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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-53

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-317

1.0 INTRODUCTION

By letter dated February 12, 1988, as supplemented on March 21, March 25 (2 letters) and April 14, 1988, the Baltimore Gas and Electric Company (BG&E or the licensee) submitted a request for an amendment to its operating license for Calvert Cliffs Unit No. 1 to allow operation for a tenth cycle at a 100% rated core power of 2700 MWt (Ref. 1). The licensee also submitted proposed modifications to the Technical Specifications (TS) for Cycle 10. Cycle 10 will have a 24 month cycle length as compared to 18 months for the previous cycle.

The licensee submitted a final camera-ready copy of the previously requested TS on April 14, 1988.

The supplements to the February 12, 1988 submittal did not affect the proposed TS change noticed in the Federal Register on April 15, 1988, with correction on April 29, 1988, and did not affect the staff's proposed no significant hazards determination.

The NRC staff has reviewed the application and the supporting documents (Refs. 2 & 3) and has prepared the following evaluation of the fuel design, nuclear design, thermal-hydraulic design, and TS changes.

2.0 EVALUATION OF FUEL DESIGN

2.1 Fuel Assembly Description

The Cycle 10 core consists of 217 fuel assemblies. Ninety-six fresh (unirradiated) Batch M assemblies will replace previously irradiated assemblies. Of these 96 fresh assemblies, 92 will be manufactured by Combustion Engineering (CE) and four by Advanced Nuclear Fuels (ANF) Corporation, and are placed in the Cycle 10 core as an aid in qualifying ANF fuel for 24 month cycle operation. The 92 fresh CE assemblies will consist of 16 unshimmed Batch M assemblies and 76 Batch M* assemblies each containing 12 B_4C rods for neutronic shimming and having an initial assembly average enrichment of 4.08 weight percent (w/o) U-235. The four ANF Batch MX demonstration assemblies contain 12 fuel bearing Gd_2O_3 rods for shimming and have an initial assembly average enrichment of 3.85 w/o U-235.

2.2 Mechanical Design

The mechanical design of the CE Batch M reload fuel is identical to the Batch K fuel previously inserted in Calvert Cliffs Unit 1. All CE fuel to be loaded for the Cycle 10 core was reviewed to ascertain that adequate shoulder gap clearance exists. Analyses were performed with approved models and the licensee concluded that all shoulder gap and fuel assembly length clearances are adequate for Cycle 10. The replacement control element assembly (CEA) to be used in the center location of the core will have the same reconstituted features as the replacement CEA installed in the reference cycle.

The mechanical design features of the ANF lead fuel assemblies are described in Reference 3. Most of the assembly and core interface dimensions are identical to the CE fuel assemblies. Differences in the upper and lower end fitting height and overall assembly height should not affect the performance of either fuel assembly. Experience with similar ANF fuel designs co-residing adjacent to CE reload fuel in the Maine Yankee, Fort Calhoun, and St. Lucie Unit 1 cores have caused no unexpected problems or operational difficulties. Therefore, the staff finds the ANF lead assemblies to be mechanically compatible with the co-resident CE fuel during Cycle 10.

2.3 Thermal Design

The thermal performance of the CE fuel in Cycle 10 was evaluated using the FATES3B fuel evaluation model (Ref. 4). The staff issued an SER (Ref. 5) approving the use of FATES3B for BG&E licensing submittals. The licensee analyzed a composite, standard fuel pin that enveloped the various CE fuel batches in Cycle 10. The analysis modeled the power and burnup levels representative of the peak pin at each burnup interval. Although the burnup range analyzed for the peak pin was greater than that expected at the end of Cycle 10, approximately 0.3% of the fuel pins will achieve burnups greater than the 52,000 MWD/T value approved for CE fuel (Ref. 6) if Cycles 9 and 10 are operated to their maximum burnups. In response to the staff's request, the licensee confirmed that these few high burnup pins will be in low power regions of the Cycle 10 core and the maximum pressure within these pins will not reach the nominal reactor coolant system pressure of 2250 psia (Ref. 7).

Evaluations have been performed to show that the four ANF lead assemblies are thermally compatible with the existing CE fuel assemblies and meet the appropriate fuel thermal design criteria required by the staff (Ref. 3).

Based on its review of the information discussed above, the staff concludes that the evaluation of the thermal design of the CE and ANF fuel for Cycle 10 is acceptable.

3.0 EVALUATION OF NUCLEAR DESIGN

3.1 Fuel Management

The Cycle 10 core consists of 217 fuel assemblies, each having a 14 by 14 fuel rod array. A general description of the core loading is given in Section 2.1 of this SER. The highest U-235 enrichment occurs in the CE Batch M fuel assemblies which contain an assembly average enrichment of 4.08 w/o U-235. The Calvert Cliffs fuel storage facilities have been approved for storage of fuel of maximum enrichment of 4.10 w/o U-235 and, therefore, the fresh Batch M assemblies are acceptable from a fuel storage aspect.

The Cycle 10 core will use a low-leakage fuel management scheme. With the proposed loading, the Cycle 10 reactivity lifetime for full power operation is expected to be 21,400 MWD/T based on a Cycle 9 length of 11,800 MWD/T. The analyses presented by the licensee will accommodate a Cycle 10 length between 20,600 MWD/T and 21,800 MWD/T based on Cycle 9 lengths between 9,800 MWD/T and 11,800 MWD/T.

3.2 Power Distribution

Hot full power (HFP) fuel assembly relative power densities are given in the reload analysis report for beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) unrodded configurations. Radial power distributions at BOC and EOC are also given for control element assembly (CEA) Bank 5, the lead regulating bank, fully inserted. These distributions are characteristic of the high burnup end of the Cycle 9 shutdown window and tend to increase the radial power peaking in the Cycle 10 core. The four ANF lead test assemblies were calculated to have maximum pin power peaking at least 10% lower than the maximum pin peaking in the core under all expected Cycle 10 operating conditions. The distributions were calculated with approved methods and include the increased power peaking which is characteristic of fuel rods adjacent to water holes. In addition, the safety and setpoint analyses conservatively include uncertainties and other allowances so that the power peaking values actually used are higher than those expected to occur at any time in Cycle 10. Therefore, the predicted Cycle 10 power distributions are acceptable.

3.3 Reactivity Coefficients

In order to accommodate 24 month cycles, the moderator temperature coefficient (MTC) limit above 70% power is raised from $+0.2\times10^{-4}$ delta rho/° F to a value which varies linearly from $+0.3\times10^{-4}$ delta rho/° F at 100% power to $+0.7\times10^{-4}$ delta rho/° F at 70% power. The staff has previously expressed concern about the positive MTC effect on the generic anticipated transients without scram (ATWS) assumptions and BG&E has stated that they will address the generic ATWS implications, if any, in the future. In the interim, the staff has approved operation for core designs with allowable positive MTC values provided that the MTC becomes negative at 100% power and equilibrium xenon conditions. The licensee has predicted a negative MTC at hot full power, equilibrium xenon conditions of -0.2×10^{-4} delta rho/° F for Cycle 10 and has committed to a full power negative value at equilibrium xenon conditions (Ref. 7).

The Doppler coefficient for Cycle 10 is a best estimate value expected to be accurate to within 15%. These reactivity coefficient values are bounded by the values used in the safety analyses for the reference cycle (Calvert Cliffs Unit 2 Cycle 8). The staff, therefore, finds the values of the MTCs and Doppler coefficients to be acceptable.

3.4 Control Requirements

The CEA worths and shutdown margin requirements at the most limiting time for the Cycle 10 nuclear design, that is, for the EOC, are presented in Reference 7. These values are based on an EOC, hot zero power (HZP), steamline break accident. At EOC 10, the reactivity worth with all CEAs inserted is 9.0% delta rho. An allowance of 1.1% delta rho is made for the stuck CEA which yields the worst results for the EOC HZP steamline break accident. An allowance of 2.0% delta rho is made for CEA insertion in accordance with the power dependent insertion limit (PDIL). The calculated scram worth is the total CEA worth less the worth of the stuck CEA and less the worth of CEA insertion to the PDIL and is 5.9% delta rho. Deducting 0.8% delta rho for physics uncertainty and bias yields a net available scram worth of 5.1% delta rho. Since the TS EOC shutdown margin at zero power is 5.0% delta rho, a margin of 0.1% delta rho exists in excess of the TS shutdown margin. Therefore, sufficient CEA worth is available to accommodate the reactivity effects of the steam line break event at the worst time in core life allowing for the most reactive CEA stuck in the full withdrawn position. The staff concludes that the licensee's assessment of reactivity control is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming a stuck CEA that results in the worst reactivity condition for an EOC, HZP steamline break accident. Thus, the control requirements are acceptable.

3.5 Safety Related Data

Other safety related data such as limiting parameters of dropped CEA reactivity worth and the maximum reactivity worth and planar power peaks associated with an ejected CEA for Cycle 10 are identical to the values used in the reference cycle and are, therefore, acceptable.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

4.1 DNBR Analysis

Steady state thermal-hydraulic analysis of CE fuel for Cycle 10 is performed using the approved core thermal-hydraulic code TORC and the CE-1 critical heat flux correlation (Ref. 8). The core and hot channel are modeled with the approved method described in CENPD-206-P-A (Ref. 9). The design thermal margin analysis is performed using the fast running variation of the TORC code, CETOP-D (Ref. 10), which has been approved for Calvert Cliffs with the appropriate hot assembly inlet flow starvation factors to assure its conservatism with respect to TORC. The engineering hot channel factors for heat flux, heat input, rod pitch and cladding diameter are combined statistically with other uncertainty factors using the approved extended statistical combination of uncertainties (ESCU) method described in CEN-348(B)-P (Ref. 11) to arrive at an equivalent departure from nucleate boiling ratio (DNBR) limit of 1.15 at a 95/95 probability/confidence level.

DNBR analyses were also performed to assess the performance of the ANF lead assemblies (Ref. 3) using the XCOBRA-III code (Ref. 12) and the ANF approved thermal-hydraulic methodology for mixed fuel cores (Ref. 13). The XNB departure from nucleate boiling correlation (Ref. 14) has been shown to be applicable to co-resident CE and ANF fuel (Refs. 14 & 15) and the staff concludes that it is acceptable to apply it to the mixed Cycle 10 core containing the four ANF lead fuel assemblies. The results indicate that the ANF lead assemblies exhibit higher MDNBRs than the hot CE assembly due to the 5% lower assembly power at which the ANF lead assemblies were simulated. Since the insertion of the ANF lead assemblies does not significantly affect the minimum DNBR (MDNBR) of the hot CE assembly, which establishes the core MDNBR, the staff concludes that the core MDNBR is essentially unchanged by insertion of the four ANF lead assemblies and thus the design criterion on DNBR is satisfied by the mixed core containing ANF lead assemblies. Thus, the results of the DNBR analysis are acceptable.

4.2 Fuel Rod Bowing

The fuel rod bow penalty accounts for the adverse impact on MDNBR of random variations in spacing between fuel rods. The methodology for determining rod bow penalties for Calvert Cliffs was based on the NRC approved methods presented in the CE topical report on fuel and poison rod bowing (Ref. 16). The penalty at 45,000 MWD/T burnup is 0.006 in MDNBR. This penalty is included in the ESCU uncertainty allowance discussed above. For those assemblies with average burnup in excess of 45,000 MWD/T, sufficient margin exists to offset rod bow penalties. The staff, therefore, concludes that the analysis of fuel rod bow penalty is acceptable.

5.0 EVALUATION OF SAFETY ANALYSES

5.1 Non-LOCA Events

For the non-LOCA safety analyses, the licensee has determined that the key input parameters for the transient and accident analyses lie within the bounds of those of the reference cycle (Unit 2 Cycle 8). As noted in Section 6.0, the shutdown margin TS is being changed from a singular value to a variable ranging from 3.5% delta rho at BOC to 5.0% delta rho at EOC. The EOC shutdown margin requirement is determined by the steam line rupture event and a reevaluation of this event at EOC 10 with the revised shutdown margin has indicated that it is less limiting than the reference analysis. The staff, therefore, concludes that the non-LOCA transient and accident events for Cycle 10 are bounded by the reference analyses and, therefore, the results of the non-LOCA safety analysis are acceptable.

5.2 LOCA Events

The large break loss of coolant accident (LOCA) has been reanalyzed for Cycle 10 to demonstrate that a peak linear heat generation rate (PLHGR) of 15.5 kw/ft complies with the acceptance criteria of 10 CFR 50.46 for emergency core cooling systems (ECCS) for light water reactors. The Cycle 10 analysis, as the reference cycle analysis, was performed with the 1985 CE evaluation model which was approved in Reference 17. The Cycle 10 analysis showed that the double ended guillotine pipe break at the pump discharge with a discharge coefficient of 0.6 (0.6 DEG/PD) gave the highest peak clad temperature. Table 8.1-1 of the reload report provides the input parameters for the fuel for Cycle 10 and the reference cycle. Table 8.1-2 presents the results of the analysis for the limiting break for Cycle 10 and the reference cycle. The results for the limiting Cycle 10 break show that (1) the peak clad temperature is 1983° F which is well below the acceptance criterion of 2200° F and (2) the maximum local and core wide oxidation values are 4.14% and less than 0.51%, respectively, and these are well below the acceptance criteria of 17% and 1%, respectively. The analysis considered up to 500 plugged tubes per steam generator and a 40 second safety injection pump response time. Since the Cycle 10 large break LOCA ECCS analysis has shown that both the peak clad temperature and clad oxidation meet the acceptance criteria of 10 CFR 50.46, the operation of Cycle 10 at an allowable PLHGR of 15.5 kw/ft is acceptable.

The licensee reports that analyses have confirmed that small break loss of coolant accident (SBLOCA) results for Calvert Cliffs Unit 1 Cycle 8, which is the reference cycle for SBLOCA, bound the Calvert Cliffs Unit 1 Cycle 10 results. Unlike the large break LOCA analysis, the SBLOCA considered only 100 plugged tubes per steam generator. The increased safety injection pump response time considered in the large break analysis also was not evaluated for the SBLOCA analysis. Since the acceptance criteria for the SBLOCA are met, the operation of Cycle 10 at an allowable PLHGR of 15.5 kw/ft, with up to 100 plugged tubes per steam generator, is acceptable.

6.0 TECHNICAL SPECIFICATIONS

As indicated in the staff's evaluation of the nuclear design, provided in Section 3, the operating characteristics of Cycle 10 were calculated with approved methods. The proposed TS are the results of the cycle specific analyses for, among other things, power peaking and control rod worths. The analyses performed include the implementation of a low-leakage fuel shuffle pattern with fuel enrichments and burnable poison loadings and distributions chosen to provide a cycle length of 24 months. Some of the requested TS changes involve changes to both Unit 1 and Unit 2 TS. Each proposed change is discussed below.

6.1 Figure 2.2-2 Thermal Margin/Low Pressure Trip Setpoint-Part 1

Figure 2.2-2 is modified due to a revision in the curve fit for the TM/LP trip setpoint to accommodate the implementation of the extended statistical combination of uncertainties methodology. The setpoint analysis uses this methodology and the licensee has determined that acceptable results are obtained for Cycle 10. The changes to Figure 2.2-2 are, therefore, acceptable.

6.2 Figure 2.2-3 Thermal Margin/Low Pressure Trip Setpoint-Part 2

Figure 2.2-3 is modified for the same reason as Figure 2.2-2 and the change is acceptable for the same reason.

6.3 Bases 2.1.1 and 2.2.1

The text is modified to replace a specific MDNBR value with the phrase DNB SAFDL. The use of a phrase in place of a specific MDNBR value was recommended in the extended SCU methodology (Ref. 11) and approved by the staff (Ref. 18). The change is, therefore, acceptable.

6.4 Technical Specification 3.1.1.1 Shutdown Margin

Two modifications are proposed for this TS. First, the shutdown margin is changed from a constant value to text which refers to a new Figure 3.1-1b which presents shutdown margin as a function of time in cycle. Since the required shutdown margin varies throughout the cycle due to fuel depletion, boron concentration and moderator temperature and this variation with cycle time has been incorporated in all the appropriate safety analyses for Cycle 10, this change is acceptable.

The shutdown margin at EOC is increased from 3.5% delta k/k to 5.0% delta k/k. The analysis of the Cycle 10 steam line rupture analysis, which is limiting at hot zero power EOC conditions, supports this change and it is, therefore, acceptable.

6.5 Technical Specification 3.1.1.4 Moderator Temperature Coefficient

The MTC limit above 70% power is being raised from $+0.2\times10^{-4}$ delta rho/° F to a value which varies linearly from $+0.3\times10^{-4}$ delta rho/° F at 100% power to $+0.7\times10^{-4}$ delta rho/° F at 70% power. This change is being implemented to accommodate 24 month cycles and to facilitate initial reactor startup at the beginning of the cycle. The licensee has committed to a negative MTC at hot full power, equilibrium xenon conditions. As mentioned in Section 3.3, this value has been predicted to be -0.2×10^{-4} delta rho/° F. The feedline break analysis which supports this change is applicable to Cycle 10 and, therefore, the proposed change is acceptable.

6.6 Figure 3.1-2 CEA Group Insertion Limits

The transient insertion limit between 90% and 100% power is being increased from an allowed insertion limit which varies linearly from 35% for Bank 5 at 90% power to 25% at 100% power, to a constant value of 35%. This change, which is being made to enhance the ability to control axial oscillations near EOC, has been incorporated into all of the Cycle 10 physics, safety and setpoint analyses and is, therefore, acceptable.

6.7 Figure 2.2-1 Axial Power Distribution Trip LSSS

Figure 2.2-1 is modified to increase the positive and negative axial shape index (ASI) regions below 70% power. The setpoint analysis uses the modified results given by Figure 2.2-1 and the licensee has determined that acceptable results are obtained for Unit 1 Cycle 10 and Unit 2 Cycle 8. The changes to Figure 2.2-1 are, therefore, acceptable for both units.

6.8 Figure 3.2-2 Linear Heat Rate Axial Flux Offset Control Limits And Figure 3.2-4 DNB Axial Flux Offset Control Limits

These Figures are modified to increase the negative ASI limits below 50% power. The licensee has evaluated the effect of the proposed new limits on the Unit 1 Cycle 10 and Unit 2 Cycle 8 transient analyses, margin to fuel centerline melt limits, margin to DNB limits, margin to LOCA PLHGR limit, core power versus planar radial peaking factor LCO, TM/LP LSSS, and core power versus integrated radial peaking factor LCO and has determined that acceptable results are obtained. The changes are, therefore, acceptable for Unit 1 Cycle 10 and Unit 2 Cycle 8.

7.0 SUMMARY

The staff has reviewed the fuel system design, nuclear design, thermal-hydraulic design, and the transient and accident analysis information presented in the Calvert Cliffs Unit 1 Cycle 10 reload submittals. Based on this review, which is described above, the staff concludes that the proposed Cycle 10 reload and associated modified TS are acceptable. This conclusion is further based on the following: (1) previously reviewed and approved methods were used in the analyses; (2) the results of the safety analyses show that all safety criteria are met; and (3) the proposed TS are consistent with the reload safety analyses.

8.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of the facilities' components located within the restricted areas as defined in 10 CFR 20 and changes in surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

9.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 16, 1988

PRINCIPAL CONTRIBUTOR:

L. Kopp

REFERENCES

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2. Attachment to B-88-011 Calvert Cliffs Unit Cycle 10 License Submittal.

3. Appendix to B-88-011 Calvert Cliffs Unit 1 Cycle 10 License Submittal, ANF-88-019.

4. "Improvements to Fuel Evaluation Model," CEN-161(B)-P, Supplement 1-P (proprietary), April 1986.

5. Letter from Scott A. McNeil (NRC) to J. A. Tiernan (BG&E), dated February 4, 1987.

6. Letter from E. J. Butcher (NRC) to A. E. Lundvall, Jr. (BG&E), "Safety Evaluation for Topical Report CENPD-369-P, Revision 1-P," October 10, 1985.

7. Letter from J. A. Tiernam (BG&E) to NRC, "Unit 1 Cycle 10 Response to Request for Additional Information," March 25, 1988.

8. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," CENPD-162-P-A, April 1975.

9. "TORC Code, Verification and Simplified Modeling Methods," CENPD-206-P-A, June 1981.

10. "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," CEN-191(B)-P, December 1981.

11. "Extended Statistical Combination of Uncertainties," CEN-348(B)-P, January 1987.

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13. "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," XN-NF-82-21(P)(A), Rev. 1, September 1983.

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16. "Fuel and Poison Rod Bowing," CENPD-225-P-A, June 1983.

17. Letter from D. M. Crutchfield (NRC) to A. E. Scherer (CE), "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports," July 31, 1986.

18. Letter from S. A. McNeil (NRC) to J. A. Tiernan (BG&E), 'Safety Evaluation of Topical Report CEN-348(B)-P, "Extended Statistical Combination of Uncertainties,"' October 21, 1987.