June 30, 1987

Docket Nos. 50-317
and 50-318

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Mr. J. A. Tiernan
Vice President - Nuclear Energy
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Dear Mr. Tiernan:

SUBJECT: REVISED SAFETY EVALUATION SUPPORTING AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-69

By letter dated May 4, 1987, the Commission issued Amendment No. 108 to Facility Operating License No. DPR-69 and the related Safety Evaluation in support of cycle 8 operations for the Calvert Cliffs Nuclear Power Plant, Unit 2.

Subsequently, our staff was informed through your letter of May 26, 1987 that the transient analysis input values for the Bank 5 Inserted radial peaking factors as provided in the February 6, 1987 reload application were erroneous. However, you asserted that the correct radial peaking factor values were used in the transient analysis performed for the cycle 8 reload.

To properly reflect the correction of these erroneous radial peaking factor values, revision of the Safety Evaluation (SE) issued in support of Amendment No. 108 was necessitated.

A copy of the revised SE is enclosed. All changes are indicated by marginal bars. The staff has determined that these corrections to the original SE do not change our previous conclusions regarding the acceptability of the Technical Specification changes approved in Amendment No. 108 to Facility Operating License No. DPR-69.

Sincerely,

Scott Alexander McNeil, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosure:
As stated

cc: See next page

PDI-1	PDI-1	SRXB	SRXB	OGC	PDI-1
CVogan	SMcNeil	DFieno	MHodges		RCapra
6/29/87	6/29/87	7/2/87	7/1/87	7/6/87	6/30/87

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Calvert Cliffs Nuclear Power Plant

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

REVISED

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 108

TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-318

1.0 INTRODUCTION

By letter dated February 6, 1987 (Ref. 1), the Baltimore Gas and Electric Company (BG&E or the licensee) made application to amend the Technical Specifications of Calvert Cliffs Unit 2. The proposed changes would modify the Technical Specifications to permit operation for an eighth cycle (Cycle 8). The safety analysis performed and the resulting modifications to the Calvert Cliffs Unit 2 Technical Specifications are described in the document attached to Reference 1. Additional information was provided by BG&E in References 2 and 3 in response to staff requests for additional information (Refs. 4 and 5). Revisions to the safety analysis presented in Reference 1 were submitted on April 7, 1987 (Ref. 13). These revisions were due to an actual end of Cycle 7 burnup of 13,580 MWD/MTU which exceeded the value of 13,300 MWD/MTD assumed in the original safety analysis.

The licensee submitted a final camera-ready copy of the previously requested Technical Specifications on April 17, 1987 (Reference 14).

On May 4, 1987, the Commission issued Amendment No. 108 to Facility Operating License No. DPR-69 and the related Safety Evaluation for Calvert Cliffs Nuclear Power Plant, Unit No. 2 (Ref. 16).

The licensee identified to our staff through the letter dated May 26, 1987, (Ref. 17) that the values of the transient analysis input data for the Bank 5 Inserted radial peaking factors as provided in the February 6, 1987 application were erroneous. However, the licensee asserted that the correct radial peaking factor values were utilized in the transient analyses performed for Unit 2 Cycle 8. Additionally, the description of the major characteristics of the Unit 2 Cycle 8 transient analyses was flawed in that it incorrectly stated due to omission that the values of the Bank 5 Inserted radial peaking factors were within the bounds of the reference cycle.

Neither the supplements to the February 6, 1987 submittal made prior to the issuance of the May 4, 1987 Safety Evaluation nor the submittal of May 26, 1987 affected the proposed TS change noticed in the Federal Register on March 25, 1987 nor did they affect the staff's proposed no significant hazards determination.

The safety analysis for the previous seventh cycle of operation at Calvert Cliffs Unit 2 is being used by the licensee as the reference cycle for the proposed eighth cycle of operation. Cycle 7 operated with no anomalies that would affect Cycle 8. Cycle 7 operated with core reactivity, power distributions and peaking factors closely following calculated values. Where conditions are identical or limiting in the seventh cycle safety analysis, the staff's previous Safety Evaluation (Ref. 6) continues to apply.

The staff's revised evaluation of the safety analysis for the Calvert Cliffs Unit 2 Cycle 8 reload follows.

2.0 DESCRIPTION OF THE CYCLE 8 CORE

The fuel management scheme is based on a low leakage core design with loading pattern, fresh fuel enrichments, and burnable poison loading and distribution chosen to provide a Cycle 8 length of approximately 24 months. The Calvert Cliffs Unit 2 core contains 217 fuel assemblies, each of which is a 14X14 array containing 176 fuel/burnable poison rods and five control rod guide tubes. The loading pattern is as follows: Eighty-eight fresh fuel assemblies, designated batches K, K*, and K/, are distributed in a nearly checkerboard pattern (red squares) throughout the core with 8 batch K fuel assemblies located on the core flats and 8 batch K fuel assemblies located on other core periphery locations. Sixty batches J and J* and 69 batches H and H* irradiated fuel assemblies are shuffled to new locations in the core in the black squares of the near checkerboard pattern. Most of the peripheral core locations are occupied by the thrice-burned batch H fuel assemblies. The initial enrichment of batches K, K*, and K/ is 4.08 weight percent uranium-235; while batches J and H have initial enrichments of 4.05 weight percent uranium-235; and while batches J* and H* have initial enrichments of 3.40 weight percent uranium-235. Batches K* and K/ fuel assemblies contain burnable poison rods. Batch K* contains 12 burnable poison rods in a fuel assembly while batch K/ contains 8 burnable poison rods in a fuel assembly. The burnable poison is boron-10 at a loading of 0.036 grams per inch.

Reactivity control for Cycle 8 will be provided by 77 full-length control element assemblies (CEAs), 72 assemblies with burnable poison rods containing B_4C admixed with Al_2O_3 , and soluble boron in the coolant. Seventy-six of the CEAs contain the neutron absorber B_4C in each finger. The centrally located CEA has Al_2O_3 in the four outer fingers and B_4C in the center finger. The core loading pattern of fuel enrichments, burnup, and burnable poison will assure a negative moderator temperature coefficient (MTC) at hot, full power, equilibrium xenon conditions (see page 1-1 of attachment to Reference 1) per agreement with the staff.

The safety analysis provided in the reload report transmitted by Reference 1 demonstrates the safe operation of Calvert Cliffs Unit 2 throughout Cycle 8. This safety analysis is based on a licensed core power level of 2700 MWt, an end of Cycle 7 burnup range of 11,300 MWD/MTU to 13,700 MWD/MTU, and a Cycle 8 length which varies from 19,800 to 21,500 MWD/MTU including a coastdown in inlet temperature to 537°F and a coastdown in power to about 75%. The following sections discuss the staff's evaluation of the safety analysis for the approximate 24-month, low-leakage fuel management for Calvert Cliffs Unit 2 Cycle 8.

3.0 EVALUATION OF THE FUEL SYSTEM DESIGN

The mechanical design of the batch K reload fuel is identical to the batch J fuel previously inserted in Calvert Cliffs Unit 2 with the exception that some batch K fuel assemblies contain burnable poison rods. The burnable poison rods are nearly identical in design to the burnable poison rods of the batch G/ fuel assemblies that operated in the reference cycle. All fuel to be loaded for the Cycle 8 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 8. The replacement CEA to be used in the center location of the core will have nearly the same reconstituted features as the replacement CEAs installed in the reference cycle, except that all five fingers of the Cycle 8 replacement CEA are reconstituted.

All fuel assemblies in Cycle 8 will have stainless steel sleeves inserted in the guide tubes to prevent guide tube wear. A modified short sleeve design was used in batch J fuel of the reference cycle and will also be used in the fresh batch K fuel assemblies. The licensee states that this short sleeve design is considered to be the permanent modification for mitigating guide tube wear and will be used in all future Calvert Cliffs reloads.

The thermal performance of Cycle 8 was evaluated using the FATES3B fuel evaluation model (Ref. 7). The staff issued an SER (Ref. 8) approving the use of FATES3B for BG&E licensing submittals. The licensee analyzed a composite, standard fuel pin that enveloped the various fuel batches in Cycle 8. The analysis modeled the power and burnup levels representative of the peak pin at each burnup interval. The burnup range was greater than that expected at the end of Cycle 8.

Based on its review of the information discussed above, the staff concludes that the evaluation of the fuel system design is acceptable for the Calvert Cliffs Unit 2 Cycle 8 reload.

4.0 EVALUATION OF THE NUCLEAR DESIGN

To support Cycle 8 operation of Calvert Cliffs Unit 2, the licensee has provided analyses using analytical methods and design bases established in licensing topical reports that have been approved by the NRC. The licensee has provided a comparison of the core physics parameters for Cycles 7 and 8 as calculated with these approved methods. There are differences in the neutronics parameters compared between Cycle 7 and 8. These differences can be attributed to differences in cycle lengths, burnable poison loadings and distribution, and the fuel enrichments and loading pattern. Using the Cycle 8 physics and parameters, the licensee evaluated seventeen transients and accidents. The licensee determined that Cycle 8 parameters were conservative when compared to analyses previously accepted for thirteen transients and accidents. Three transients and accidents were reanalyzed and a fourth was reevaluated. The staff's review of the four events will be discussed in a later section.

The control rod worths and shutdown margin requirements at the most limiting time for the Cycle 8 nuclear design, that is, for the end of the cycle (EOC), are presented in Table 5-2 of Reference 1 and in response to Question 1 of Reference 4 (Ref. 2). Table 5-2 is based on an EOC, hot zero power (HZP), steamline break accident. At EOC8, the reactivity worth with all control rods inserted is 9.4% delta K/K. An allowance of 1.7% delta K/K is made for the stuck CEA which yields the worst results for the EOC HZP steamline break accident. An allowance of 2.0% delta K/K is made for control rod insertion in accordance with the power dependent insertion limit (PDIL). The calculated scram worth is the total control rod worth less the worth of the stuck control rod and less the worth of control rod insertion to the PDIL and is 5.7% delta K/K. Deducting 0.7% delta K/K for physics uncertainty and bias yield a net available scram worth of 5.0% delta K/K. Since the Technical Specification shutdown margin is 4.5% delta K/K, a margin of 0.5% delta K/K exists in excess of the Technical Specification shutdown margin. Therefore, sufficient control rod worth is available to accommodate the reactivity effects of the steamline break event at the worst time in core life allowing for the most reactive control rod stuck in the full withdrawn position and allowing for physics uncertainties and bias. In addition, the response to Question 1 of Reference 4 indicates acceptable shutdown margin for the Cycle 8 core in going from hot, full-power conditions to shutdown conditions, including the effects of the worst stuck rod. The staff has reviewed the calculated control rod worth and uncertainties in these worth based upon comparisons of calculations with measurements in Combustion Engineering (CE) licensing topical reports and startup reports for various CE plants. On the basis of this review, 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 control rod that results in the worst reactivity condition for an EOC, HZP steamline break accident.

The licensee has made a number of changes in the nuclear design of Cycle 8. The changes are as follows:

1. An increase in cycle lifetime to 24 months with a core burnup between 18,900 to 20,400 MWD/MTU (dependent upon Cycle 7 burnup) including a coastdown in inlet temperature to 537°F and a coastdown in power to about 75%.

2. The use of additional burnable poison rods to aid in reactivity control and to assure a negative MTC at hot, full-power, equilibrium xenon conditions,

3. A change in the prototype, centrally located CEA whose central element now contains B_4C.

4. The axial shape index (ASI) limits have been modified to accommodate the 24 month cycle, however, the PDIL remains the same as for Cycle 7.

5. The positive MTC limit above 70% power is being increased from $+0.2X10^{-4}$ delta K/K/°F at 70% power, to a value which varies linearly from $+0.3X10^{-4}$ delta K/K/°F at 100% power to $+0.7X10^{-4}$ delta K/K/°F at 70% power, and

6. The use of the coarse mesh/fine mesh codes ROCS/DIT rather than PDQ for the generation of incore detector constants for use in determining the measured core power distribution and the use of MC instead of PDQ for the core analysis of fuel pin by fuel pin data.

The ROCS code with the MC module has been approved by the staff. Based on its review, the staff concludes that these changes in the Cycle 8 nuclear design are acceptable since the nuclear design, including transient and accident analyses, and the resulting Technical Specifications for Cycle 8 includes the effect of these changes calculated with approved methods.

5.0 EVALUATION OF THE THERMAL-HYDRAULIC DESIGN

The thermal hydraulic parameters of Cycle 8 are essentially the same as those of the reference cycle. The small difference in the total heat transfer area, due to a larger number of burnable poison rods in Cycle 8, are accounted for in the analysis. The safety limit DNBR value for Cycle 8 remains at 1.21. This safety limit involves the use of the TORC and CETOP computer codes, the CE-1 critical heat flux correlation, and a 0.006 DNBR penalty for rod bow effects on DNBR for burnups up to 45,000 MWD/MTU. The licensee states that no rod bow DNBR penalty is necessary for burnups above 45,000 MWD/MTU because of the reduced power capabilities which provide adequate margin to offset increased rod bow penalties. The DNBR limit of 1.21 is based on the statistical combination of uncertainties methodology.

Based on its review and because previously used and approved methods were employed in the analysis, the staff concludes that the thermal-hydraulic design of Cycle 8 is acceptable.

6.0 EVALUATION OF NON-LOCA TRANSIENT AND ACCIDENT ANALYSIS

For the non-LOCA safety analysis, 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 except for the boron dilution, full-length CEA drop, CEA ejection, steamline break events, and the Bank 5 Inserted radial peaking factors. For the events with bounded input parameters the results and conclusions of the reference cycle are valid for Cycle 8 and, therefore, no reanalysis has been performed. The licensee states that the use of the FATES3B (Ref. 7) fuel evaluation model had no significant impact on generated data.

The licensee reevaluated the steamline break accident to accommodate the implementation of 24 month cycles and determined that the results were less limiting than those previously reported. The licensee evaluated the full length CEA drop event because of a revised Doppler reactivity curve, changed to accommodate 24 month cycles, and found that the changed Doppler reactivity had no significant impact on the event. Since previous conclusions and analyses remain bounding for these two events, no results were presented in the reload report by the licensee.

The licensee evaluated the effect of increasing the Bank 5 Inserted radial peaking factors. They determined that this increase would have no significant impact on the safety analyses for the design basis events.

The boron dilution event was reanalyzed for Modes 2, 3, and 4. For Modes 1, 5, and 6 the boron dilution event is bounded by the reference cycle. The time to criticality was determined by using the same expression as used in the reference calculation made for Calvert Cliffs Unit 1 Cycle 8. High critical boron concentrations and low inverse boron worths were assumed by the licensee in calculating conservative values of the calculated time to criticality.

The calculated times are still large compared to the acceptance criterion (45-55 minutes vs. 15 minutes). The staff concludes, therefore, that the analysis of the boron dilution event for Modes 2, 3, and 4 is acceptable.

The zero power CEA ejection accident was reanalyzed to establish generic values for the CEA ejection worth and the post-ejected radial power peak. The neutronic parameters for the analysis were chosen so that the most conservative value occurring during the cycle was chosen for each parameter. The NRC approved analysis methodology was used to analyze this accident. The variable high power trip was assumed to initiate at 40% of full rated power. The 40% power setpoint includes 10% for uncertainty. The staff concludes, therefore, that the analysis methods and assumptions employed by the licensee are acceptable.

The results presented indicate that the peak average fuel rod enthalpy was less than 200 calories per gram. This meets the staff's acceptance criterion of 280 calories per gram and is, therefore, acceptable. For estimating radiological doses, the licensee used a fuel failure criterion that has not been accepted by the staff. In response to Question 2 of Reference 4 (Ref 2), the licensee performed radiological dose analyses assuming that 10% of the fuel fails. The staff believes that the 10% failed fuel amount represents a conservative failed fuel estimate for CEA ejection accidents. The licensee's response indicates that the radiological dose consequences for the CEA ejection accident are well within (25%) the 10 CFR Part 100 exposure guidelines.

In response to Question 3 of Reference 4 (Ref. 2), the licensee presented information in the fuel loading error event. The licensee states that the reference analysis conclusions are generic and still applicable to the Cycle 8 reload. A significant misloading would result in a quadrant power tilt as determined by the incore detectors or as perturbations in the measured core power distributions. The core power distribution, as measured with the incore detector system, must agree closely with calculated power distributions to meet acceptance criteria. The licensee further states that plant procedures provide a high confidence that the core is properly loaded.

Contrary to staff position, the licensee contends that the feedwater line break (FLB) event is not a design basis event (DBE) for Calvert Cliffs Units 1 and 2. To permit Unit 2 Cycle 8 operation, prior to resolution of this issue, the licensee presented a response (Refs. 3 and 15) to a staff request for additional information on the FLB event (Ref. 5). The FLB event was reanalyzed with cycle specific input parameters. The full power MTC was taken as $+0.3X10^{-4}$ delta K/K/°F. For Cycle 8 the FLB analysis is bounded by the results of previously approved NRC reviews of the event (Ref. 9). In particular, the results for the reanalyzed FLB event indicate that the peak reactor coolant system (RCS) pressure is less than 110% of the design RCS pressure. As the FLB analysis was performed and is bounded by the results of previously approved NRC reviews, the staff concludes that Unit 2 Cycle 8 operation should not be constrained by resolution of this issue. The issue of whether or not the FLB event is a DBE will be addressed by the licensee outside of the Cycle 8 reload review, but no later than March 31, 1988 as committed by the licensee in Refs. 3 and 15.

Based on the review described above, the staff concludes that the non-LOCA transient and accident evaluations presented by the applicant are acceptable.

7.0 EVALUATION OF ECCS ANALYSIS

The large break loss of coolant accident (LOCA) has been reanalyzed for Cycle 8 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 8 analysis was performed with the 1985 CE evaluation model which was approved in Reference 10. The analysis for the reference cycle (Cycle 7) utilized a previously approved CE evaluation model. The methodology, except for model differences, was the same for the analysis of Cycle 8 as it was for Cycle 7. The Cycle 8 analysis methodology used, in addition, the recently approved (Ref. 8) FATES3B fuel evaluation model (Ref. 7).

The Cycle 8 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 8 and the reference cycle. Table 8.1-2 presents the results of the analysis for the limiting break for Cycle 8 and the reference cycle. It is apparent from the results presented in Tables 8.1-1 and 8.1-2 that, in spite of reactor core and model differences, there are only small differences between Cycle 8 and the reference cycle for the limiting break. The results for the limiting Cycle 8 break show that (1) the peak clad temperature is 1903°F which is well below the acceptance criterion of 2200°F and (2) the maximum local and core wide oxidation are 3.3% and less than 0.51%, respectively, and these are well below the acceptance criteria of 17% and 1% respectively. Since the Cycle 8 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 8 at an allowable peak linear heat generation rate 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 2 Cycle 8 results. Since the acceptance criteria for the SBLOCA are met, the operation of Cycle 8 at an allowable peak linear heat generation rate of 15.5 kW/ft, with up to 100 plugged tubes per steam generators, is acceptable.

8.0 STARTUP TESTING

The startup testing program has been changed from that of the reference cycle program. The changes include the elimination of the 50% power measurements and the implementation of supplemental power distribution measurements at other power levels. These changes will reduce startup testing time. The staff has reviewed this testing program and the response to Question 4 to Reference 4 (Ref. 2). This testing program conforms to the scope of the ANS 19.6.1 standard on reload startup physics tests for pressurized water reactors (Ref. 11) which has been reviewed and approved by the staff (Ref. 12). The staff concludes, therefore, that this startup test program is acceptable since it will enable the licensee to confirm that the as-loaded core conforms to the Cycle 8 nuclear design.

9.0 TECHNICAL SPECIFICATIONS

As indicated in the staff's evaluation of the nuclear design, provided in Section 4, the operating characteristics of Cycle 8 were calculated with approved methods. The proposed Technical Specifications 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 loading and distribution chosen to provide a cycle length of 24 months. The staff concludes that the Technical Specification changes proposed by the licensee in the Cycle 8 reload report are acceptable. Each proposed Technical Specification change is discussed below.

1. Figure 2.2-1 Peripheral Axial Shape Index

   Figure 2.2-1 is modified to reduce the acceptable operation region to accommodate the implementation of a 24 month, low leakage fuel cycle. The setpoint analysis uses the modified results given by Figure 2.2-1 and the licensee has determined that acceptable results are obtained for Calvert Cliffs Unit 2 Cycle 8. The changes to Figure 2.2-1 are, therefore, acceptable.

2. Technical Specification 3/4.1.1.1 Shutdown Margin

   The shutdown margin is being increased for T_{avg} greater than 200°F from 3.5% delta k/k to 4.5% delta k/k. The analysis of the steamline break accident, which is limiting at HZP EOC conditions, supports this change. The changes to Technical Specification 3/4.1.1.1 are, therefore, acceptable.

3. Technical Specification 3.1.1.4 Moderator Temperature Coefficient

   The MTC limit above 70% power is being raised from $-0.2X10^{-4}$ delta K/K/°F to a value which varies linearly from $+0.3X10^{-4}$ delta K/K/°F at 100% power to $+0.7X10^{-4}$ delta K/K/°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. The Cycle 8 analysis has included this changed MTC. Based on these considerations, the proposed changes to Technical Specification 3.1.1.4 are, therefore, acceptable.

4. Bases 3/4.1.1.1 and 3/4.1.1.2 Shutdown Margin

   The shutdown margin for T_{avg} greater than 200°F is being changed from 3.5% delta k/k to 4.5% delta k/k. These changes to the Bases are acceptable for the same reasons as stated in item 2 above.

10.0 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 2 Cycle 8 reload submittals. Based on this review, which is described above, the staff concludes that the proposed Cycle 8 reload and associated modified Technical Specifications 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.

3. The proposed Technical Specifications are consistent with the reload safety analyses.

11.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the 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 not significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and that there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §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 the amendment.

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

Dated:

PRINCIPAL CONTRIBUTOR:

D. Fieno

12.0 REFERENCES

1. Letter from J. A. Tiernan (BG&E) to NRC, dated February 6, 1987.

2. Letter from J. A. Tiernan (BG&E) to NRC, dated March 27, 1987.

3. Letter from J. A. Tiernan (BG&E) to NRC, dated March 17, 1987.

4. Letter from S. A. McNeil (NRC) to J. A. Tiernan (BG&E), dated March 12, 1987.

5. Letter from S. A. McNeil (NRC) to J. A. Tiernan (BG&E), dated February 19, 1987.

6. Letter from David H. Jaffe (NRC) to A.E. Lundvall, Jr. (BG&E), dated November 21, 1985.

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

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

9. Letter from D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BG&E), dated January 10, 1983.

10. Letter from D. M. Crutchfield (NRC) to A. E. Scherer (CE), dated July 31, 1986.

11. American National Standard - "Reload Startup Physics Tests for Pressurized Water Reactors," ANS 19.6.1, Draft 9, December 1984.

12. Letter from Robert E. Carter (NRC) to Tawfik M. Raby (Secretary of N-17 Standards Committee), dated April 18, 1985.

13. Letter from J. A. Mihalcik (BG&E) to NRC, dated April 7, 1987.

14. Letter from M. E. Bowman (BG&E to NRC, dated April 17, 1987

15. Letter from J. A. Mihalcik (BG&E) to NRC, dated March 25, 1987.

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

17. Letter from J. A. Mihalcik (BG&E) to NRC, dated May 26, 1987.