of the plant to accidents and transients is in accordance with the design. Procedures to be followed if the acceptance criteria are exceeded should be discussed.

   Recommended parameters to be tested should include, but not necessarily be limited to, the following:

   a. Control Rod Drive Tests and Scram Time (Cold and Hot).

   b. Verification of Shutdown Margin (Highest Worth Control Rod Withdrawn).

   c. Critical Eigenvalue Comparison (Non-Voided Moderator Condition) for a Fixed Control Rod Pattern (deviation of a percent or more in reactivity should be immediately reported and explained).

   d. Power Distribution Comparison at a Given Control Rod Pattern and Power Level (>50% rated power with equilibrium xenon - anomalies should be reported and explained).

   e. TIP Reproducibility Test (>75% rated power).

   f. Core Power Symmetry Test (>75% rated power).

2. A summary report of the physics startup tests should be submitted to NRC within 45 days following completion of the startup test program. This report should include both measured and predicted values. If the difference between the measured and predicted value exceeded the acceptance criterion, the report should discuss the actions that were taken and justify the adequacy of these actions. Appendix B presents an outline which may be used for the physics startup tests summary report.

   A core loading map showing location and orientation of the reload and irradiated fuel should be provided. It should include the position of any test assemblies and should show initial enrichment distribution of replacement fuel, initial burnup distribution of irradiated fuel, and burnable poison distribution and concentration (if any). The map should include locations of control rods and indicate bank designations of each rod. Loading sequences, verification techniques for placement, and any pertinent shutdown margin or verification tests performed during actual reloading should be presented.

3. All tests that result in recalibration, new baseline settings, or other set point identifications should be reported according to Regulatory Guide 1.16 "Reporting of Operating Information - Appendix A Technical Specifications".

4. Tests and inspections performed to verify and characterize design aspects and parameters of the reload fuel system components should be described. Planned operational surveillance and subsequent post irradiation testing of fuel rods, burnable poison rods, and control rods should also be described. Planned comparisons between characterization tests and inspections, surveillance, and post irradiation tests for the reload fuel and the irradiated fuel remaining in the core should be presented.

APPENDIX B

SUMMARY REPORT OF PHYSICS STARTUP TESTS

I. A description of the test method and objectives for each test.

II. A comparison of test data with the predicted values and with the acceptance criteria, including tables of predicted values, measured values, and percent difference between predicted and measured values. Discuss the significance of the differences observed between appropriate predicted and measured values relative to the parameters used in accident analysis, development of operating philosophy, and technical specification requirements.

III. Deficiencies relating to design and fabrication found during conduct of the tests. Report the average fuel depletion (MWD/MTU), the number of fuel assemblies removed during the refueling, and the number of "leakers" based on wet or dry sipping tests.

IV. System modifications and corrective actions required and the schedule for their implementation.

V. Justification for acceptance of systems or components not in conformance with design predictions or performance requirements.

VI. Conclusions regarding system or component adequacy.