AUG 19 1981

Docket File
DCS- MS-016

Docket No. 50-317
50-318

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. A. E. Lundvall, Jr.
Vice President - Supply
Baltimore Gas & Electric Company
P.O. Box 1475
Baltimore, Maryland 21203

Dear Mr. Lundvall:

The Commission has issued the enclosed Amendment Nos. 57 and 39 to Facility Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated March 23, 1981.

The amendments revise the Appendix A Technical Specifications (TS) to reduce the monitoring and reporting requirements now specified to detect core barrel movement.

The requirements of the TS 3.4.11 and 4.4.11 for monitoring and reporting core barrel movement were required after the Palisades plant experienced excessive barrel motion. The vendor, Combustion Engineering (CE), devised a generic design modification which has been made to both Calvert Cliffs units. Considerable operating experience since has not revealed any excessive core barrel motion nor has inspection revealed excessive wear of the core barrel to reactor vessel interface nor in the flange area. Under those conditions we find that the proposed action statement reporting requirement changes are acceptable.

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We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

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We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

A copy of the Notice of Issuance is enclosed.

Sincerely,

Original signed by:

Charles M. Trammell, III for

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 57 to DPR-53
2. Amendment No. 39 to DPR-69
3. Notice of Issuance

cc: w/enclosures
See next page

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Baltimore Gas and Electric Company

cc:
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Calvert County Board of County Commissioners
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Mr. Ralph E. Architzel
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cc w/enclosure(s) and incoming
dated: 3/23/81
Administrator, Power Plant Siting Program
Energy and Coastal Zone Administration
Department of Natural Resources
Tawes State Office Building
Annapolis, MD 21204

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 57
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated March 23, 1981, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 57, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell III

for Robert A. ~~Clark~~, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: August 19, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 57

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 4-29
3/4 4-30
B 3/4 4-12

BASES

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.3 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when on or more of the RCS cold legs are $\leq$ 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq$ 46°F (34°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for the ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 CORE BARREL MOVEMENT

This specification is provided to ensure early detection of excessive core barrel movement if it should occur. Core barrel movement will be detected by using four excore neutron detectors to obtain Amplitude Probability Distribution (APD) and Spectral Analysis (SA). Baseline core barrel movement Alert Levels and Action Levels will be confirmed during each reactor startup test program following a core reload.

Data from these detectors is to be reduced in two forms. Root mean square (RMS) values are computed from the APD of the signal amplitude. These RMS magnitudes include variations due both to various neutronic effects and internals motion. Consequently, these signals alone can only provide a gross measure of core barrel motion. A more accurate assessment of core barrel motion is obtained from the Auto and Cross Power Spectral Densities (PSD, XPSD), phase (ϕ) and coherence (COH) of these signals. These data result from the SA of the excore detector signals.

A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of any fuel cycle.

3/4.4.12 LETDOWN LINE EXCESS FLOW

This specification is provided to ensure that the bypass valve for the excess flow check valve in the letdown line will be maintained closed during plant operation. This bypass valve is required to be closed to ensure that the effects of a pipe rupture downstream of this valve will not exceed the accident analyses assumptions.

REACTOR COOLANT SYSTEM

CORE BARREL MOVEMENT

LIMITING CONDITION FOR OPERATION

3.4.11 Core barrel movement shall be limited to less than the Amplitude Probability Distribution (APD) and Spectral Analysis (SA) Alert Levels for the applicable THERMAL POWER level.

APPLICABILITY: MODE 1.

ACTION:

a. With the APD and/or SA exceeding their applicable Alert Levels, POWER OPERATION may proceed provided the following actions are taken:

   1. APD shall be measured and processed at least once per 24 hours,

   2. SA shall be measured at least once per 24 hours and shall be processed at least once per 7 days, and

   3. A Special Report, identifying the cause(s) for exceeding the applicable Alert Level, shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days of detection.

b. With the APD and/or SA exceeding their applicable Action Levels, measure and process APD and SA data within 24 hours to determine if the core barrel motion is exceeding its limits. With the core barrel motion exceeding its limits, reduce the core barrel motion to within its Action Levels within the next 24 hours or be in HOT STANDBY within the following 6 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11 Routine Monitoring Core barrel movement shall be determined to be less than the APD and SA Alert Levels by using the excore neutron detectors to measure APD and SA at the following frequencies:

a. APD data shall be measured and processed at least once per 7 days.

b. SA data shall be measured and processed at least once per 31 days.

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. DPR-69

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated March 23, 1981, 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. 39, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell III

for Robert A. ~~Clark~~, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: August 19, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 4-30
3/4 4-31
B 3/4 4-12

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

4.4.10.1.2 Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping - The unencapsulated welds greater than 4 inches in nominal diameter in the main steam and main feedwater piping runs located outside the containment and traversing safety related areas or located in compartments adjoining safety related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition and Addenda through Summer 1976, for Class 2 components.

a. System integrity and baseline data shall be established by performing a 100% volumetric examination of each weld prior to exceeding 18 months of operation.

b. Each weld shall be examined in accordance with the above ASME Code requirements, except that 100% of the welds shall be examined, cumulatively, during each 10 year inspection interval. The welds to be examined during each inspection period shall be selected to provide a representative sample of the conditions of the welds. If these examinations reveal unacceptable structural defects in one or more welds, an additional 1/3 of the welds shall be examined and the inspection schedule for the repaired welds shall revert back to the first 10 year inspection program. If additional unacceptable defects are detected in the second sampling, the remainder of the welds shall also be inspected.

CORE BARREL MOVEMENT

LIMITING CONDITION FOR OPERATION

3.4.11 Core barrel movement shall be limited to less than the Amplitude Probability Distribution (APD) and Spectral Analysis (SA) Alert Levels for the applicable THERMAL POWER level.

APPLICABILITY: MODE 1.

ACTION:

a. With the APD and/or SA exceeding their applicable Alert Levels, POWER OPERATION may proceed provided the following actions are taken:

   1. APD shall be measured and processed at least once per 24 hours,

   2. SA shall be measured at least once per 24 hours and shall be processed at least once per 7 days, and

   3. A Special Report, identifying the cause(s) for exceeding the applicable Alert Level, shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days of detection.

b. With the APD and/or SA exceeding their applicable Action Levels, measure and process APD and SA data within 24 hours to determine if the core barrel motion is exceeding its limits. With the core barrel motion exceeding its limits, reduce the core barrel motion to within its Action Levels within the next 24 hours or be in HOT STANDBY within the following 6 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11 Routine Monitoring Core barrel movement shall be determined to be less than the APD and SA Alert Levels by using the excore neutron detectors to measure APD and SA at the following frequencies:

a. APD data shall be measured and processed at least once per 7 days.

b. SA data shall be measured and processed at least once per 31 days.

REACTOR COOLANT SYSTEM

3/4.12 LETDOWN LINE EXCESS FLOW

LIMITING CONDITION FOR OPERATION

3.4.12 The bypass valve for the excess flow check valve in the letdown line shall be closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the above bypass valve open, restore the valve to its closed position within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.12 The bypass valve for the excess flow check valve in the letdown line shall be determined closed within 4 hours prior to entering MODE 4 from MODE 5.

BASES

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.3 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are $\leq$ 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq$ 46°F (34°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for the ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 CORE BARREL MOVEMENT

This specification is provided to ensure early detection of excessive core barrel movement if it should occur. Core barrel movement will be detected by using four excore neutron detectors to obtain Amplitude Probability Distribution (APD) and Spectral Analysis (SA). Baseline core barrel movement Alert Levels and Action Levels will be confirmed during each reactor startup test program following a core reload.

Data from these detectors is to be reduced in two forms. Root mean square (RMS) values are computed from the APD of the signal amplitude. These RMS magnitudes include variations due both to various neutronic effects and internals motion. Consequently, these signals alone can only provide a gross measure of core barrel motion. A more accurate assessment of core barrel motion is obtained from the Auto and Cross Power Spectral Densities (PSD, XPSD), phase (ϕ) and coherence (COH) of these signals. These data result from the SA of the excore detector signals.

A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of any fuel cycle.

3/4.4.12 LETDOWN LINE EXCESS FLOW

This specification is provided to ensure that the bypass valve for the excess flow check valve in the letdown line will be maintained closed during plant operation. This bypass valve is required to be closed to ensure that the effects of a pipe rupture downstream of this valve will not exceed the accident analyses assumptions.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-317 AND 318

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY

OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 57 and 39 to Facility Operating Licenses Nos. DPR-53 and DPR-69, issued to Baltimore Gas and Electric Company, which revised Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2. The amendments are effective as of the date of issuance.

The amendments reduce the monitoring and reporting requirements now specified to detect core barrel movement.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendments.

For further details with respect to this action, see (1) the application for amendment dated March 23, 1981, (2) Amendment Nos. 57 and 39 to License Nos. DPR-53 and DPR-69, and (3) the Commission's related letter dated August 19, 1981. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D.C. and at the Calvert County Library, Prince Frederick, Maryland. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 19th day of August, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell III

Charles M. Trammell, III, Acting Chief
Operating Reactors Branch #3
Division of Licensing