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DEisenhut
ACRS (16)
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OPA (CMiles)
ORx
TBAbernathy
JRBuchanan

Docket No. 50-318

APR 22 1977

Baltimore Gas and Electric Company
ATTN: Mr. A. E. Lundvall, Jr.
Vice President - Supply
Gas and Electric Building
Charles Center
Baltimore, Maryland 21203

Gentlemen:

The Commission has issued the enclosed Amendment No. 6 to Facility Operating License No. DPR-69 for Calvert Cliffs Nuclear Power Plant Unit No. 2. The amendment is in response to your application dated March 3, 1977, as supplemented by letter dated March 24, 1977, which requested our prompt action to preclude a plant shutdown. The amendment incorporates improvements to the Technical Specifications similar to changes recently reviewed and approved for Unit No. 1 in Amendment No. 20 to Facility Operating License No. DPR-53.

Your proposed Technical Specifications have been modified to include other changes noted during our review of your March 3, 1977 submittal. These modifications are consistent with the Technical Specifications of Unit No. 1 and meet our current requirements. We have discussed these changes with representatives of your staff who agree the changes are necessary for consistency of operation of Unit No. 2 and Unit No. 1 with a common control room. Other minor editorial corrections were also made.

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

4/22/77 3:15 pm
Notified Bob Denton
at plant since all
licensing people
were out. eze

Sincerely,
Original signed by
Paul W. O'Connor for
Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 6 to License No. DPR-69
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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OFFICE	DOR:ORB #2	DOR:ORB #2	DOR:SAS	OELD	DOR:ORB#2
SURNAME	EAReeves:ah	RMDiggs	JMMcGough		DKDavis
DATE	4/15/77	4/15/77	4/15/77	4/21/77	4/22/77

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Region III Office
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Mr. Bernad Fowler
President, Board of County
Commissioners
Prince Frederick, Maryland 20678

cc w/enclosures and cy of
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3/24/77:
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Director, Department of State
Planning
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Baltimore, Maryland 21201
(w/4 cys. of enclosures)

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
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 March 3, 1977, as supplemented by letter dated March 24, 1977, 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 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. 6, 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

Paul W. O'Connor, for

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 22, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 6

FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

1. Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. 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.

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2. Replace page 2.1-1 of the Appendix B Technical Specificatons with the enclosed revised page. A vertical marginal line indicates the area of change.

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 for the various combinations of two, three and four reactor coolant pump operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

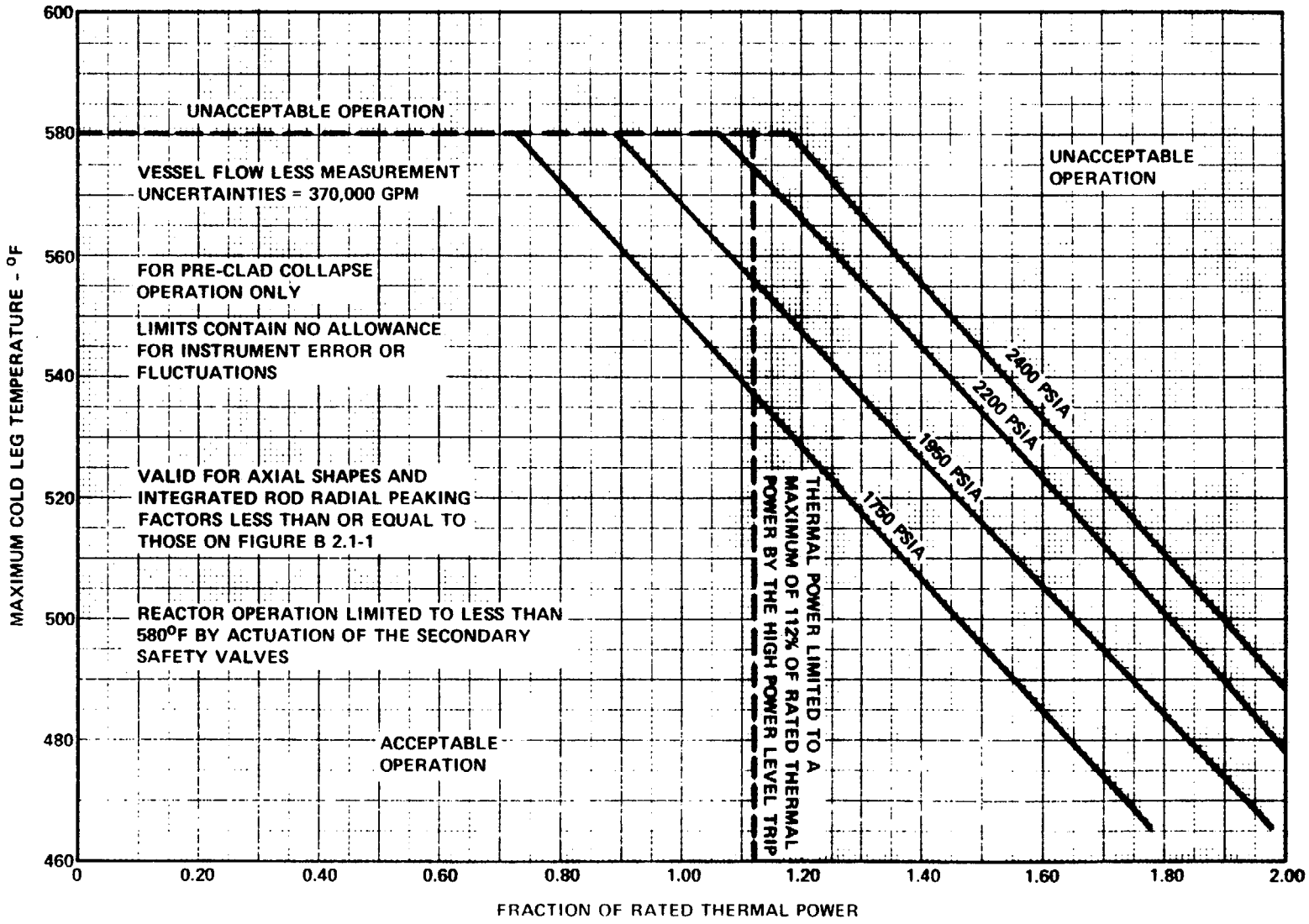

Figure 2.1 - 1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT FOUR REACTOR COOLANT PUMPS OPERATING

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the above required boric acid pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid storage tank and one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water tank with:

1. A minimum contained borated water volume of 9,978 gallons,

2. A minimum boron concentration of 1720 ppm, and

3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is < 35°F.

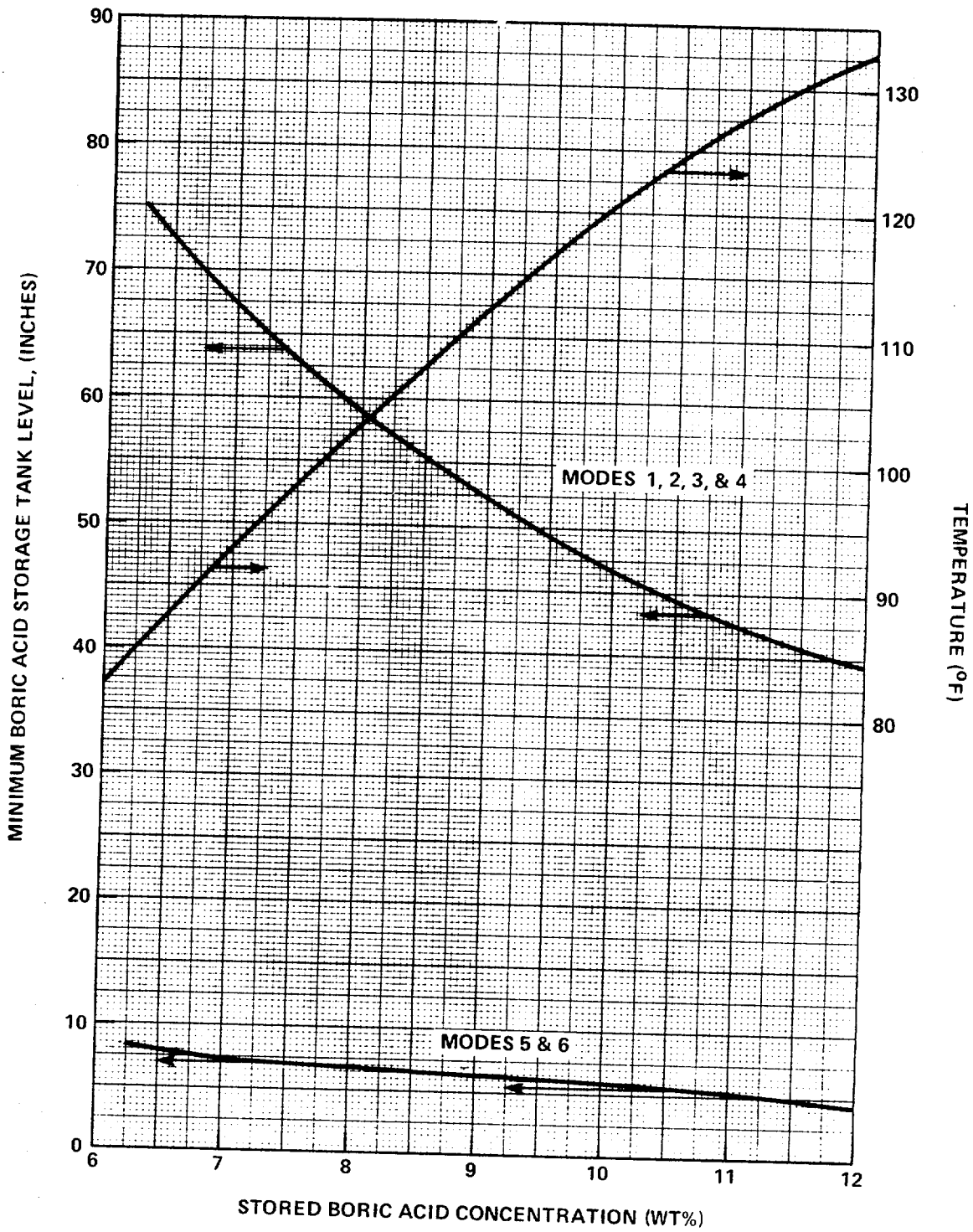

FIGURE 3.1-1

Minimum Boric Acid Storage Tank Volume and Temperature as a Function of Stored Boric Acid Concentration

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following three borated water sources shall be OPERABLE:

a. Two boric acid storage tank(s) and one associated heat tracing circuit per tank with the contents of the tanks in accordance with Figure 3.1-1 and the boron concentration limited to ≤ 8%, and

b. The refueling water tank with:

1. A minimum contained borated water volume of 400,000 gallons,

2. A boron concentration of between 1720 and 2700 ppm, and

3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. At least one per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume in each water source, and

3. Verifying the boric acid storage tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 35°F.

LIMITING CONDITION FOR OPERATION

2. Declare the CEA inoperable. After declaring the CEA inoperable, POWER OPERATION may continue for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided the remainder of the CEAs in the group with the inoperable CEA are aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

g. With more than one full length CEA inoperable or misaligned from any other CEA in its group by 15 inches (indicated position) or more, be in at least at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length CEA shall be determined to be within 7.5 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Motion Inhibit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full length CEA not fully inserted shall be determined to be OPERABLE by movement of at least 7.5 inches in any one direction at least once per 31 days.

4.1.3.1.3 The CEA Motion Inhibit shall be demonstrated OPERABLE at least once per 31 days by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit also prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All part length CEAs shall be withdrawn to at least 132.0 inches.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With a maximum of one PLCEA withdrawn to less than 132.0 inches, either:

a. Withdraw the PLCEA to at least 132.0 inches within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each part length CEA shall be determined withdrawn to at least 132.0 inches by:

a. Verifying the positions of the PLCEAs prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER, and

b. Verifying, at least once per 31 days, that electric power has been disconnected from its drive mechanism by maintaining the feeder breaker open under administrative control.

*See Special Test Exceptions 3.10.1 and 3.10.2.

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.

b. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the determination made in Specification 4.2.1.3.c, and

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the THERMAL POWER does not exceed the value determined by the following relationship:

$$\frac{L}{17.0} \times M$$

where:

1. L is the maximum allowable linear heat rate as determined from Figure 3.2-1 and is based on the core average burnup at the time of the latest incore flux map.

2. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

1. Flux peaking augmentation factors as shown in Figure 4.2-1,

2. A measurement-calculational uncertainty factor of 1.10,

3. An engineering uncertainty factor of 1.03,

4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and

5. A THERMAL POWER measurement uncertainty factor of 1.02.

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometers to make the nuclear power signals agree with calorimetric calculation if absolute difference is > 1.5%. During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to ≤ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Above 15% of RATED THERMAL POWER, adjust "ΔT Pwr Calibrate" potentiometers to null "Nuclear Pwr - ΔT Pwr". During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(5) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
9. CVCS ISOLATION		
West Penetration Room/ Letdown Heat Exchanger Room Pressure - High	≤ 0.5 psig	≤ 0.5 psig

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIS	
Containment Isolation	Not Applicable
d. RAS	
Containment Sump Recirculation	Not Applicable
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 30*/30**
3. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 30*/30**
b. Containment Isolation	≤ 30
c. Containment Fan Coolers	≤ 35*/10**
4. Containment Pressure--High	
a. Containment Spray	≤ 60*/60**
5. Containment Radiation-High	
a. Containment Purge Valves Isolation	≤ 6

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6*.

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to ≤ 80% of RATED THERMAL POWER and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. Power Level-High
   2. Reactor Coolant Flow-Low
   3. Thermal Margin/Low Pressure
   4. Axial Flux Offset

b. With two reactor coolant pumps in opposite loops not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to ≤ 51.1% of RATED THERMAL POWER and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in opposite loops:

   1. Power Level-High
   2. Reactor Coolant Flow-Low
   3. Thermal Margin/Low Pressure
   4. Axial Flux Offset

c. With two reactor coolant pumps in the same loop not in operation, STARTUP and/or continued POWER OPERATION may proceed provided the water level in both steam generators is maintained above the Steam Generator Water Level-Low trip setpoint, the THERMAL POWER is restricted to ≤ 46.8% of RATED THERMAL POWER,

*See Special Test Exception 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

and the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in the same loop:

1. Power Level-High
2. Reactor Coolant Flow-Low
3. Thermal Margin/Low Pressure
4. Axial Flux Offset

MODES 3, 4# and 5#:

Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump.* The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour to accomodate transition between shutdown cooling pump and reactor coolant pump operation, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SURVEILLANCE REQUIREMENTS

4.4.1 The Reactor Protective Instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if switch is made while operating, or

b. Prior to reactor criticality if switch is made while shutdown.

#See Special Test Exception 3.10.5.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Baseline Monitoring Core barrel movement Alert Levels and Action Levels, as determined by APD and SA monitoring of the excore neutron detectors, shall be determined at nominal THERMAL POWER levels of 25%, 50%, 75% and 100% of RATED THERMAL POWER during the reactor startup test program; these Alert Levels and Action Levels shall be reported in a Special Report pursuant to Specification 6.9.2 within 31 days after initially reaching 100% of RATED THERMAL POWER.

4.4.11.2 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 at least once per 24 hours and shall be processed at least once per 7 days.

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

4.4.11.3 Reports The results of all periodic APD and SA monitoring shall be included in the Annual Operating Report for the period in which the monitoring was performed.

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.

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained borated water volume of 400,000 gallons,

b. A boron concentration of between 1720 and 2700 ppm, and

c. A minimum water temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 35°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

1. All penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.4.1, and

2. All equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$, 0.20 percent by weight of the containment air per 24 hours at P_a, 50 psig, or

2. $\leq L_t$ 0.042 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.

b. A combined leakage rate of $\leq$ 0.60 L_a for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either P_a (50 psig) or at P_t (25 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH ONE STEAM GENERATOR

Maximum Number of Inoperable Safety Valves on The Operating Steam Generator	Maximum Allowable Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	40
2	35
3	29

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (± 1%)	ORIFICE SIZE
a. RV-3992/4000	985 psig	R
b. RV-3993/4001	985 psig	R
c. RV-3994/4002	995 psig	R
d. RV-3995/4003	995 psig	R
e. RV-3996/4004	1015 psig	R
f. RV-3997/4005	1015 psig	R
g. RV-3998/4006	1035 psig	R
h. RV-3999/4007	1035 psig	R

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-52-42	S.I. TO LOOP 21B 46'-0"	I	Yes	No
2-52-43	S.I. TO LOOP 22A 48'-0"	I	Yes	No
2-52-44	S.I. TO LOOP 22A 48'-0"	I	Yes	No
2-52-45	S.I. TO LOOP 22A 48'-0"	I	Yes	No
2-52-46	S.I. TO LOOP 22A 48'-0"	I	Yes	No
2-52-47	S.I. TO LOOP 22A 48'-0"	I	Yes	No
2-52-48	S.I. TANK #22A DISCHARGE 56'-0"	I	Yes	No
2-52-48A	S.I. TANK #22A DISCHARGE 56'-0"	I	Yes	No
2-52-49	S.I. TANK #22A DISCHARGE 55'-8"	I	Yes	No
2-52-50	S.I. TO LOOP 22A 48'-0"	I	Yes	No
2-52-50A	S.I. TO LOOP 22A 48'-0"	I	Yes	No
2-52-51	S.I. TO LOOP 22A 42'-0"	I	Yes	No
2-52-52	S.I. TO LOOP 22A 42'-0"	I	Yes	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-52-53	S.I. TO LOOP 22B 48'-0"	I	Yes	No
2-52-53A	S.I. TO LOOP 22B 48'-0"	I	Yes	No
2-52-54	S.I. TO LOOP 22B 48'-0"	I	Yes	No
2-52-55	S.I. TO LOOP 22B 48'-0"	I	Yes	No
2-52-56	S.I. TO LOOP 22B 48'-0"	I	Yes	No
2-52-57	S.I. TANK #22B DISCHARGE 48'-0"	I	Yes	No
2-52-58	S.I. TANK #22B DISCHARGE 56'-0"	I	Yes	No
2-52-58A	S.I. TANK #22B DISCHARGE 56'-0"	I	Yes	No
2-52-59	S.I. TANK #22B DISCHARGE 55'-0"	I	Yes	No
2-52-60	S.I. TO LOOP 22B 46'-0"	I	Yes	No
2-52-61	S.I. TO LOOP 22B 41'-9"	I	Yes	No
2-52-61A	S.I. TO LOOP 22B 41'-9"	I	Yes	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-67-2A	SUPPLY TO REFUELING POOL #21 40'0"	I	Yes	No
2-67-3	RETURN LINE FROM REFUELING POOL #21 26'10"	I	Yes	No
2-67-4	RETURN LINE FROM REFUELING POOL #21 26'10"	I	Yes	No
2-83-1	MS PIPING FROM S/G #22 (UPSTREAM MSIV) 27'	A	No	No
2-83-2	MS PIPING FROM S/G #22 (UPSTREAM MSIV) 27'	A	No	No
2-83-2A	MS PIPING FROM S/G #22 (UPSTREAM MSIV) 27'	A	No	No
2-83-3	MS PIPING FROM S/G #21 (UPSTREAM MSIV) 27'	A	No	No
2-83-4	MS PIPING FROM S/G #21 (UPSTREAM MSIV) 27'	A	No	No
2-83-4A	MS PIPING FROM S/G #21 (UPSTREAM MSIV) 27'	A	No	No
2-83-25	STEAM SUPPLY TO AUX. FEED PUMPS (UPSTREAM MOV 4071) 27'	A	No	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-83-25A	STEAM SUPPLY TO AUX. FEED PUMPS (UPSTREAM MOV 4071) 27'	A	No	No
2-83-26	STEAM TO AUX. FEED PUMPS (DOWN-STREAM MOV 4070) 27'	A	No	No
2-83-28	STEAM TO AUX. FEED PUMPS (UPSTEAM MOV 4070) 27'	A	No	No
2-83-29	MOV 4070 BY-PASS LINE 27'	A	No	No
2-83-29A	MOV 4070 BY-PASS LINE 27'	A	No	No
2-83-30	STEAM SUPPLY TO #22 AFP 27'	A	No	No
2-83-31	M.S. PIPING FROM S/G #22 45'	I	Yes	No
2-83-32	M.S. PIPING FROM S/G #22 45'	I	Yes	No
2-83-33	M.S. PIPING FROM S/G #22 45'	I	Yes	No
2-83-34	M.S. PIPING FROM S/G #22 45'	I	Yes	No
2-83-45	STEAM TO AUX. FEED PUMPS (DOWNSTREAM MOV 4070) 27'	A	No	No
2-83B-1	MSIV #22 HYDRAULIC SUPPLY 27'	A	No	No

3/4.7.10 WATERTIGHT DOORS

LIMITING CONDITION FOR OPERATION

3.7.10 The following watertight doors shall be closed except when the door is being used for normal entry and exit:

a. ECCS Pump Room Doors (4).

b. Service Water Pump Room to Heater Bay Doors (2).

c. Auxiliary Feed Pump Room to Heater Bay Doors (2).

d. Emergency Escape Hatch, Service Water Pump Room from Penetration Room.

e. Main Steam Piping Area from Piping Penetration Room Door.

f. Passage to Main Steam Piping Area Door.

g. Warehouse to Intake Structure Door, Elevation 12'.

h. Outside to Intake Structure Door.

i. Warehouse to Intake Structure Door Elevation 29'.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the above doors open, restore the door to its closed position within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.10 The above watertight doors shall be determined closed at least once per 12 hours.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators (one of which may be a swing diesel generator capable of serving both Unit 1 and Unit 2) each with:

   1. Separate day fuel tanks containing a minimum volume of 375 gallons of fuel,

   2. A common fuel storage system consisting of two independent storage tanks each containing a minimum volume of 18,250 gallons of fuel, and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status

ACTION (Continued)

within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each independent circuit between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 32,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of carbon sample obtained from a test canister demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.); or

2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of ≥ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510-1975 (130°C, 95% R.H.) and the samples are prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reistalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%, and

b) Verifying that the HEPA filter bank remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 32,000 cfm ± 10%.

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.10.5 The reactor coolant circulation requirements of Specification 3.4.1 may be suspended and all reactor coolant pumps and shutdown cooling pumps may be de-energized during the time intervals required 1) for local leak rate testing of containment penetration number 41 pursuant to the requirements of Specification 4.6.1.2.d and 2) to permit maintenance on valves located in the common shutdown cooling suction line or on the shutdown cooling flow control valve (CV-306) provided:

a. No operations are permitted which could cause dilution of the reactor coolant system boron concentration,

b. The xenon reactivity is $\leq$ 0.1% $\Delta k/k$ and is approaching stability, and

c. The SHUTDOWN MARGIN requirement of Specification 3.1.1.2 is verified at least once per 8 hours when no shutdown cooling or reactor coolant pumps are in operation.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving local leak rate testing of containment penetration number 41, maintenance on valves located in the common shutdown cooling suction line, and maintenance on valve CV-306.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The charging pumps shall be verified de-energized and the charging flow paths shall be verified closed at least once per hour.

4.10.5.2 The xenon reactivity shall be determined to be $\leq$ 0.1% $\Delta k/k$ and approaching stability within 1 hour prior to suspending reactor coolant circulation.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 2.4% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} \leq 200^{o}F$, the reactivity transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 3813 gallons of 7.25% boric acid solution from the boric acid tanks or 47,204 gallons of 1720 ppm borated water from the refueling water tank. However, to be consistent with the ECCS requirements, the RWT is required to have a minimum contained volume of 400,000 gallons during MODES 1, 2, 3 and 4.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 1% Δk/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 439 gallons of 7.25% boric acid solution from the boric acid tanks or 9,978 gallons of 1720 ppm borated water from the refueling water tank.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA and to a large misalignment (≥ 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (≤ 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (≥ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is probable that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the accident analyses. Measurement with $T_{avg} \geq$ 515°F and with all reactor coolant pumps operating ensures that the

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.

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 at nominal THERMAL POWER levels of 25%, 50%, 75% and 100% of RATED THERMAL POWER will be determined during the reactor startup test program.

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 the first 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.

environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses.

3/4.7.8 HYDRAULIC SNUBBERS

The hydraulic snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. The only snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The inspection frequency applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during an inspection of these snubbers determines the time interval for the next required inspection of these snubbers. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Observed failures of these sample snubbers will require functional testing of additional units. To minimize personnel exposures, snubbers installed in high radiation zones or in especially difficult to remove locations may be exempted from these functional testing requirements provided the OPERABILITY of these snubbers was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.10 WATERTIGHT DOORS

This specification is provided to ensure the protection of safety related equipment from the effects of water or steam escaping from ruptured pipes or components in adjoining rooms.

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 NO FLOW TESTS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low TERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 COOLANT CIRCULATION

This special test exception permits all forced circulation of reactor coolant to be suspended during required local leak rate testing of containment penetration number 41 (shutdown cooling) and during maintenance on the common shutdown cooling suction line or on the shutdown cooling flow control valve (CV-306).

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,614 ± 460 cubic feet at a nominal T_{avg} of 532°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with a nominal 18 inch center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to ≤ 0.96 with the storage pool filled with unborated water. The k_{eff} of ≤ 0.96 includes the conservative allowances described in Section 9.7.2 of the FSAR.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 63 feet.

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 390 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Reactor Coolant System	500 heatup and cooldown cycles	70°F to 532°F to 70°F
	400 reactor trip cycles	100% to 0% RATED THERMAL POWER
	10 Primary Hydrostatic Tests	3125 psia and 60°F > NDTT
	320 Primary Leak Tests	2500 psia and 60°F > NDTT
Steam Generator	10 Secondary Hydrostatic Tests	1250 psia Secondary Side and temperature ≥ 100°F
	320 Secondary Leak Tests	1000 psia Secondary Side With Primary - Secondary Δp of 820 psi and shell side temperature between 100°F and 200°F

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Thermal Limitations on Condenser Cooling Water Discharge

Objective

The purpose of this specification is to limit thermal stress to organisms which may be entrained in the condenser cooling water of the plant or which may be in the vicinity of the thermal plume.

Specification

2.1.1 Maximum ΔT Across Condenser

This limit applies to the temperature rise of the cooling water as it passes through the condensers. With the exception of periods defined by Specification 4.2, the difference between the flow-weighted hourly average discharge and intake temperatures of the condenser cooling water shall not exceed 10°F during periods of normal operation. During each period when less than the full cooling water flow is available, the flow-weighted hourly average temperature rise shall not exceed 12°F for more than two hours.

Monitoring Requirement

The plant computer will scan the temperature sensors located in each of the 6 condenser inlet pipes and in the discharge tunnel at least once each minute. These temperature readings will be used to calculate a flow-weighted hourly temperature average for both the inlet and outlet condenser cooling water. These averages along with the difference between them will be logged every hour by the computer. During operating periods when the computer is inoperative or under repair, operating condenser inlet and outlet temperature readings shall be logged each hour. The precision of the temperature instrumentation is 0.5°F and the overall accuracy of the system is 1.0°F.

Amendment No. ~~2~~, 6

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 6 TO LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 2

DOCKET NO. 50-318

INTRODUCTION

By application for license amendment dated March 3, 1977, as supplemented by letter dated March 24, 1977, Baltimore Gas and Electric Company (BG&E) requested changes to the Technical Specifications for Calvert Cliffs Unit No. 2. The proposed changes would incorporate improvements to the Technical Specifications similar to changes recently reviewed and approved for Unit No. 1 in Amendment No. 20 to Facility Operating License No. DPR-53 issued February 11, 1977. By supplementary letter dated March 24, 1977, BG&E requested prompt NRC staff review to preclude an unnecessary plant shutdown the fourth week of April 1977.

DISCUSSION AND EVALUATION

The NRC staff has recently approved BG&E proposed changes to upgrade the Technical Specifications for Calvert Cliffs Unit No. 1. During the preparation and review of the upgraded specifications for Unit No. 1 certain editorial corrections and upgrading became evident for Unit No. 2 specifications. The licensee by letter dated March 3, 1977, identified changes which were proposed as improvements to the Unit No. 2 Technical Specifications, Appendix A, and one change to the Appendix B specifications.

The NRC staff has reviewed the proposed changes and agrees that these changes are consistent with changes recently reviewed and approved for Calvert Cliffs Unit No. 1. Also since (1) the Unit No. 2 facility is essentially identical to Unit No. 1 and share a common control room, the change would add essential consistency in operation, (2) the changes would add desirable flexibility, (3) one change would delete a requirement to shutdown the plant for checking valve lineups which would otherwise expose personnel to unnecessary high radiation doses, and (4) other changes would make minor editorial corrections; the proposed changes are acceptable.

In addition, the NRC staff noted during the review of the licensee's proposals that other changes were required. We added the following requirements which existed for Unit No. 1 and had been inadvertently left out of the Unit No. 2 specifications:

1. Specification 3.1.2.8

   A maximum boron concentration of 2700 ppm in the refueling water tank and a boron concentration limit of 8%. This change is consistent with the Unit No. 1 values and the ECCS requirements for Unit No. 2.

2. Specification 3.4.1

   A special test exception (Specification 4.10.5) for use during local leak rate testing of the containment penetration for the shutdown cooling system and during maintenance on the common portions of the shutdown cooling line. The exception permits all forced circulation of reactor coolant to be suspended during the specified test or maintenance work. The change is consistent with the approved Unit No. 1 specifications.

3. Specification 3.4.12

   A requirement that the bypass valve for the excess flow check valve in the letdown line be closed for Modes 1, 2, 3 and 4. The requirement ensures that the effects of a pipe rupture downstream will not exceed the assumptions in the accident analysis. The change is consistent with the approved Unit No. 1 specifications.

4. Specification 3.7.10

   Requirements to maintain water tight doors closed except for normal entry and exit. This ensures protection of safety related equipment from the effects of water or steam from ruptured systems in adjoining rooms. The change is consistent with the approved Unit No. 1 specifications.

Other minor editorial changes were made. All changes were discussed with BG&E representatives who agree with the need for the changes. The changes include only such safety considerations which the NRC staff has previously reviewed and found to meet our current requirements.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does 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 amendment involves 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 this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 22, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-318

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 6 to Facility Operating License No. DPR-69, issued to Baltimore Gas and Electric Company (the licensee), which revised the license and its appended Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant Unit No. 2 (the facility) located in Calvert County, Maryland. The amendment is effective as of its date of issuance.

This amendment modified the Technical Specifications of the facility in order to have its specifications consistent with the Technical Specifications recently approved for Calvert Cliffs Nuclear Power Plant Unit No. 1. This consistency is necessary because these units are essentially identical and both units share a common control room. This amendment also incorporated a number of miscellaneous editorial changes.

The application for the amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 3, 1977, as supplemented by letter dated March 24, 1977, (2) Amendment No. 6 to License No. DPR-69, and (3) the Commission's related Safety Evaluation. 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 20678. A single 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 Operating Reactors.

Dated at Bethesda, Maryland, this 22nd day of April 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W. O'Connor

Paul W. O'Connor, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors