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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

DOCKET NO. 50-318

INTRODUCTION

By its letters dated February 12, 1988 and February 7, 1989, as supplemented on March 30, April 21, April 25 and May 8, 1989, the Baltimore Gas and Electric Company (BG&E; the licensee) submitted requests to amend its operating license and Technical Specifications (TS) for Calvert Cliffs Unit No. 2 to support operation for a ninth cycle at a 100% rated core power of 2700 MWt. Cycle 9 will have a 24 month cycle length as did the previous cycle. The Unit 1 Cycle 10 design is the reference design for Unit 2 Cycle 9.

The March 30, 1989 supplement provided additional information concerning core burnup expected for Cycle 9 while the April 21 and 25, 1989 letters added further discussion of the potential effects of intermediate core loading configurations with higher enrichment fuel. Finally, the May 8, 1989 submittal provided camera-ready copies of the proposed TS changes.

These supplemental submittals did not affect the proposed TS changes noticed in the Federal Register (54 FR 14714) on April 12, 1989 and did not affect the staff's proposed determination that no significant hazards would result from these changes.

The NRC staff has reviewed the applications and the supporting documents and has prepared the following evaluation of the fuel design, nuclear design, thermal-hydraulic design, transient and accident analysis, and TS changes.

2.0 EVALUATION OF FUEL DESIGN

2.1 Fuel Assembly Description

The Cycle 9 core consists of 217 fuel assemblies. Ninety-two fresh (unirradiated) Batch L assemblies manufactured by Combustion Engineering (CE) will replace previously irradiated assemblies. Four of these Batch L assemblies will contain erbia (Er_2O_3) as the burnable poison material instead of B_4C. These assemblies will have an initial enrichment of 3.81 weight percent U-234 and are being placed in the Cycle 9 core as lead test assemblies to aid in qualification of the use of erbia as an acceptable burnable poison for use in 24-month cycle cores.

2.2 Mechanical Design

The mechanical design of the CE Batch L reload fuel is essentially identical to the Batch M fuel approved for use in Cycle 10 of Calvert Cliffs Unit 1 with the following exceptions:

(a) The fuel rod plenum spring in the Batch L fuel has been redesigned to maximize the available rod internal void volume. This modification helps reduce high end of cycle (EOC) internal gas pressures.

(b) The overall length of the Batch L B_4C burnable poison rod has been increased so that the poison rod length is now the same as the fuel rod length. This allows the same type cladding tube to be used for both rod types.

(c) The size and number of crimp holes in the upper end of each of the five guide tubes of each Batch L assembly have been modified. This design change allows the fuel assembly upper end fitting guide tube posts to be reusable if the assembly must be disassembled for fuel rod reconstitution.

(d) The Batch L lower end fitting, flow hole configuration has been modified to a new smaller hole, more debris-resistant design. In this design, nine small, chamfered holes replace each of the larger holes in the reference cycle design, thus forming a smaller diameter flow path more restrictive to the intrusion of reactor coolant system debris into the fuel assembly.

The staff finds these design changes have been adequately considered in all aspects of the nuclear, mechanical, thermal-hydraulic, and transient safety analyses for Cycle 9. In addition, all CE fuel to be loaded for the Cycle 9 core was reviewed to ascertain that adequate shoulder gap clearance exists. Analyses were performed with NRC approved models and the licensee concluded that all shoulder gap and fuel assembly length clearances are adequate for Cycle 9. Thus, the changes are acceptable.

2.3 Thermal Design

The thermal performance of the CE fuel in Cycle 9 was evaluated using the FATES3B fuel evaluation model which has been approved by the NRC for BG&E licensing submittals. The licensee analyzed a composite standard fuel rod that enveloped the various fuel batches in Cycle 9. The analysis modeled the power and burnup levels representative of the peak rod at each burnup interval and bounds the erbia bearing fuel rods. Although the burnup range analyzed for the peak rod was greater than that expected at the end of Cycle 9, approximately one percent of the fuel rods will achieve burnups greater than the 52,000 MWD/T value approved for CE fuel if Cycles 8 and 9 are operated to their maximum burnups. In order to limit the maximum internal hot gas pressure throughout Cycle 9 to a value that is less than the nominal reactor coolant system (RCS) pressure of 2250 psia, the allowable peak linear heat generation rate (PLHGR) in the peak rod was reduced to 15.2 Kw/ft. The licensee has confirmed that the maximum relative power density of any rod which exceeds 52,000 MWD/T will be at least 30% below the single rod peak in the core and the maximum pressure within any of these rods also will not reach the nominal RCS pressure.

Based on the above analysis performed using NRC approved methodology and on the fact that evaluations performed by the licensee have shown that the four erbia lead test assemblies are thermally compatible with the other fuel assemblies and meet all the appropriate fuel thermal design criteria required by the staff, the staff concludes that the thermal design of the Cycle 9 fuel is acceptable.

3.0 EVALUATION OF NUCLEAR DESIGN

3.1 Fuel Management

The Cycle 9 core consists of 217 fuel assemblies, each having a 14 by 14 fuel rod array. The highest U-235 enrichment occurs in the non-erbia bearing Batch L fuel assemblies which contain an assembly average enrichment of 4.30 weight percent U-235. The Calvert Cliffs fuel storage facilities have been analyzed for storage of fuel of maximum enrichment of 5.0 weight percent U-235 and, therefore, the fresh Batch L assemblies are acceptable from a fuel storage aspect.

The Calvert Cliffs refueling procedures allowed placement of fuel assemblies in intermediate positions during core alterations. During an analysis of refueling configurations, the licensee discovered that the potential existed for placing several 4.3 weight percent fresh highly reactive fuel assemblies together and losing shutdown margin below the minimum required 5 percent, or in the extreme, having an inadvertent criticality. Since a significant amount of reactivity could be added to a subcritical geometry by placing a single fresh 4.3 weight percent fuel assembly in certain intermediate locations, subscritical multiplication may no longer provide adequate warning of an approach to critically. Taking no credit for control element assemblies (CEAs) and assuming that the minimum refueling boron concentration exists, it was determined that an inadvertent criticality could occur under the extreme conditions of grouping a number of such highly reactive assemblies together. Therefore, BG&E issued a written report, in accordance with 10 CFR Part 21, "Reporting of Defects and Noncompliance," to the NRC on March 15, 1989, detailing the potential loss of shutdown margin that could occur during refueling.

As documented in a letter from George C. Creel (BG&E) to NRC dated April 21, 1989, BG&E has revised their refueling procedures to ensure that a fuel assembly will not be placed in an intermediate position during core alternations without first verifying its potential reactivity. Fuel will only be positioned in intermediate core locations which will contain fuel of equal or greater reactivity in the final core configuration. In order to prevent inadvertent misplacement of fuel assemblies, the revised procedures also require operators to identify a fuel assembly as new or irradiated by its appearance. An irradiated assembly will have an oxide layer and appear black while new fuel will not. In addition, all core locations will be verified prior to insertion and after insertion. These additional procedures should ensure that a new fuel assemblies are placed where they are specified to be positioned. The staff finds this to be an acceptable means for assuring that the required shutdown margin is maintained at all times during refueling and that an inadvertent criticality will not occur.

The Cycle 9 core will use a low-leakage fuel management scheme. With the proposed loading, the Cycle 9 reactivity lifetime for full power operation is expected to be 20,650 MWD/T based on a Cycle 8 length of 18,300 MWD/T. The analyses presented by the licensee will accommodate a Cycle 9 length between 20,400 MWD/T and 21,500 MWD/T based on Cycle 8 lengths between 17,000 MWD/T and 19,000 MWD/T.

3.2 Power Distribution

Hot full power (HFP) fuel assembly relative power densities are provided in the reload analysis report for beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) unrodded confirguations. Radial power distributions at BOC and EOC are also provided for CEA Bank 5, the lead regulating bank, fully inserted. These distributions are characteristic of the high burnup end of the Cycle 8 shutdown window and tend to increase the radial power peaking in the Cycle 9 core. The four Batch LE lead test assemblies (Er_2O_3) were calculated to have maximum pin power peaking at least 10% lower than the maximum pin peaking in the core under all expected Cycle 9 operating conditions. The distributions were calculated with approved methods and include the increased power peaking that 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 9. Therefore, the predicted Cycle 9 power distributions are acceptable.

3.3 Moderator Temperature Coefficient (MTC)

The Cycle 9 moderator temperature coefficient (MTC) positive limit varies linearly from $+0.3\times10^{-4}$ delta rho/°F at 100% power to $+0.7\times10^{-4}$ delta rho/°F at 70% power and below. The negative limit is -2.7×10^{-4} delta rho/°F. The NRC has previously expressed concern about positive MTC effects 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 NRC 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 agreed to this commitment and has predicted a negative MTC at HFP equilibrium conditions ranging from -0.04×10^{-4} delta rho/°F at BOC to -2.3×10^{-4} delta rho/°F at EOC for Cycle 9.

3.4 Control Requirements

The CEA worths and shutdown margin requirements for Unit 2 Cycle 9 are most limiting at EOC. The licensee's assessed shutdown margin requirements for Cycle 9 are based on the results of the EOC, hot zero power (HZP), steamline break event. After consideration of all reactivity uncertainties and biases, a worst case assessment for Cycle 9 results in a 0.4% delta rho margin in excess of the proposed 5.0% delta rho EOC TS requirement. Therefore, sufficient CEA worth is available to accommodate the reactivity effects of the steamline break event at the worst time during Cycle 9 allowing for the most reactive CEA stuck in the fully withdrawn position. Based on a review of the licensee's analysis, the staff finds the control requirements are satisfactorily met.

3.5 Incore Monitoring

The incore detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in the same manner as those for the reference cycle. However, changes to the on-line incore limiting condition for operation (LCO) monitoring system have been proposed for Cycle 9.

The INCA computer code, which is currently used for power distribution surveillance, would be replaced by the CECOR 3.3 code. Since CECOR 3.3 calculates a full core solution and can be used to obtain 3-dimensional power distribution data as compared to INCA which gives an octant solution, this change should result in more accurate monitoring. CECOR 3.3 would be used on-line and is virtually identical to and gives the same results as the CECOR 2.0 code which is currently utilized off-line. The use of CECOR 3.3 in an on-line network to monitor compliance with the linear heat rate (LHR) and departure from nucleate boiling (DNB) LCO TS would incorporate present separate monitoring systems, specifically the alarm Limit System for LHR and the Better Axial Shape Selection System (BASSS) for DNB, into one linked system. BASSS would use the CECOR 3.3 calculated radial peaking factor instead of the presently required TS limit. CECOR 3.3 would also supply BASSS with a core average axial shape index based on the full core solution which would be used together with the Bank 5 rod position and the unrodded radial peaking factor to calculate the available DNB overpower margin and alarms as at present. BASSS would provide the capability to monitor the LCO on total planar radial peaking factor, F_{xy}^{T} and total integrated radial peaking factor, F_{r}^{T}. If the TS limits on these were exceeded during normal operation, BASSS would activate an alarm and would calculate the proper trade-off with maximum allowed power to ensure that the axial power distribution and thermal margin/low pressure (TM/LP) trips remain conservative. The proposed TS are worded to support either the full core CECOR 3.3 measured power distribution or the present INCA octant measured power distributions.

The staff finds these proposed changes acceptable and appropriately included in the Cycle 9 safety analyses and proposed TS modifications.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

4.1 DNBR Analysis

Steady state Departure from Nucleate Boiling Ratio (DNBR) analyses of Cycle 9 at the rated power level of 2700 MWt have been performed using the approved core thermal-hydraulic code TORC and CETOP, and the CE-1 critical heat flux correlation. The cycle specific TORC and CETOP models used for designing Cycle 9 account for the small flow hole configuration used in the lower end fitting of the Batch L fresh fuel. Engineering hot channel factors and conservatisms are combined statistically with other uncertainty methods using the approved Extended Statistical Combination of Uncertainties (ESCU) methodology to arrive at an equivalent DNBR limit of 1.15 at a 95/95 probability/confidence level. The Cycle 9 DNBR analyses bound the four Batch LE lead test assemblies without crediting ESCU methods since the maximum single fuel rod peak is at least 10% below the maximum single fuel rod peak in the core. Therefore, the hottest fuel rod in the core is never located in any of the lead test assemblies at any point throughout Cycle 9. The DNBR analysis for Cycle 9 is, therefore, acceptable.

4.2 Fuel Rod Bowing

The fuel rod bow penalty accounts for the adverse impact on minimum DNBR of random variations in spacing between fuel rods. The methodology for determining rod bow penalties for Calvert Cliffs was based on NRC approved methods. 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

All of the key input parameters for the transient and accident analyses are bounded by (or are conservative with respect to) those of the reference cycle (Unit 1 Cycle 10) with the exception of the following:

1. Cycle 9 Batch L fuel utilities a small flow hole debris resistant design.

2. The maximum auxiliary feedwater (AFW) flow assumed in the safety analyses was increased from 1300 gpm for the reference cycle to 1550 gpm for Cycle 9 in anticipation of larger measured AFW flow.

3. The maximum assumed number of plugged U-tubes per steam generator was increased to 500 for all but the small break LOCA Cycle 9 analyses.

The licensee has determined by reanalysis or reevaluation that the results of all events affected by these input parameter changes remain bounded by those of the reference cycle. Based on a review of this analysis, the staff concludes that the non-LOCA transient and accident events for Cycle 9 are bounded by the reference analyses and the results of the non-LOCA safety analyses are acceptable.

5.2 LOCA Events

The large break LOCA has been reevaluated for Cycle 9 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. Blowdown hydraulic calculations have shown that the slight increase in pressure drop due to the Batch L debris-resistant fuel design is insignificant. In addition, a reduction of 260 gpm in low pressure safety injection (LPSI) flow was assumed in order to establish an increased margin between analysis flow values and actual measured flow values. The licensee confirmed that there remains adequate safety injection flow to maintain a full downcomer with the reduced flow and, therefore, the reduction in LPSI flow does not affect the results. Since all other fuel related parameters for Cycle 9 remain bounded by those of the reference cycle, the staff concludes that the large break LOCA is bounded by the reference cycle analysis. Therefore, operation of Unit 2 Cycle 9 at a PLHGR of 15.5 Kw/ft and a power level of 2754 MWt (102% of 2700 MWT) is in compliance with the 10 CFR 50.46 acceptance criteria. The allowable PLHGR is being decreased to 15.2 Kw/ft for Cycle 9 to accommodate the more limiting fuel performance data associated with this second 24 month cycle in Unit 2.

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 Unit 2 Cycle 9 results. Unlike the large break LOCA analysis which considered 500 plugged tubes per steam generator, the SBLOCA considered only 150 plugged tubes per steam generator. This effect as well as the effect of the reduction in LPSI flow and the incorporation of the debris-resistant fuel design were evaluated by the licensee and were found to have no significant impact on the SBLOCA. Therefore, acceptable SBLOCA ECCS performance is also demonstrated at a PLHGR of 15.5 Kw/ft and a reactor power level of 2754 MWt. The allowable PLHGR is being decreased to 15.2 Kw/ft for Cycle 9 to accommodate the more limiting fuel performance data associated with this second 24 month cycle in Unit 2. The staff has reviewed the licensee's analysis and concludes that it complies with 10 CFR 50.46 and the results are acceptable.

5.3 Fuel Handling Accident

The licensee evaluated the potential impact of an increase in core fuel enrichment to 4.35 weight percent U-235 and concluded that the design basis accidents (DBAs), previously analyzed by the licensee in the Calvert Cliffs Updated Final Safety Analysis Report (UFSAR) (Revision 8, August 24, 1988), bound any potential radiological consequences of DBAs that could result with 4.35 weight percent U-235.

The NRC staff reviewed the licensee's submittals as compared to NUREG/CR 5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors" and the Standard Review Plan for Pressurized Water Reactors. The staff agrees that the only potential DBA consequences that could result from the use of higher enrichment fuel, would be an increase in the thyroid doses that could result from a postulated design basis fuel handling accident. NUREG/CR 5009 estimates that the I-131 fuel gap actively in the peak fuel rod with 60,000 MWD/MT burnup (5.292 weight percent U-235) could be as high as 12%. This value is approximately 20% higher than the value normally used by the staff in evaluating fuel handling accidents (Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage facilities for Boiling and Pressurized Water Reactors").

The NRC staff, consequently, reevaluated the fuel handling accident for the Calvert Cliffs Unit 2 facility with an increase in iodine gap activity in the fuel damaged for a postulated fuel handling accident. The following table provides 1) handling accident thyroid doses presented in the NRC Safety Evaluation Report for Calvert Cliffs, dated August 28, 1972, and 2) the increased (by 20%) thyroid doses resulting from fuel initially enriched to 5.292 weight percent U-235, with burnup to 60,000 MWD/MT. The resulting doses are small fractions of the applicable regulatory requirements at 10 CFR Part 100.

Based upon the above, the staff concludes that 1) the bounding doses potentially increased are the thyroid doses at the Exclusion Area and Low Population Zone boundaries resulting from postulated fuel handling accidents, 2) these doses remain well within the 300 Rem thyroid exposure guideline values set forth in 10 CFR Part 100, and 3) the small calculated increase is not significant. Consequently, the staff has determined that the proposed increase in the core fuel enrichment limit to 4.35 weight percent U-235 is acceptable.

Table 1

Thyroid Doses as a Consequences of DBA Fuel Handling Accident

Exclusion Area		Low Population Zone	
Thyroid Dose (Rem)		Thyroid Dose (Rem)	
A*	B**	A*	B**
5.	6.	2.	2.4

*A Original SER dose (8/28/72)
**B Doses with higher enrichment and burnup (5.29%; 60,000 MWD/MT)

6.0 TECHNICAL SPECIFICATIONS

The following paragraphs summarize the proposed changes to the TS requested to support operation of Unit 2 Cycle 9. Some of these changes have already been implemented for Unit 1 Cycle 10. Others involve specific Unit 2 Cycle 9 changes such as the use of the new on-line incore LCO monitoring system discussed in Section 3.5 of this SER.

a. Thermal Margin/Low Pressure Trip Setpoint

TS Figures 2.2-2 and 2.2-3 are modified to accommodate the implementation of the ESCU methodology used originally in the thermal-hydraulic analysis of the reference cycle (Unit 1 Cycle 10) and now in the Unit 2 Cycle 9 reload design. Use of the ESCU methodology requires changes to the coefficients of the PVAR equation presented in Figures 2.2-2 and 2.2-3, ensuring agreement of the Thermal Margin/Low Pressure trip setpoint with the Unit 2 Cycle 9 DNBR analysis. Therefore, these changes are acceptable.

b. Axial Power Distribution Trip Loss

TS Figure 2.2-1, "Peripheral Axial Shape Index, Y_I vs Fraction of Rated Thermal Power," is modified by increasing the acceptable operation region below 70% Rated Thermal Power and by shaping the negative Axial Shape Index (ASI) side to accommodate the increased core average linear heat generation rate (CALHGR) of Unit 2 Cycle 9 which is the second 24-month cycle in Unit 2. The CALHGR is increased for Cycle 9 because of the increased number of B_4C shims. These modifications were considered in the Unit 2 Cycle 9 reload safety analyses as well as in the setpoint analysis and acceptable results were obtained. Consequently, these changes are acceptable.

TS Basis B 2.2.1 is modified to identify the DNBR Specified Acceptable Fuel Design Limit (SAFDL) value of 1.15. This change is resultant from use of the ESCU methodology and is identical to that approved for the reference cycle and is acceptable.

c. Shutdown Margin

TS 3.1.1.1 and 4.1.1.1.1 (with the addition of new Figure 3.1-1b) are modified to establish a shutdown margin limit for Unit 2 Cycle 9 as a function of time in cycle. The actual required shutdown margin varies throughout the cycle due to changes in the core such as fuel depletion, boron concentration, and moderator temperature. The TS references the new Figure 3.1-1b, which provides a shutdown margin limit line varying from 3.5% delta rho required shutdown margin at BOC to 5.0% delta rho shutdown margin at EOC. The Unit 2 Cycle 9 safety analyses consider the limits of Figure 3.1-1b and are bounded by the reference cycle. This change is identical to the shutdown margin change approved for the reference cycle and as so, is acceptable.

TS Bases B 3/4.1.1.1 and B 3/4.1.1.2 are modified to explain the basis of the proposed change in the shutdown margin limit. This change is acceptable.

d. CEA Group Insertion Limits

TS Figure 3.1-2, "CEA Group Insertion Limits vs Fraction of Allowable Thermal Power for Existing RCP Combination," is modified to allow greater insertion of Group 5 CEA's between 90% and 100% power. Specifically, Group 5 insertion is changed from 25% at 100% power to 35% insertion between 90% and 100% power. This change allows an additional 13.5 inches of CEA insertion of the lead regulating group for enhanced control of axial power oscillations. The Unit 2 Cycle 9 reload safety analyses assume the proposed limits of Figure 3.1-2. This change is identical to the Figure 3.1-2 change approved in the reference cycle and is acceptable.

e. Peaking Factors

TS Limiting Condition for Operation (LCO) 3.1.3.1 and Figure 3.1-3, "Allowable Time to Realign CEA Versus Initial Total Integrated Radial Peaking Factor," are modified to incorporate an increase in the maximum allowed F_r^T from 1.65 to 1.70. This is done to accommodate the increased neutron flux peaking associated with the second 24-month cycle for Unit 2. The setpoint analysis performed in the support of Unit 2 Cycle 9 considers this proposed change in F_r^T. Also, this TS reflects the use of the CECOR 3.3/BASSS network which is considered in the Unit 2 Cycle 9 setpoint analysis. Therefore, the changes are acceptable. The text of TS Basis B3/4.1.3 is modified to reflect these changes.

Figure 3.2-3b, "Total Planar Radial Peaking Factor vs N," is modified to indicate a reduction in its acceptable value region due to a reduction in the 100% power F_{xy}^T value of 1.54 to 1.50. This reduction is needed to accommodate the increased core average linear heat generation rate of the second Unit 2 24-month reload core design with its higher number of B_4C shims. This change is acceptable as the setpoint analysis takes credit for this modification in demonstrating acceptable results for Cycle 9.

Implementation of CECOR 3.3/BASSS as the on-line incore LCO monitoring system necessitated the proposed changes in TS 3.2.2.1, 3.2.3, 4.2.2.1.2, 4.2.2.1.3, 4.2.2.1.4, 4.2.2.2.2, 4.2.2.2.3, 4.2.2.2.4, 4.2.3.2, 4.2.3.3, 4.2.3.4, 4.2.5.3 and B 3/4.1.3 to ensure they adequately consider the CECOR 3.3/BASSS system. This implementation was shown to be acceptable in Section 3.5 of this SER.

LCO 3.2.3 and TS Figure 3.2-3c, "Total Integrated Radial Peaking Factor vs Allowable Fraction of Rated Thermal Power" are modified to reflect the increase in F_r^T from 1.65 to 1.70 to accommodate the increased nuclear flux peaking associated with this second 24-month cycle for Unit 2 and implementation of the CECOR 3.3/BASS on-line incore monitoring system. The Unit 2 Cycle 9 setpoint analyses, which was performed using NRC approved methodology, supports this change in F_r^T and, hence, the change is acceptable.

f. Peak Linear Heat Rate

Figure 3.2-1, "Allowable Peak Linear Heat Rate vs Burnup," is modified to indicate the reduced allowable peak linear heat rate (APLHR), for Unit 2 Cycle 9, from 15.5 Kw/ft to 15.2 Kw/ft. The APLHR is reduced to maintain the maximum Cycle 9 internal pin pressure below reactor coolant system pressure of 2250 psia. This change in APLHR is considered in the Unit 2 Cycle 9 fuel performance analysis. Also, the LOCA analyses have shown that a PLHR as high as 15.5 Kw/ft complies with the acceptance criteria. Therefore, this change is acceptable.

g. DNB Parameters

The text of 3.2.5 and Table 3.2-1 is modified by changing the phrase "core power" to "thermal power" to maintain consistency in the TS. This change is administrative in nature and is therefore acceptable.

TS Basis B 3/4.2.5 is modified to specify the new DNBR SAFDL value if 1.15 which results from use of the ESCU methodology. The change is acceptable.

h. Auxiliary Feedwater Flow Rate

TS Basis B 3/4.7.1.2 is modified by increasing the maximum allowed auxiliary feedwater flow from 1300 gpm to 1550 gpm. An evaluation of the effects of increasing this flow was performed by the licensee and it was determined that the results on the safety analyses for Unit 2 Cycle 9 are bounded by previously reported and approved analyses. The change is acceptable.

i. Axial Shape Index

TS Figure 3.2-2, "Linear Heat Rate Axial Flux Offset Control Limits" and Figure 3.2-4, "DNB Axial Flux Offset Control Limits," 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 2 Cycle 9 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. Thus, these changes are acceptable.

j. Core Enrichment

TS 5.3.1 is modified to indicate an increase in the maximum enrichment for a reload core from 4.1 weight percent (w/o) to 4.35 w/o U-235. This change is proposed to allow the higher enriched Unit 2 Cycle 9 second 24-month cycle reload core. All aspects of the Cycle 9 reload core design consider the proposed higher enrichment. The Unit 1 and 2 fuel storage facilities have been approved for storage of fuel of maximum enrichment of 5.0 w/o U-235. This approach was provided in the NRC Safety Evaluation dated January 10, 1990. Therefore, this change is acceptable.

7.0 STARTUP TEST PROGRAM

The startup testing program proposed for Unit 2 Cycle 9 is similar to that used in the reference cycle. However, a change is proposed in the manner in which core symmetry is verified by using incore power distribution monitoring with only minimal CEA symmetry testing. This monitoring would be done using measured power distributions generated by the new CECOR 3.3 system which does the solution in full core as contrasted to the previous INCA system whose solution is for an eighth core. The staff has approved this new monitoring system in Section 3.5 of this SER. The licensee has also performed an analysis using approved methodologies which demonstrates that the incore monitoring in full core, in conjunction with minimal CEA symmetry testing, will detect any significant fuel assembly misloadings. Therefore, the staff finds this proposed change in core symmetry confirmation to be acceptable.

8.0 SUMMARY

The staff has reviewed the fuel system design, nuclear design, thermal-hydraulic design, startup test program, and the transient and accident analysis information presented in the Calvert Cliffs Unit 2 Cycle 9 reload submittals. In addition, the modifications made to the refueling procedures as a result of the licensee's 10 CFR 21 notification concerning potential loss of shutdown margin were also reviewed.

Based on this review, which is described above, the staff concludes that the proposed Unit 2 Cycle 9 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.

9.0 INTENT

The intent of the proposed changes is to authorize Unit 2 Cycle 9 operation.

10.0 STATE CONSULTATIONS

The Commission made a proposed determination that this amendment involves no significant hazards consideration, which was published in the Federal Register on April 12, 1989 (54 FR 14714). No hearing requests or intervention petitions were received. The State of Maryland was consulted on this matter and had no comments on the determination.

11.0 ENVIRONMENTAL CONSIDERATION

This amendment, with the exception of changes in the enrichment limits for fuel located in the Unit 2 core, involves a change in the installation or use of the facility's components located within the restricted areas as defined in 10 CFR 20 and changes in surveillance requirement. The staff has determined that this amendment involves 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 this amendment involves no significant hazards considerations and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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 this amendment.

As for the changes in TS core enrichment limits, pursuant to 10 CFR 5.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on January 30, 1989 (53 FR 4352). Based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect upon the quality of the human environment.

12.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 10, 1990

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