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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 14, 1980

Docket No. 50-317

Mr. A. E. Lundvall, Jr.
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Dear Mr. Lundvall:

In the process of reviewing your Cycle 5 reload request dated September 22, 1980, we find that additional information as detailed in the enclosure is needed to complete our review. The additional information being requested was previously sent to Mr. J. Lippold of your staff by telecopy on November 7, 1980.

In order to meet the agreed upon schedule for this review, please provide the additional information by at least November 19, 1980.

Sincerely,

Robert A. Clark, Chief
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Enclosure:
Request for Additional
Information

cc: w/enclosure
See next page

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION NO. III

CYCLE 5 RELOAD REVIEW

Calvert Cliffs, Unit 1

Docket No. 50-317

1 Enclosure 2 makes a distinction between Experimental Test Assemblies and Lead Prototype Assemblies.

a. Are the 4 so-called PROTOTYPE assemblies in Cycle 5 Experimental Test Assemblies or Lead Prototype Assemblies according to the definition in the attachment.

b. If these 4 assemblies are Lead Prototype Assemblies adequate surveillance must be performed at EOC-5 to support following core reloads of that fuel type. If this is the case, please decribe the fuel surveillance that is planned for EOC-5 and how the results will be provided to the NRC. Please note that this surveillance should be more extensive than that performed for mature cores or first cores, but poolside techniques may suffice.

c. If these 4 assemblies are Experimental Test Assemblies, surveillance at EOC-5 is not required but results from any post-irradiation examination (PIE) would be of interest to the NRC. In this case, please indicate what PIE results would be made available to the NRC.

2 Appendix D provides a description of the PROTOTYPE fuel designs but contains no safety analysis. A safety analysis is needed to show that the effects of the design changes have been accounted for and that there will be no significant adverse impact on reactor safety in terms of reduction of safety margins or increased radiological releases during either normal or accident conditions. Inasmuch as the PROTOTYPE demonstration's "raison d'etre" is high burnup, this safety analysis should incorporate an analysis of high-burnup effects on the PROTOTYPE designs, taking into account potential burnup (exposure) related damage mechanisms which might be aggravated by burnups higher than traditionally achieved.

3 What are the target burnups of the PROTOTYPE assemblies (in terms of both assembly average and lead rod)?

4 Would one of the three identical PROTOTYPE assemblies be irradiated through a fourth cycle, or would one assembly with spacer pellets and graphite coatings be the candidate?

5 Appendix D provides a description of the PROTOTYPE fuel designs. Please provide additional detail information, including sketches as necessary, for each new modified design feature for the PROTOTYPE fuel assemblies.

6 Section 15.4.7 of the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-75/087) requires an analysis of possible fuel loading errors such as the loading of one or more fuel assemblies into improper locations. Discuss the analyses for each misloading case (including the worst case) considered and show that either the error is detectable (and thus remedial) or that the error is inconsequential and within the nuclear uncertainty or that the offsite consequences of any core damage due to undetected errors are a small fraction of 10 CFR 100 guidelines.

7 A partial list of physics characteristics for Cycles 4 and 5 was presented in the refueling license amendment. Provide a list of final Cycle 5 physics characteristics if different from the original submittal (Tables 5-1 thru 5-6) including the maximum radial power peaks expected to occur (F_r and F_{xy} with uncertainties and biases).

8 Please provide the following burnup values:

a. The actual core average exposure achieved during Cycle 4.
b. Each batch average exposure for EOC 4.
c. Each batch average exposure for EOC 5.

9 Cycle 5 will employ several different guide tube and sleeve designs that are specifically choosen to mitigate or investigate guide tube wear. Please provide a table for the Cycle 5 core inventory that identifies these different fuel assembly designs. The table should indicate the number of fuel assemblies in each category--sleeved, sleeve design, guide tube design--and the number of fuel assemblies in each category to reside under CEAs.

10 From batch B of Cycle 4 there are 15 C-E/EPRI test rods that will be loaded into fuel assembly D047 for Cycle 5 operation. The creep collapse evaluation for 12 of these rods was based on an inspection performed during a previous outage rather than on the standard analysis. That inspection did not reveal large cladding ovalities; consequently, you concluded that no collapse during Cycle 5 would be expected. Since these test rods will be inspected during this outage, please confirm this conclusion based on current observation.

11 Because of the need for higher enrichments associated with extended burnup cycles, the 52 batch G/ fuel assemblies to be loaded for Cycle 5 operation are to each employ 8 burnable poison pins. Was the design of these burnable poison pins the same as that used in the initial core loading? If so, were

the revised C-E design modifications and manufacturing process changes--to preclude the earlier type of hydride-induced failures--employed? If these poison pins for Cycle 5 operation are of a different design, please provide appropriate design drawings and descriptions. It would also be helpful to describe the results of any other licensee's operating experience with any new type of poison pin design that would be used in Cycle 5.

12 Differential growth between fuel rods and the fuel assembly structure is burnup dependent and progresses in such a manner that mechanical interference will occur when the shoulder gap is consumed. What assurance exists that the shoulder gap in the high-burnup fuel assemblies comprising the Cycle 5 core is adequate to preclude this interference.

13 Section 7.3.1 of the Calvert Cliffs Unit 1 Cycle 5 submittal (1) shows CEA ejection accident results based on an enthalpy threshold for cladding failure. The enthalpy values provided in the submittal have never been approved and have been shown (2) to be non-conservative with respect to regulatory guides assumptions for this event at similar C-E NSSS-designed plants. Provide a reanalysis of the CEA ejection accident based on a minimum DNBR for cladding failure as described in Regulatory Guide 1.77 (3).

14 The licensee has previously submitted information (4,7) to show that an NRC-supplied correction (5,6) of the fission gas release model in the Calvert Cliffs Unit 1 fuel performance analysis would not result in rod internal pressure exceeding nominal primary system pressure nor would it result in any adverse effect on LOCA or other safety analysis. The burnup and peak linear heat generation rate used in that analysis do not reflect the values anticipated for Cycle 5 operation. Provide results of an analysis for these conditions showing continued acceptability of these results for Cycle 5 operation.

15 The NRC staff has been generically evaluating three materials models that are used in ECCS evaluation models. Those models are cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have (a) met and discussed our review with Combustion Engineering and other industry representatives (8), (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis" (9), and (c) required fuel vendors and licensees of LWRs with Zircaloy cladding to confirm that their plants would continue to be in conformance with the ECCS criteria of 10 CFR 50.46 if the materials models of NUREG-0630 were substitued for those models of their ECCS evaluation models (10 and 11).

BG&E responded (12) to the NRC request of Reference 10. Please either amend as necessary or confirm that that BG&E response (12) continues to be applicable for Cycle 5 operation of Calvert Cliffs Unit 1 in light of changes in plant variables for Cycle 5 (e.g., the increase in peak LHGR to 15.5 kW/ft).

16 For Cycle 4 operation, BG&E imposed (13) restrictions on CEA movement while at primary system temperatures below 400°F. This restriction was in response to a C-E recommended operational guideline to reduce CEA drag forces on relaxed crimp joints (i.e., the bond which holds sleeve inserts to guide tubes). Is the continuation of this restriction prudent for Cycle 5 operation? If not, please describe the basis for discontinuation.

17 The reload safety analysis (1) provides an insufficient discussion on fuel rod bowing and its effect on DNBR margin. The two sentences that are provided do not identify the bow-magnitude and reduction-in-DNBR correlations used for the Calvert Cliffs Unit 1 analysis. Neither is there any information given on the input parameters used for the calculations. Futhermore, the BG&E discussion of fuel rod bowing references the Combustion Engineering generic topical report CENPD-225, "Fuel & Poison Rod Bowing" that has not been approved for licensing applications.

Please identify the correlations that were used and cite their NRC approval Provide the input that was used for the Cycle 5 analysis. Describe any generic or plant-specific DNBR margins that were required to offset fuel rod bowing effects.

18 Before startup, provide a preliminary justification for continued operation with guide tubes sleeved and with reduced flow. As customary, formal justification should be submitted within 30 days after returning to power, and that justification should address the need for continuing surveillance during the next outage.

19 In previous cycles the thermal margin/low pressure trip setpoints included an allowance to compensate for pressure measurement error, trip system processing error, and time delay associated with providing effective termination of the occurence that exhibits the most rapid decrease in margin to the safety limit. For Cycle 4, this 84 psia allowance was made up of a 22 psia pressure measurement allowance and a 62 psia time delay allowance. The CEA withdrawal transient initiated at rated power resulted in the maximum pressure bias factor of 62 psia. In the CEA withdrawal analysis for Cycle 5, a maximum pressure bias factor of 70 psia is obtained but is appears that no additional pressure measurement allowance term is included in the TM/LP Tech. Spec. bases (B 2-7). Also, power measurement error potential (5%) and temperature measurement uncertainty (2°F) were dropped. Since the SCU and CEAW topicals will not be reviewed for this reload shouldn't the TM/LP trip setpoints be rederived as in previous cycles?

References

1. Calvert Cliffs Unit 1 Cycle 5 Refueling License Amendment, BG&E, September 22, 1980.

2. Arkansas Nuclear One, Unit 2, FSAR, Amendment No. 34, pages 15.120-3a/7.

3. "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Regulatory Guide 1.44, May 1974.

4. Letter from A. E. Lundvall, Jr., BG&E, to D. L. Ziemann, NRC, dated December 31, 1976.

5. Letter from D. L. Ziemann, NRC, to A. E. Lundvall, Jr., BG&E, dated November 23, 1976.

6. R. O. Meyer, C. E. Beyer and J. C. Voglewede, "Fission Gas Release from Fuel at High Burnup," NRC Report NUREG-0418, March 1978.

7. Letter from A. E. Lundvall, Jr., BG&E, to D. L. Ziemann, NRC, dated February 23, 1977,

8. Memorandum from R. P. Denise, NRC, to R. J. Mattson, "Summary Minutes of Meeting on Cladding Rupture Temperature, Cladding Strain, and Assembly Flow Blockage," November 20, 1979.

9. D. A. Powers and R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NRC Report NUREG-0630, April 1980.

10. Letter from D. G. Eisenhut, NRC, to all Operating Light Water Reactors, dated November 9, 1979.

11. Memorandum from H. R. Denton, NRC, to Commissioners, "Potential Deficiencies in ECCS Evaluation Models," November 26, 1979.

12. Letter from A. E. Lundvall, BG&E, to D. G. Eisenhut, NRC, dated January 31, 1980.

13. Letter from A. E. Lundvall, BG&E, to R. W. Reid, NRC, dated May 31, 1979.

ATTACHMENT

Staff Recommendations for Fuel Surveillance in

Commercial Power Plants

Experimental Test Assemblies--The irradiation of experimental test assemblies is encouraged. Since these assemblies will be limited to a small core fraction and since a safety analysis will have been performed, no extensive PIE is (generally) needed to assure safe operation of the plant containing the test assemblies.

Lead Prototype Assemblies--Lead prototype assemblies differ from experimental test assemblies inasmuch as a following core reload is scheduled. Surveillance of the lead prototype assemblies would thus be required in support of the following core reload but not (generally) to assure safety of the cycle containing the lead prototype assemblies. An instructive example is the requirement for Surry to perform surveillance on 17x17 lead prototype assemblies in support of a first core 17x17 fuel in Trojan. The surveillance requirement exists because of the timing of the Trojan core loading; if timely results from Surry were not obtained, the assurance of safe operation of Trojan would be compromised. Westinghouse acted as a broker in that case and got Surry to make commitments to NRC in behalf of the Trojan submittal.

First Core Loading--Detailed surveillance, including interim examinations, is required to confirm the safety analysis of a new fuel design. This detailed surveillance has been required on the first two plants to use the design in order to sample a statistically large number of assemblies and also to sample effects of different manufacturing and operating histories.

Mature Core Loadings--Simplified surveillance is now required on a routine basis for mature fuel designs. This requirement is a result of activities on Regulatory Guide 1.119 (withdrawn in favor of SRP revisions) and appears in SRP-4.2, Rev. 1. The requirement is an attempt to catch anomalies that result from insidious changes in plant operation or fabrication histories. Recent problems with poison rod failures and guide tube wear support the need for such wide-scale surveillance.