UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 232
License No. DPR-53

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated September 1, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Facility Operating License No. DPR-53 is hereby amended to read as follows:

2. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 232 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Guy S. Vissing for

Sheri R. Peterson, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 15, 1999

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 208
License No. DPR-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated September 1, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Facility Operating License No. DPR-69 is hereby amended to read as follows:

2. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 208 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Guy S. Vissing for

Sheri R. Peterson, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 15, 1999

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 232 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 208 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1.1-1 and -2	1.1-1 and -2
2.0-1 and —2	2.0-1 and —2
2.0-3	
3.3.1-9	3.3.1-9
3.3.1-11	3.3.1-11
3.3.6-3	3.3.6-3
3.4.1-1 and —2	3.4.1-1 and —2
3.7.1-4	3.7.1-4
3.8.1-12	3.8.1-12
3.8.1-14	3.8.1-14
5.0-33 through -38	5.0-33 through -38

1.0 USE AND APPLICATION

1.1 Definitions

-----------------------------------NOTE-----------------------------------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AXIAL SHAPE INDEX (ASI)	ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core. $$ASI = \frac{\text{lower} - \text{upper}}{\text{lower} + \text{upper}}$$
AZIMUTHAL POWER TILT (T_q)	AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric core locations.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall

1.1 Definitions

include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

Analog Channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY, including alarm and/or trip functions.

Bistable Channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1.1-1.

2.1.1.2 In MODES 1 and 2, the peak linear heat rate (LHR) shall be ≤ 22.0 kW/ft.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.0 SAFETY LIMITS (SLs)

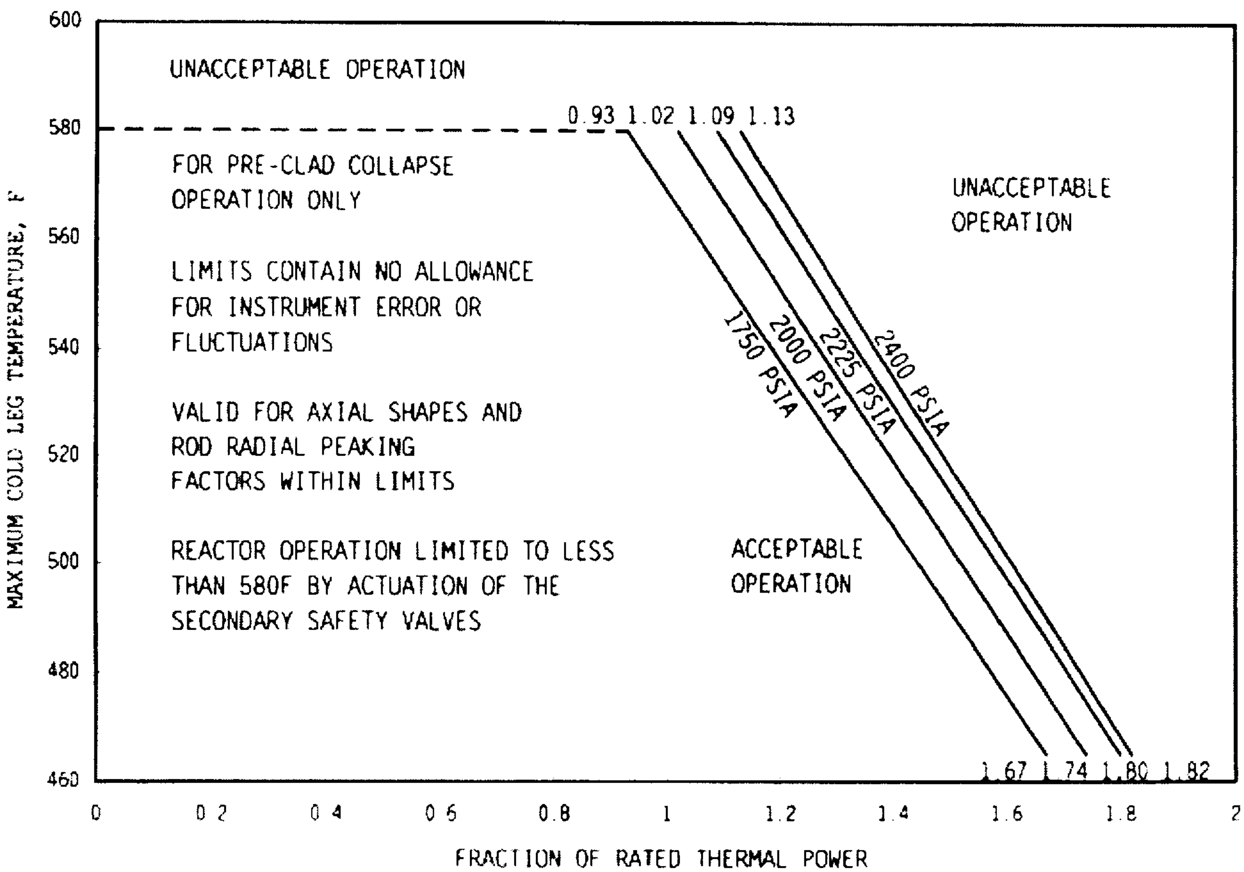

Figure 2.1.1-1
Unit 1 and Unit 2 Reactor Core Thermal Margin Safety Limit

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Power Level-High	1, 2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.8 SR 3.3.1.9	≤ 10% RTP above current THERMAL POWER but not < 30% RTP nor > 107% RTP
2. Rate of Change of Power-High[a]	1, 2	SR 3.3.1.1[f] SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8	≤ 2.6 dpm
3. Reactor Coolant Flow-Low[b]	1, 2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9	≥ 92% of Design Flow
4. Pressurizer Pressure-High	1, 2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.8 SR 3.3.1.9	≤ 2400 psia
5. Containment Pressure-High	1, 2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.8 SR 3.3.1.9	≤ 4.0 psig

Table 3.3.1-1 (page 3 of 3)
Reactor Protective System Instrumentation

FUNCTION	MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9b. Asymmetric Steam Generator Transient (ASGT)[b]	1, 2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9	≤ 135 psid
10. Loss of Load[d]	1[e]	SR 3.3.1.6 SR 3.3.1.7	NA

(a) Bistable trip unit may be bypassed when NUCLEAR INSTRUMENT POWER is < 1E-4% RTP or > 12% RTP. Bypass shall be automatically removed when NUCLEAR INSTRUMENT POWER is ≥ 1E-4% RTP and < 12% RTP.

(b) Bistable trip unit may be bypassed when NUCLEAR INSTRUMENT POWER is < 1E-4%. Bypass shall be automatically removed when NUCLEAR INSTRUMENT POWER is ≥ 1E-4% RTP. During testing pursuant to LCO 3.4.16, trips may be bypassed below 5% RTP.

(c) Bistable trip unit may be bypassed when steam generator pressure is < 785 psig. Bypass shall be automatically removed when steam generator pressure is ≥ 785 psig.

(d) Bistable trip unit may be bypassed when NUCLEAR INSTRUMENT POWER is < 15% RTP. Bypass shall be automatically removed when NUCLEAR INSTRUMENT POWER is ≥ 15% RTP.

(e) Trip is only applicable in MODE 1, NUCLEAR INSTRUMENT POWER ≥ 15% RTP.

(f) CHANNEL CHECK only applies to Wide Range Logarithmic Neutron Flux Monitor.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG-LOVS instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.2	Perform CHANNEL CALIBRATION with setpoint Allowable Values as follows: 1. Transient Degraded Voltage Function ≥ 3630 V and ≤ 3790 V; Time Delay: ≥ 7.6 seconds and ≤ 8.4 seconds; 2. Steady State Degraded Voltage Function ≥ 3820 V and ≤ 3980 V Time Delay: ≥ 97.5 seconds and ≤ 104.5 seconds; and 3. Loss of voltage Function ≥ 2345 V and ≤ 2555 V Time Delay: ≥ 1.8 seconds and ≤ 2.2 seconds at 2450 V.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2200 psia;

b. RCS cold leg temperature (T_c) ≤ 548°F; and

c. RCS total flow rate ≥ 340,000 gpm.

APPLICABILITY: MODE 1.

------------------------------NOTE------------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RCS DNB parameter(s) not within limits.	A.1	Restore parameter(s) to within limit.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure ≥ 2200 psia.	12 hours
SR 3.4.1.2	Verify RCS cold leg temperature ≤ 548°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate ≥ 340,000 gpm.	12 hours
SR 3.4.1.4	Verify measured RCS total flow rate is within limits.	24 months

Table 3.7.1-2
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING[1] (psig)
Steam Generator #1	Steam Generator #2	
RV-3992	RV-4000	935-995
RV-3993	RV-4001	935-995
RV-3994	RV-4002	935-1035
RV-3995	RV-4003	935-1035
RV-3996	RV-4004	935-1050
RV-3997	RV-4005	935-1050
RV-3998	RV-4006	935-1050
RV-3999	RV-4007	935-1050

(1) Lift settings for a given steam line are also acceptable if any two valves lift between 935 and 995 psig, any two other valves lift between 935 and 1035 psig, and the four remaining valves lift between 935 and 1050 psig.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	Verify interval between each sequenced load block is within ± 10% of design interval for the load sequencer.	31 days
SR 3.8.1.9	-------------------NOTE------------------- All DG starts may be preceded by an engine prelube period. -- Verify each DG starts from standby condition and achieves, in ≤ 10 seconds, voltage > 4060 V and frequency > 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 4060 V and ≤ 4400 V and frequency of > 58.8 Hz and ≤ 61.2 Hz.	184 days
SR 3.8.1.10	Verify manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit.	24 months
SR 3.8.1.11	-------------------NOTE------------------- Momentary transients outside the load and power factor limits do not invalidate this test. -- Verify each DG, operating at a power factor of ≤ 0.85, operates for ≥ 60 minutes while loaded to ≥ 4000 kW for DG 1A and ≥ 3000 kW for DGs 1B, 2A, and 2B.	24 months
SR 3.8.1.12	Verify each DG rejects a load ≥ 500 hp without tripping.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	-------------------NOTE------------------- All DG starts may be preceded by an engine prelube period. -- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated Engineered Safety Feature actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. maintains steady state voltage ≥ 4060 V and ≤ 4400 V, 4. maintains steady state frequency of ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	24 months

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1	SHUTDOWN MARGIN
3.1.3	Moderator Temperature Coefficient
3.1.4	CEA Alignment
3.1.6	Regulating Control Element Assembly Insertion Limit
3.2.1	Linear Heat Rate
3.2.2	Total Planar Radial Peaking Factor
3.2.3	Total Integrated Radial Peaking Factor
3.2.5	AXIAL SHAPE INDEX
3.3.1	RPS Instrumentation - Operating
3.9.1	Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CENPD-199-P, Latest Approved Revision, "C-E Setpoint Methodology: C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986

2. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II," December 1979

5.6 Reporting Requirements

3. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2," January 1980

4. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Calvert Cliffs Units 1 and 2," March 1980

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

6. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated June 24, 1982, Unit 1 Cycle 6 License Approval (Amendment No. 71 to DPR-53 and SER)

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

8. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated October 21, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-348(B)-P, Extended Statistical Combination of Uncertainties"

9. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986

10. CENPD-162-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"

11. CENPD-207-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-Uniform Axial Power Distribution"

5.6 Reporting Requirements

12. CENPD-206-P-A, Latest Approved Revision, "TORC Code, Verification and Simplified Modeling Methods"

13. CENPD-225-P-A, Latest Approved Revision, "Fuel and Poison Rod Bowing"

14. CENPD-266-P-A, Latest Approved Revision, "The ROCS and DIT Computer Code for Nuclear Design"

15. CENPD-275-P-A, Latest Approved Revision, "C-E Methodology for Core Designs Containing Gadolinia - Urania Burnable Absorbers"

16. CENPD-382-P-A, Latest Approved Revision, "C-E Methodology for Core Designs Containing Erbium Burnable Absorbers"

17. CENPD-139-P-A, Latest Approved Revision, "C-E Fuel Evaluation Model Topical Report"

18. CEN-161-(B)-P-A, Latest Approved Revision, "Improvements to Fuel Evaluation Model"

19. CEN-161-(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model," April 1989

20. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated February 4, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-161-(B)-P, Supplement 1-P, Improvements to Fuel Evaluation Model"

21. CEN-372-P-A, Latest Approved Revision, "Fuel Rod Maximum Allowable Gas Pressure"

22. Letter from Mr. A. E. Scherer (CE) to Mr. J. R. Miller (NRC), dated December 15, 1981, LD-81-095, Enclosure 1-P, "C-E ECCS Evaluation Model Flow Blockage Analysis"

5.6 Reporting Requirements

23. CENPD-132, Supplement 3-P-A, Latest Approved Revision, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS"

24. CENPD-133, Supplement 5, "CEFLASH-4A, a FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985

25. CENPD-134, Supplement 2, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core," June 1985

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

27. CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977

28. Letter from Mr. R. L. Baer (NRC) to Mr. A. E. Scherer (CE), dated September 6, 1978, "Evaluation of Topical Report CENPD-135, Supplement 5"

29. CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977

30. CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977

31. Letter from Mr. K. Kniel (NRC) to Mr. A. E. Scherer (CE), dated September 27, 1977, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P"

5.6 Reporting Requirements

32. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977

33. Letter from Mr. C. Aniel (NRC) to Mr. A. E. Scherer, dated April 10, 1978, "Evaluation of Topical Report CENPD-138, Supplement 2-P"

34. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. J. R. Miller (NRC) dated February 22, 1985, "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application"

35. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated May 20, 1985, "Safety Evaluation Report Approving Unit 1 Cycle 8 License Application"

36. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. R. A. Clark (NRC), dated September 22, 1980, "Amendment to Operating License No. 50-317, Fifth Cycle License Application"

37. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated December 12, 1980, "Safety Evaluation Report Approving Unit 1, Cycle 5 License Application"

38. Letter from Mr. J. A. Tiernan (BG&E) to Mr. A. C. Thadani (NRC), dated October 1, 1986, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Request for Amendment"

39. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated July 7, 1987, Docket Nos. 50-317 and 50-318, Approval of Amendments 127 (Unit 1) and 109 (Unit 2)

40. CENPD-188-A, Latest Approved Revision, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"

5.6 Reporting Requirements

41. The power distribution monitoring system referenced in various specifications and the BASES, is described in the following documents:

    i. CENPD-153-P, Latest Approved Revision, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System"

    ii. CEN-119(B)-P, "BASSS, Use of the Incore Detector System to Monitor the DNB-LCO on Calvert Cliffs Unit 1 and Unit 2," November 1979

    iii. Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated February 7, 1989, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket 50-318, Request for Amendment, Unit 2 Ninth Cycle License Application"

    iv. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated January 10, 1990, "Safety Evaluation Report Approving Unit 2 Cycle 9 License Application"

42. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated May 11, 1995, "Approval to Use Convolution Technique in Main Steam Line Break Analysis - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90897 and M90898)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.