May 20, 1985

Docket No. 50-317

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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 No. 104 to Facility Operating License No. DPR-53 for Calvert Cliffs Nuclear Power Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your applications dated December 31, 1984, February 22, 1985 (partial response), and February 26, 1985.

This amendment provides Technical Specifications changes for startup testing and operation of Calvert Cliffs Unit 1 for fuel cycle 8.

A copy of the related Safety Evaluation also is enclosed. The notice of issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

/S/

David H. Jaffe, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 104 to DPR-53
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrences

ORB#3:DL	*ORB#3:DL	*ORB#3:DL	*OELD	*AD:OR:DL
PMKreutzer	DJaffe	JRMiller	CWoodard	GCLainas
/ /85	5/14/85	5/14/85	5/15/85	5/17/85

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P PDR

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PMKreutzer	DJaffe	JRMiller		GCLainas
5/14/85	5/14/85	5/14/85	5/15/85	/17/85

Mr. A. E. Lundvall, Jr.
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

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Energy Administration, Power Plant
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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. 104
License No. DPR-53

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

   A. The applications for amendment by Baltimore Gas & Electric Company (the licensee) dated December 31, 1984, February 22, 1985, and February 26, 1985 comply 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 applications, 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.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 104 , 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

E. G. Tourigny, Acting Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 20, 1985

ATTACHMENT TO LICENSE AMENDMENT NO.104

FACILTIY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

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 areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 1-1	3/4 1-1
3/4 1-5	3/4 1-5
3/4 1-9	3/4 1-9
3/4 1-11	3/4 1-11
3/4 1-13	3/4 1-13
3/4 1-16	3/4 1-16
---	3/4 1-16a
3/4 2-2	3/4 2-2
3/4 2-5	3/4 2-5
3/4 2-11	3/4 2-11
3/4 5-5a	3/4 5-5a
3/4 7-1	3/4 7-1
3/4 7-4	3/4 7-4
3/4 10-1	3/4 10-1
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 5-2	B 3/4 5-2
B 3/4 7-1	B 3/4 7-1

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.7×10^{-4} $\Delta k/k/^{o}F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER.

b. Less positive than 0.2×10^{-4} $\Delta k/k/^{o}F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.7×10^{-4} $\Delta k/k/^{o}F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER above 90% of RATED THERMAL POWER, within 7 EFPD after initially reaching an equilibrium condition at or above 90% of RATED THERMAL POWER.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} > 200^{o}F$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 3.5%* $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 3.5%* $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 2300 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.5%* $\Delta k/k$:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

---

* Adherence to Technical Specification 3.1.3.6 as specified in Surveillance Requirements 4.1.1.1.1 assures that there is sufficient available shutdown margin to match the shutdown margin requirements of the safety analyses.

**See Special Test Exception 3.10.1.

With $K_{eff} \geq 1.0$.

##With $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Two flow paths from the boric acid storage tanks required to be OPERABLE pursuant to Specifications 3.1.2.8 and 3.1.2.9 via either a boric acid pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the concentrated boric acid tanks is above the temperature limit line shown on Figure 3.1-1.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying on a SIAS test signal that 1) each automatic valve in the flow path actuates to its correct position, and 2) each boric acid pump starts.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE:

a. At least once per 18 months by verifying that each charging pump starts automatically upon receipt of a Safety Injection Activation Test Signal.

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

BORIC ACID PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

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 3% Δk/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 Specifications 4.0.5 and 4.1.2.2.

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 9,844 gallons,

2. A minimum boron concentration of 2300 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.

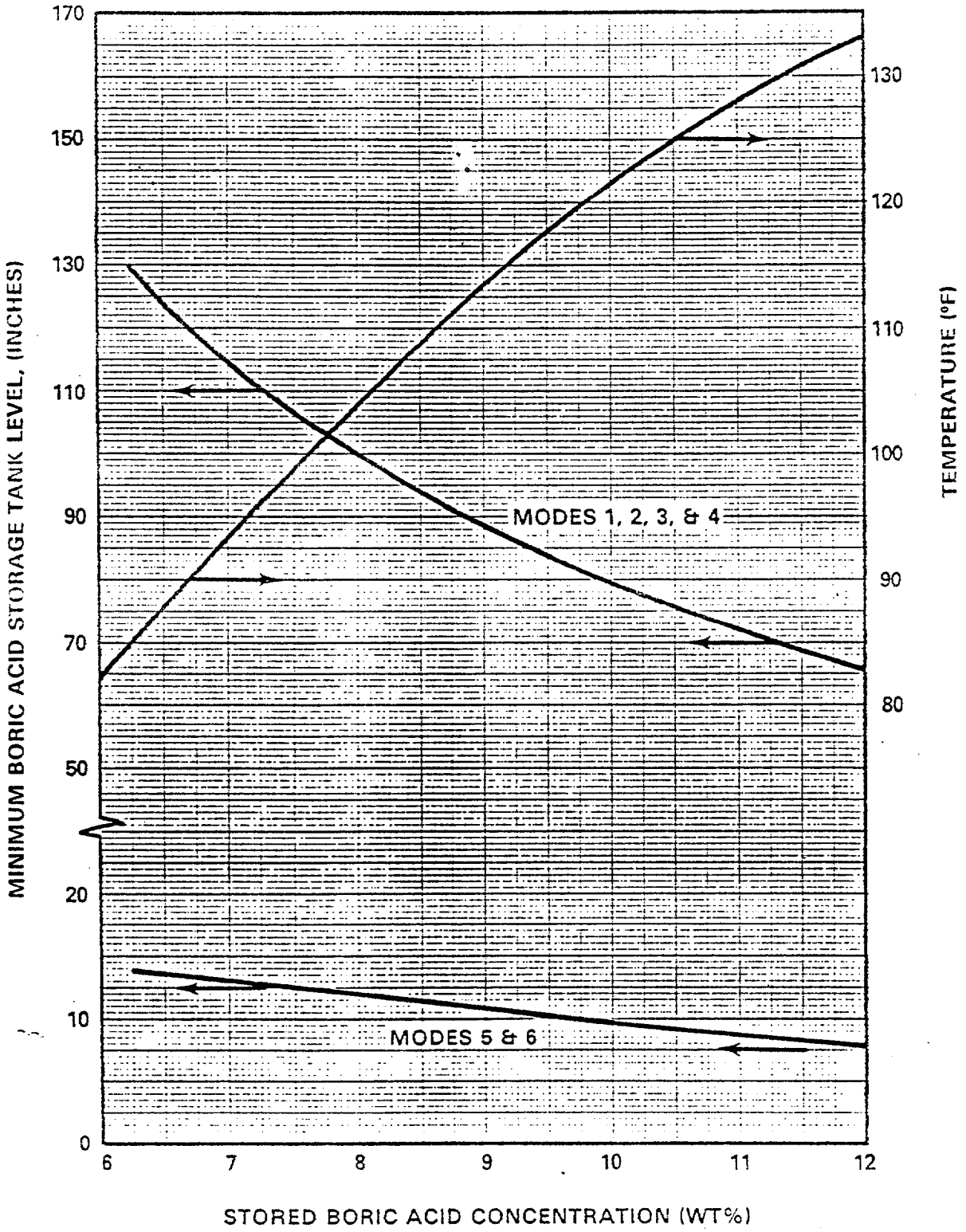

FIGURE 3.1-1

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 one of the following two combinations of 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%, or

b. Boric Acid Storage Tank 12 OPERABLE per Specification 3.1.2.8.a and the refueling water tank with

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

2. A boron concentration of between 2300 and 2700 ppm,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 100°F in MODE 1.

APPLICABILITY: MODE 1 > 80% of RATED THERMAL POWER.

ACTION:

a. With neither combination of borated water sources OPERABLE but at least two of the individual borated water sources OPERABLE, restore at least one of the combinations defined in Specification 3.1.2.8 to OPERABLE status within 72 hours or reduce power to less than 80% of RATED THERMAL POWER within the next 6 hours.

b. With only one borated water source OPERABLE, within 1 hour either restore at least two of the individual borated water sources to OPERABLE status or reduce power below 80% of RATED THERMAL POWER and comply with Specification 3.1.2.9.

SURVEILLANCE REQUIREMENTS

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

a. At least once 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 < 40°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.9 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 $\leq$ 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 2300 and 2700 ppm,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 100°F in MODE 1.

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 3% $\Delta k/k$ 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

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

a. At least once 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 < 40°F.

*At $\leq$ 80% of RATED THERMAL POWER.

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 12 hours that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

b. 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.

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the AXIAL SHAPE INDEX is maintained within the limits of Figure 3.2-2, where 100 percent of the allowable power represents the maximum THERMAL POWER allowed by the following expression:

$$M \times N$$

where:

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

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F^T_{xy} curve of Figure 3.2-3b.

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. A measurement-calculational uncertainty factor of 1.062,

2. An engineering uncertainty factor of 1.03,

3. A linear heat rate uncertainty factor of 1.002 due to axial fuel densification and thermal expansion, and

4. A THERMAL POWER measurement uncertainty factor of 1.02.

DELETED

TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2.1 The calculated value of F^T_{xy}, defined as $F^T_{xy} = F_{xy}(1+T_q)$, shall be limited to $\leq$ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.70, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits of Figure 3.2-3a and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+T_q)$ and F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030.

*See Special Test Exception 3.10.2.

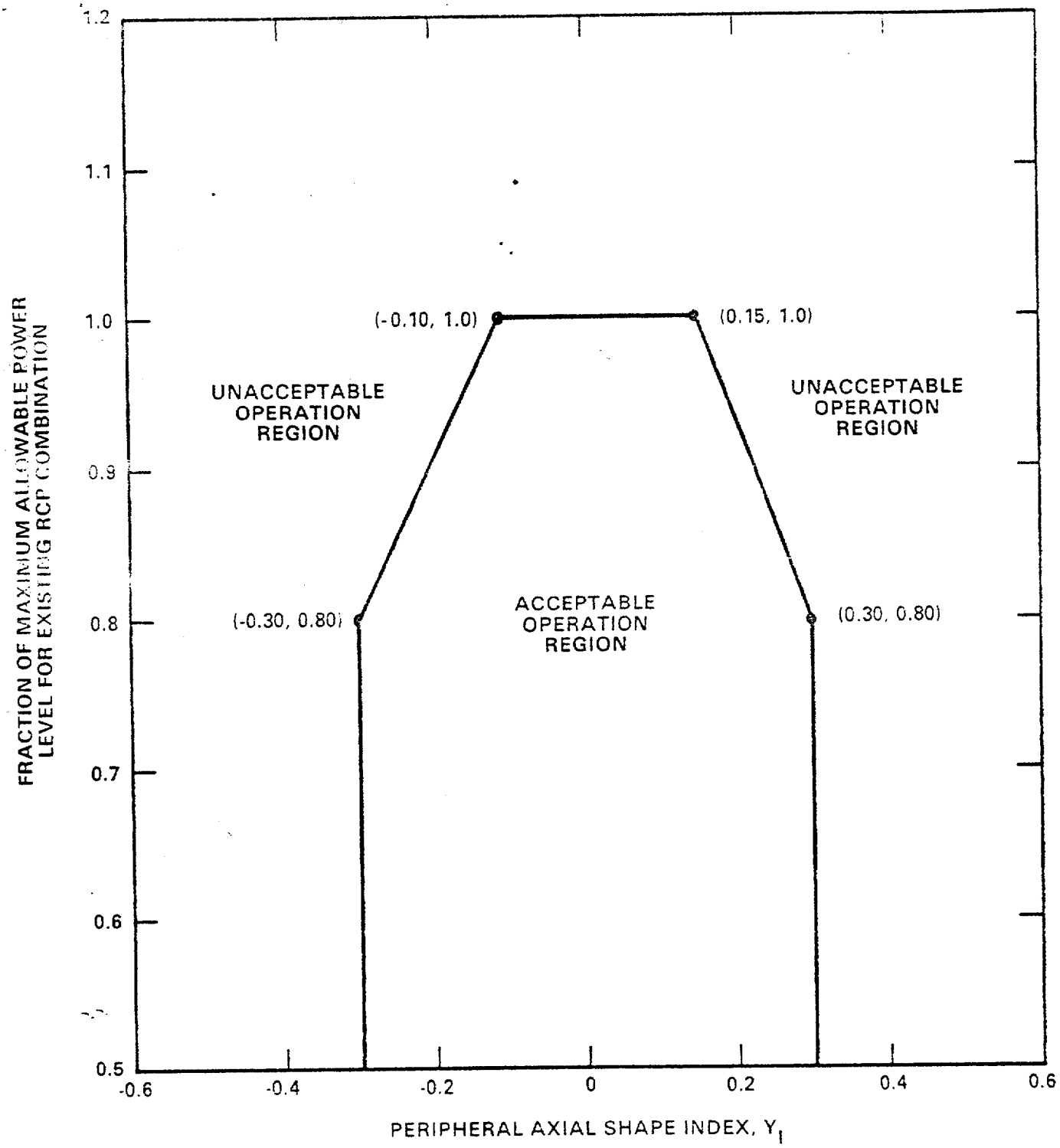

Figure 3.2-4
DNB Axial Flux Offset Control Limits

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.030.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.030 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 12 hours, and

b. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER IS > 75% of RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 100 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 4.0 ± 0.1 grams of TSP from a TSP storage basket is submerged, without agitation, in 3.5 ± 0.1 liters of 77 ± 10°F borated water from the RWT, the pH of the mixed solution is raised to ≥ 6 within 4 hours.

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

a. High-Pressure Safety Injection pump.

b. Low-Pressure Safety Injection pump.

g. By verifying the correct position of each electrical position stop for the following Emergency Core Cooling System throttle valves:

1. During each performance of valve cycling required by Specification 4.0.5 by observation of valve position on the control boards.

SURVEILLANCE REQUIREMENTS (Continued)

2. Within 4 hours following completion of maintenance on the valve or its operator by measurement of stem travel when the ECCS subsystems are required to be OPERABLE:

HPSI SYSTEM

Valve Number	Valve Number
MOV-616	MOV-617
MOV-626	MOV-627
MOV-636	MOV-637
MOV-646	MOV-647

h. By performing a flow balance test during shutdown following completion of HPSI system modifications that alter system flow characteristics and verifying the following flow rates for a single HPSI pump system*:

1. The sum of the three lowest flow legs shall be greater than 470** gpm.

i. By verifying that the HPSI pumps develop a total head of 2900 ft. on recirculation flow to the refueling water tank when tested pursuant to Specification 4.0.5.

* A HPSI pump system is a HPSI pump and one of two safety injection headers.
**These limits contain allowances for instrument error, drift or fluctuation.

3.4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE*.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided that, within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the following 30 hours.

b. With one reactor coolant loop and associated steam generator in operation and with one or more main steam line code safety valves associated with the operating steam generator inoperable, operation in MODES 1, 2 and 3 may proceed provided:

   1. That at least 2 main steam line code safety valves on the non-operating steam generator are OPERABLE, and

   2. That within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5 are applicable for the main steam line code safety valves of Table 4.7-1.

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*Entry into MODE 3 is permitted to determine operability of main steam line code safety valves. During this time, at least 2 main steam line code safety valves per steam generator shall be operable.

TABLE 3.7-1

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

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	93
2	79
3	66

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	LIFT SETTINGS* ALLOWABLE	ORIFICE SIZE
a. RV-3992/4000	935-995 psig	R
b. RV-3993/4001	935-995 psig	R
c. RV-3994/4002	935-1035 psig	R
d. RV-3995/4003	935-1035 psig	R
e. RV-3996/4004	935-1065 psig	R
f. RV-3997/4005	935-1065 psig	R
g. RV-3998/4006	935-1065 psig	R
h. RV-3999/4007	935-1065 psig	R

*Lift settings for a given steam line are also acceptable if any 2 valves lift between 935 and 995 psig, any 2 other valves lift between 935 and 1035 psig and the 4 remaining valves lift between 935 and 1065 psig.

SPECIAL TEST EXCEPTIONS

MODERATOR TEMPERATURE COEFFICIENT, CEA INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, the CEA insertion and the power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to below 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or 3.2.4 are suspended.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 40 gpm of 2300 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 40 gpm of 2300 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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 minimum available SHUTDOWN MARGIN for no load operating conditions at beginning of life is 3.5% $\Delta k/k$ and at end of life is 3.5% $\Delta k/k$. The SHUTDOWN MARGIN is based on the safety analyses performed for a steam line rupture event initiated at no load conditions. The most restrictive steam line rupture event occurs at EOC conditions. For the steam line rupture event at beginning of cycle conditions, a minimum SHUTDOWN MARGIN of less than 3.5% $\Delta k/k$ is required to control the reactivity transient, and end of cycle conditions require 3.5% $\Delta k/k$. 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 3% $\Delta k/k$ shutdown margin provides adequate protection. With the pressurizer level less than 90 inches, the sources of non-borated water are restricted to increase the time to criticality during a boron dilution event.

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 system also provides coolant flow following an SIAS (e.g., during a Small Break LOCA) to supplement flow from the Safety Injection System. The Small Break LOCA analyses assume flow from a single charging pump, accounting for measurement uncertainties and flow mal-distribution effects in calculating a conservative value of charging flow actually delivered to the RCS. 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 3.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6500 gallons of 7.25% boric acid solution from the boric acid tanks or 55,627 gallons of 2300 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. The maximum boron concentration of the refueling water tank shall be limited to 2700 ppm and the maximum boron concentration of the boric acid storage tanks shall be limited to 8% to preclude the possibility of boron precipitation in the core during long term ECCS cooling.

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.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 3) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) a measurement-calculational uncertainty factor of 1.062, 2) an engineering uncertainty factor of 1.03, 3) an allowance of 1.002 for axial fuel densification and thermal expansion, and 4) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F^T_{xy} AND F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy}, F^T_r or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would not be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F^T_{xy} = F_{xy}(1 + T_q)$ and $F^T_r = F_r(1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F^T_{xy}, F^T_r and T_q are within their limits provide assurance that the actual values of F^T_{xy}, F^T_r and T_q do not exceed the assumed values. Verifying F^T_{xy} and F^T_r after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.23 throughout each analyzed transient.

In addition to the DNB criteria, there are two other criteria which set the specification in Figure 3.2-4. The second criteria is to ensure that the limiting core power distribution at full power is less severe than the power distribution factored into the small-break LOCA analysis. This results in a limitation on the allowed negative AXIAL SHAPE INDEX value at full power. The third criteria is to maintain limitations on peak linear heat rate at low power levels resulting from Anticipated Operational Occurrences (AOOs). Figure 3.2-4 is used to assure the LHR criteria for this condition because the linear heat rate LCO, for both ex-core and in-core monitoring, is set to maintain only the LOCA kw/ft requirements which are limiting at high power levels. At reduced power levels, the kw/ft requirements of certain AOOs (e.g., CEA withdrawal), tend to become more limiting than that for LOCA.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The safety injection tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

Portions of the low pressure safety injection (LPSI) system flowpath are common to both subsystems. This includes the low pressure safety injection flow control valve, CV-306, the flow orifice downstream of CV-306, and the four low pressure safety injection loop isolation valves. Although the portions of the flowpath are common, the system design is adequate to ensure reliable ECCS operation due to the short period of LPSI system operation following a design basis Loss of Coolant Incident prior to recirculation. The LPSI system design is consistent with the assumptions in the safety analysis.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to $\geq$ 7.0. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. Minimum HPSI flow requirements are based upon small break LOCA calculations which credit charging pump flow following an SIAS. Surveillance testing includes allowances for instrumentation and system leakage uncertainties. The 470 gpm requirement for minimum HPSI flow from the three lowest flow legs includes instrument uncertainties but not system check valve leakage. The OPERABILITY of the charging pumps and the associated flow paths is assured by the Boration System Specification 3/4.1.2. Specification of safety injection pump total developed head ensures pump performance is consistent with safety analysis assumptions.

3/4.5.4 REFUELING WATER TANK (RWT)

The OPERABILITY of the RWT as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1000 psig during the most severe anticipated system operational transient. The total relieving capacity for all valves on all of the steam lines is 12.18×10^6 lbs/hr at 100% RATED THERMAL POWER. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser). The main steam line code safety valves are tested and maintained in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition. The as-left lift settings will be no less than 985 psig to ensure that the lift setpoints will remain within specification during the cycle.

In MODE 3, two main steam safety valves are required OPERABLE per steam generator. These valves will provide adequate relieving capacity for removal of both decay heat and reactor coolant pump heat from the reactor coolant system via either of the two steam generators. This requirement is provided to facilitate the post-overhaul setting and OPERABILITY testing of the safety valves which can only be conducted when the RCS is at or above 500°F. It allows entry into MODE 3 with a minimum number of main steam safety valves OPERABLE so that the set pressure for the remaining valves can be adjusted in the plant. This is the most accurate means for adjusting safety valve set pressures since the valves will be in thermal equilibrium with the operating environment.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

For single loop operation (two reactor coolant pumps operating in the same loop)

$$SP = \frac{(X) - (Y)(U)}{X} \times 46.8$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

U = maximum number of inoperable safety valves per operating steam line

106.5 = Power Level - High Trip Setpoint for two loop operation

46.8 = Power Level - High Trip Setpoint for single loop operation with two reactor coolant pumps operating in the same loop

X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. A capacity of 400 gpm is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300°F when the shutdown cooling system may be placed into operation.

Flow control valves, installed in each leg supplying the steam generators, are set to maintain a nominal flow setpoint of 200 gpm plus or minus 10 gpm for operator setting band. The nominal flow setpoint of 200 gpm incorporates a total instrument loop error band of plus 25 gpm and minus 26 gpm for the motor-driven pump train. The corresponding values for the steam-driven pump train are plus 37 gpm and minus 40 gpm. The operator setting band, when combined with the instrument loop error, results in a total flow band of 164 gpm (minimum) and 235 gpm (maximum) for the motor-driven pump train. The corresponding values for the steam-driven pump train are 150 gpm (minimum) and 247 gpm (maximum). Safety analyses show that more flow during an overcooling transient and less flow during an undercooling transient could be tolerated; i.e., flow fluctuations outside this flow band but within the assumptions used in the analyses listed below, are allowable.

In the spectrum of events analyzed in which automatic initiation of auxiliary feedwater occurs, the following flow conditions are allowed with an operator action time of 10 minutes.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.104

TO FACILITY OPERATING LICENSE NO. DPR-53

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-317

Introduction

By letters dated December 31, 1984 (Reference 1), February 22, 1985 (Reference 2) and February 26, 1985 (Reference 3), Baltimore Gas and Electric Company made application to revise the Technical Specifications (TS) of Calvert Cliffs Nuclear Power Plant, Unit 1. The changes are required, in part, to permit operation of Unit 1 for Cycle 8. Other changes are the result of a reanalysis of fuel densification and clad collapse leading to a deletion of the axially dependent power augmentation factor and a reduction in peaking factor uncertainties to conform to approved values.

Several requested changes to the Unit 1 TS in Reference 2 were not associated with the reload and will be addressed in future correspondence.

Discussion and Evaluation

1. Fuel Densification and Clad Collapse

Attachment 5 to Reference 1 is a report, entitled "Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," on work performed by Combustion Engineering (CE) for Electric Power Research Institute (EPRI). The report presents the results from a review of interpellet-gap formation, ovality, creepdown and clad collapse data in modern PWR fuel rods. Based on these results, CE has reformulated its creep-collapse predictor, CEPAN, to treat finite gaps and reanalyzed the flux augmentation (spiking) factor to take account of gap formation statistics data from modern fuel.

The data obtained by CE from measurements performed on its own fuel and the published results of such measurements for fuel from other PWR vendors was examined to obtain information on the number, axial distribution and size of densification induced gaps formed in PWR fuel rods. Data were obtained on old fuel (unpressurized rods containing densifying fuel), intermediate fuel (pressurized but with densifying fuel) and modern fuel (pressurized rods containing non-densifying fuel). Densifying fuel is that which increases in density by about 3 percent when resintered in-reactor. Non-densifying fuel shows a density increase of less than 0.5 percent upon resintering.

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The report concludes that, in modern CE fuels, the maximum gap size is less than 0.025 inches and that gaps are distributed uniformly along the fuel length (gap density (number per unit length) and gap size are not a function of core height). Accordingly, the time to clad collapse and the flux augmentation factors have been recalculated. It is concluded that modern CE fuel rods have a time to clad collapse far in excess of any practical residence time. It is further concluded that the maximum augmentation factor is 1.001 for gaps less than 0.025 inches, which is insignificant with respect to other power distribution uncertainties.

The staff concurs with the conclusions of the CE report as it applies to Unit 1 Cycle 8 and for Unit 2 Cycle 7. This concurrence is supported by similar results of analyses by another fuel vendor. We, therefore, conclude that, for Cycle 8 of Unit 1 and Cycle 7 of Unit 2, no further analysis of clad collapse need be performed and that height dependent augmentation factors may be deleted from the Technical Specifications.

2. Unit 1 Cycle 8 Reload

An enclosure to Reference 2 contains the reload licensing submittal for Calvert Cliffs Unit 1 Cycle 8. The Cycle 8 length is increased over current cycles, which results in a more positive Beginning-of-Cycle moderator temperature coefficient and a more negative End-of-Cycle coefficient. A revision to the control rod group assignments is made to maintain the required shutdown margins. The range of allowable lift settings on the steam line safety valves has been increased. Maximum permitted values range from 995 to 1065 psig compared to a present range of 985 to 1035 psig. In addition, the flow requirement for the High Pressure Safety Injection (HPSI) system has been reduced. These changes are evaluated below.

2.1 Fuel Mechanical Design

The mechanical design of the new fuel to be inserted for Cycle 8 is identical to that inserted for Cycle 7 except that the shoulder gap has been increased in length in order to improve fuel performance. This increase was achieved by shortening the lower end fitting and increasing the guide tube length. The grid structure was altered to make it compatible with the earlier fuel. Fuel performance analyses have been performed which confirm that those changes have negligible effect on the new fuel performance. We conclude that the new fuel for Cycle 8 is acceptable.

A sigle SCOUT demonstration assembly and four PROTOTYPE demonstration assemblies currently in the core will be carried over into Cycle 8 without alteration. This is acceptable.

2.1.1 Clad Collapse

A generic analysis of the phenomenon of clad collapse in modern CE fuels has been submitted by the licensee[1]. Section 1 of this evaluation reports on the results of the staff review. It is concluded that a cycle-specific analysis is not required for Cycle 8.

A clad collapse analysis has been performed for the SCOUT demonstration assembly due to the presence of artificial gaps. This analysis shows that the expected time to collapse exceeds the cumulative exposure of the affected rods at the end of Cycle 8. We conclude that the Cycle 8 reload is acceptable with respect to clad collapse.

2.2 Nuclear Design

The nuclear parameters of the Cycle 8 core have been determined with the same methods and techniques as those used for the reference cycle (Cycle 7). The differences between the reactivity coefficients of the two cycles are small and include a larger critical boron concentration at Beginning of Cycle (BOC), a more positive BOC moderator temperature coefficient (MTC) and a more negative End of Cycle MTC. These differences are accounted for in the safety analyses. The total control rod worth is essentially the same for the two cycles but the distribution among the regulating groups (3, 4, and 5) is different due to the rearrangement of the group assignments. The worth of group 5 is significantly reduced while that of group 4 is increased. This rearrangement results in an increase in the potential zero power ejected rod worth for Cycle 8. The rod ejection accident at zero power has been reanalyzed and is addressed in Section 2.4.5 herein.

Power distributions have been calculated for the maximum Cycle 7 burnup case which yield largest Cycle 8 peaking factors. The expected power distributions are bounded by the values used in the safety analyses.

The interpellet gap augmentation factors can be eliminated from the analysis as described above. A 15 percent negative bias is being applied to the Fuel Temperature (Doppler) coefficient to make it consistent with the power coefficient bias used in the ROCS/DIT code. This bias is applied conservatively to power increase transients.

We find the nuclear design analysis to be acceptable. This conclusion is based on the fact that previously used and approved design methods are employed and the results are bounded by previous values or revised values are used in the safety analysis.

2.3 Thermal Hydraulic Design

The thermal hydraulic parameters of Cycle 8 are essentially the same as those of the reference Cycle 7. Small differences in the total heat transfer area due to a lower number of shims in Cycle 8 are accounted for in the analyses. The DNBR analyses were performed with previously approved methods to verify the applicability of the generic limit to Cycle 8. The effects of rod bowing were analyzed by approved methods to yield a penalty of 0.3% DNBR. The Technical Specification limit on radial power peaking includes a penalty of 0.4% to account for rod bowing.

Because previously used and approved methods were employed in the analysis, we conclude that the thermal hydraulic design is acceptable.

2.4 Transient Analysis

A re-evaluation of all Design Basis Events (DBEs) was performed to determine the effect of the following changes:

1. Inclusion of the 15 percent negative bias in the fuel temperature coefficient multiplier,

2. Increases in the allowable lift settings for the steam generator safety valves,

3. Changes in the Moderator Temperature Coefficient range,

4. Reduction in required shutdown margin from 4.3 percent to 3.5 percent reactivity change, and

5. Reduction in the HPSI pump flow rate.

Many of the events were not affected by the changes or the results were bounded by those for the reference cycle. The reanalyzed events are discussed below.

2.4.1 Boron Dilution Event

This event was reanalyzed on the basis of the reduction in the shutdown margin requirement for Modes 2, 3, and 4. The reduction in shutdown margin reduces the calculated time to criticality for this event in all three modes. However, the calculated times are still large compared to the acceptance criterion (50-60 minutes vs. 15 minutes). We conclude that the analysis of this event is acceptable.

2.4.2 Loss of Load Event

Reanalysis of this event was required by the increase in the positive moderator temperature coefficient and the higher values of the opening setpoints of the main steam safety valves. Analysis of this event was performed with methods identical to those used for the reference cycle. Initial conditions and core reactivity parameters were chosen to maximize the consequences. The results of the analysis showed that neither the DNBR safety limit nor the upset pressure limit (110 percent of design value) is violated during this event. Because approved methods are used, conservative input conditions are assumed and relevant safety limits are not violated, we conclude that the anlaysis of this event is acceptable.

2.4.3 Excess Load Event

Two such events - one at hot full power (HFP) and one at hot zero power (HZP) are analyzed. The reanalyses were necessary to include the effects of a more negative moderator temperature coefficient, a lower CEA worth available at trip (HZP only), and changes in the HPSI flow characteristics (HFP only). Analyses were performed with the same analysis methods previously used and

with conservative input assumptions with respect to moderator temperature coefficient, fuel temperature coefficient, boron worth, and operation of the pressurizer pressure control system. The reduced flow of the High Pressure Safety Injection (HPSI) system was accounted for in the analysis.

The results for the HFP event (complete opening of the steam dump and bypass valves) show that the DNBR safety limit is not violated and the core does not return to critical after scram. The results for the HZP event (complete opening of turbine control valves) show that core thermal power does not exceed 25 percent of full power and that the core remains subcritical after the scram. The DNBR safety limit is not violated during this event.

Because the analyses were performed with previously used methods, the differences in core parameters from those of the reference cycle are conservatively accounted for, and the results are within our acceptance criteria, we conclude that the analysis of the Excess Load Event is acceptable.

2.4.4 Steam Generator Malfunction Events

The malfunctions of a single steam generator - loss of load, excess load, loss of feedwater, and excess feedwater - have been examined and the determination made that the limiting event among them is the loss of load. Reanalysis of this event was required due to changes in the moderator temperature coefficient and the Main Steam Safety Valve opening setpoints.

The event is initiated by the inadvertent closure of a single main steam isolation valve. The analysis assumes conservatively large values of the negative moderator temperature coefficient, the Asymmetric Steam Generator Trip, and the steam generator safety valve opening pressure. Methods consistent with those used in the reference cycle were used to perform the analysis. The results of the analysis show that neither the DNBR nor the fuel centerline temperature LHGR limits are violated during this event. This fact, coupled with the use of previously used methods and conservative input parameters, permits the conclusion that the analysis of the steam generator malfunction event is acceptable.

2.4.5 CEA Ejection Event

The zero power control rod (CEA) ejection event has been reanalyzed to account for an increased potential ejected rod worth and post accident radial peaking factor and to allow for an increase in the positive moderator temperature coefficient. The NRC approved analysis method has been employed for this event; kinetics parameters for the analysis were chosen so that the most conservative value occurring during the cycle was chosen for each parameter. The above is conservative and acceptable.

The results show that the peak average enthalpy is less than 200 calories per gram which meets our acceptance criteria of 280 calories per gram and is acceptable.

2.4.6 Steam Line Rupture Analyses

Steam line break (SLB) events have been examined to determine those that are affected by the changes listed above. It was determined that the Cycle 7 results are bounding for all pre-trip breaks. However, post-trip breaks are affected and must be evaluated. Previous analyses have shown that post-trip breaks inside containment have more adverse consequences than those outside. Accordingly, both HFP and HZP initiated breaks were analyzed with and without loss of AC power for both events.

Analyses show that the results of the SLBs with loss of AC power on turbine trip were bounded by the results from Cycle 7. Of the breaks without loss of AC power the HFP case was limiting. Neither of the breaks approaches the DNBR safety limit. However, they do challenge the linear heat generation limit. Analysis of the HFP case was performed with conservative input parameters including end of cycle values of the moderator and fuel temperature coefficients, low boron worths in the HPSI injected water, and maximum peaking factors in core. Credit was taken for three-dimensional effects in the return-to-power calculation which was previously done in the Cycle 7 analysis and is acceptable. Additional conservatisms included immediate initiation of auxiliary feedwater flow and the use of the runout value for that flow.

The staff reviewed the radiological consequences of the Main Steamline Break (MSLB) Inside Containment, since the licensee submittal[5] indicated the possibility of some fuel failure. Although the licensee estimated less than 1 percent fuel failure, his analysis assumed 2 percent of the fuel failed. The staff analysis conservatively assumed 2 percent fuel failure. The licensee states that, during the course of a main steam line break (MSLB) inside containment, the affected steam generator blows dry at 67.6 seconds, terminating the RCS cooldown. A peak return-to-power is produced at 66.3 seconds, at which time less than 1 percent of the fuel exceeds the centerline melt temperature limit.

The licensee states that, for the Cycle 8 analysis, pre-trip MSLB inside containment radiological consequences are bounded by the outside containment MSLB consequences presented in the Cycle 7 reload license amendment submittal dated September 1, 1983. The limiting pre-trip Cycle 7 MSLB event results in estimates of less than 1 percent of the fuel pins experiencing fuel failure. The licensee conservatively assumed 2 percent fuel failure and determined the 0-2 hr Exclusion Area Boundary (EAB) thyroid dose to be 81 Rem, within the 10 CFR guideline value of 300 Rem.

The staff has performed an independent analysis of the MSLB <u>inside</u> containment event, also assuming 2754 MW_t power level, 2 percent fuel failure, 0.2 percent per day containment leak rate, and a meteorological diffusion and transport relative concentration (X/Q) of 1.9×10^{-4} sec/m^3. Since the radionuclide inventory due to fuel failed in the SLB would be discharged within containment and then leaked slowly to the environment, and it constitutes more than 95% of the released inventory, the 0-2 hr Exclusion Area Boundary thyroid dose is 5 Rem. This value is a very small fraction of, and therefore within, the 10 CFR 100 guideline value.

We conclude that acceptable methods have been employed, conservative input parameters used and acceptable consequences obtained. We find this analysis to be acceptable.

2.4.7 Large Break Loss-of-Coolant Accident (LOCA)

The large break LOCA event was reanalyzed to take into account the lowering of the minimum containment pressures from 14.7 psia to 13.7 psia, the reduction in the minimum HPSI flow, and the removal of the augmentation factors. NRC approved LOCA analysis methods were employed and LOCA acceptance criteria for peak clad temperature and clad oxidation were met with margin for a peak allowable LHGR of 15.5 kw/ft. We conclude that the large break LOCA analysis is acceptable as is the continued use of the 15.5 kw/ft LHGR limit.

2.4.8 Small Break LOCA Event

The small break LOCA was reanalyzed in order to account for differences between Cycle 8 and the reference analysis. These differences include a reduction in the HPSI flow rate, a reduction in the allowable peak LHGR (from 16.0 kw/ft to 15.5 kw/ft), an increase in low pressurizer pressure setpoints for reactor trip and safety injection actuation, and an increase in the pressure setpoints of the main steam safety valves. The NRC approved small break evaluation model was used in the analysis. A revision was made to the evaluation model to take credit for the flow from one charging pump in addition to the HPSI injection. To further offset the effect of the reduced ECCS flow the allowable negative axial shape index (ASI) is being reduced from 0.15 to 0.10.

The results of this analysis show that the criteria on fuel clad temperature and oxidation are met with margin. We conclude that the small break LOCA analysis is acceptable.

2.4.9 Fuel Handling Accident

The NRC staff analyzed the Fuel Handling Accident inside containment (FHAIC), since the increased fractional volatile radionuclide gap inventories may be increased due to the extended burnup (>38,000 MWd/t, batch average at discharge) to be employed in Cycle 8 beyond the 10% value suggested in Reg. Guide 1.25 (except for 30% for Kr-85). For the dominant radionuclide, I-131, a 22% fractional gap activity was used, corresponding to a peak linear heat generation rate of 15.5 KW/ft and an end-of-cycle (EOC) burnup of 18,100 MWd/t. The 0-2 hr radiological consequences of the FHAIC at the Exclusion Area Boundary, assuming a 0-2 hr meteorological diffusion and transport factor, X/Q, of 1.9×10^{-4} sec/m^3 are 55 Rem to the thyroid and <1 Rem whole body. These estimates are well within the SRP guideline value and are, therefore, acceptable.

The staff recognizes that its analysis of the FHAIC contains at least three very conservative assumptions. Use of a water scrubbing decontamination factor of only 100 for iodine is one of these assumptions. The others are neglect of any iodine plateout within the fuel assembly (clad and plenum internal surfaces), and the assumption that fuel is fissioned at the peak linear heat generation rate throughout Cycle 8.

3.0 Technical Specification Changes

TS associated with start-up testing and Cycle 8 operation were reviewed as described herein.

TS 3/4.1.1.1-Shutdown Margin:

The shutdown margin is being lowered from $>4.3\%\ \Delta k/k$ to $\geq 3.5\%\ \Delta k/k$ to accommodate the effects of extended burnup. The Boron Dilution, Excess Load and Steam Line Break events have been reanalyzed with the revised shutdown margin and show acceptable results. We conclude that this change is acceptable.

TS 3/4.1.1.4 Moderator Temperature Coefficient:

The range of the allowed moderator temperature coefficient (MTC) is being extended from .5 to $7 \times 10^{-4}\ \Delta k/k°F$ and from -2.5 to $-2.7 \times 10^{-4}\ \Delta k/k/°F$ to accommodate the longer fuel cycle and extended burnup. Safety analyses have been altered to include the extended range with acceptable results and we conclude that the change is acceptable.

TS 4.2.1.4 and Figure 4.2-1:

The flux peaking augmentation factors have been deleted from this TS and the axial fuel densification and thermal expansion factor has been reduced from 1.0 percent to 0.2 percent. The justification for these changes is discussed in Section 1 herein and we find these changes to be acceptable. The measurement-calculational uncertainty factor has been reduced from 7.0 to 6.2 percent. This is consistent with the latest approved evaluation of this quantity (Reference 4) and is acceptable.

TS Figure 3.2-4:

This figure is being modified to reduce the allowable value of the full power negative axial shape index from -0.15 to -0.10. This is consistent with the restriction assumed in the analysis of the small break LOCA event and is acceptable.

TS 3/4.7.1.1 and Table 4.7-1:

A footnote has been added which will permit entry into Mode 3 for the purpose of determining safety valve operability with a minimum of two operable safety valves per steam generator. Analyses have been performed to demonstrate that sufficient relief capacity exists in Mode 3 with only two operable main steam line safety valves per steam generator. We conclude that this change is acceptable.

Table 4.7-1 has been revised to define the revised values of the allowable lift setpoints for the steam generator safety valves. The revised values are consistent with or conservative with respect to the values assumed in the safety analysis and are acceptable. A footnote has been added to TS Table 4.7-1 to allow flexibility among safety valve settings while preserving the overall relief capability. This change is consistent with the safety analyses and is acceptable.

TS 3/4.1.2.2 Flow Paths - Operating:

This Specification has been altered as a result of the credit taken for charging pump flow to augment HPSI flow during the small break LOCA event. It requires that each boric acid pump starts upon receipt of a Safety Injection Actuation Signal (SIAS) as well as verifying proper valve line-up. The proposed changes would also clarify the operability requirements for the flow paths.

In order for the charging pumps, associated boric acid tanks, boric acid pumps, piping, valves and heat tracing equipment to be considered as an integral part of the emergency core cooling system, these must be safety-related, satisfy single-failure criterion and be properly tested.

The licensee has stated[5] that the charging pumps, associated boric acid tanks, boric acid pumps, piping, valves and heat tracing equipment are safety-grade and powered by class 1E power supplies and satisfy the single-failure criterion.

The operability requirements for the heat tracing and the charging pumps as well as the operability and responsiveness of the charging valves are specified in the TS. The charging pumps' surveillance testing is specified in the TS and is done quarterly as required by ASME Code Section XI[7]. Based on the information[7] submitted by BG&E, Surveillance Test Procedure (STP) 0-56 is now being modified to include response time testing for the charging pumps, boric acid pumps and their associated flow path valves.

The licensee stated that provision for recirculation for the charging pumps is not necessary since the inventory in the boric acid tank is sufficient until the reactor core is flooded following a small break LOCA. The most limiting demand on inventory is a maximum of three charging pumps operating with a combined flow of 132 gal/min and only one boric acid tank operable with a minimum inventory of 5900 gallons (per TS). The boric acid tank is emptied with this configuration in about 45 minutes. The licensee's analysis shows that, prior to that time and for the most limiting break size of 0.1 ft^2,[1] the peak clad temperature of 1877°F has already occurred and the core is covered.

We, therefore, conclude that addition of the charging pumps, boric acid pumps and auxiliaries to the ECCS and the proposed TS changes are acceptable.

TS 3/4.1.2.4 Charging Pumps - Operating:

An addition to the surveillance requirement for the charging pumps requires them to be test started on receipt of an SIAS signal. This is consistent with their use in the small break LOCA analysis and is acceptable.

TS 3/4.1.2.6 Boric Acid Pump - Operating:

The change to the Specification consists of adding a cross reference in the surveillance requirement to another boric acid pump surveillance requirement (Specification 4.1.2.2). This is a clarification and is acceptable.

TS 3/4.1.2.8 and 3/4.1.2.9, Borated Water Source - Operating:

The current TS 3/4.1.2.8 is replaced by two Specifications which are applicable over different power levels of Mode 1. The changes are necessitated by the use of credit for water from a charging pump during a small break LOCA. Such credit is necessary only when in Mode 1 above 80 percent power. Accordingly, TS 3/4.1.2.8 is applicable only in Mode 1 when power is greater than 80 percent of rated. Specification 3/4.1.2.9 is the same as the present 3/4.1.2.8 except that it is applicable in Mode 1 at less than or equal to 80 percent of rated power and in Modes 2 and 3.

In TS 3.1.2.8, "Reactivity Control System - Borated Water Sources," BG&E proposed a change in the combination of OPERABLE borated water sources from "any two out of three borated water sources" to "any one combination of (a) two boric acid tanks (BATs), or (b) a combination of BAT-12 and the refueling water tank (RWT)." This is to assure the availability of borated water source to the charging pumps in Mode 1 above 80 percent rated thermal power (RTP), thus changing the applicability of TS 3.1.2.8 from Modes 1, 2, 3 and 4 to Mode 1 above 80 percent power.

With the combination of two BATs, single failure of power train A will not affect the operability of boric acid pump-12. On the other hand, single failure of power train B will not affect the operability of gravity feed from both BATs. With the combination of BAT-12 and the RWT, single failure of either train A or train B will not disable the gravity feed and the boric acid pump-12 at the same time. Thus, a water source will be available to the charging pumps despite any active single failure. But with the combination of BAT-11 and the RWT, single failure of train A will disable both the gravity feed and boric acid pump-11 thus depriving the Reactor Coolant System (RCS) from the $\geq$ 13 gpm through the charging pumps as required by the revised LOCA analysis.

Applicability to "Modes 1, 2, 3 and 4" of TS Sections 3.1.2.8 and 3.1.2.9, "Borated Water Sources - Operating" was changed to "Mode 1 above 80%," and "Mode 1 below or equal to 80%, 2, 3 and 4, "respectively. In TS 3.1.2.8, the combination of borated water sources was also changed to assure availability of borated water sources to charging pumps following SIAS, while the reactor is in Mode 1 above 80 percent power. These changes are not necessary in TS 3.1.2.9 which requires that "at least two of the three borated water sources (RWT and two BATs) are operable."

In response to our inquiry on how the 80 percent power threshold was established, BG&E explained that the reduction in the maximum allowed power to 80 percent with less restrictive borated water source operability reduces the decay heat rate during Small Break LOCA by 20 percent. This reduces the boil off of the coolant inventory which can be equated to an inventory addition. This equivalent inventory addition was compared to that resulting from the charging flow credited for the 100 percent power analysis and found to make up for the charging pump flow.

Based on the licensee's explanation and the reanalysis for the Small Break LOCA, the proposed changes to the TS are acceptable.

TS 4.5.2h, "Surveillance Requirements - ECCS":

In TS Section 4.5.2h, "Surveillance Requirements - ECCS", the verification of sufficient high-pressure safety injection (HPSI) flow was changed from a flow requirement of 170±5 gpm in each injection leg to a minimum flow of 470 gpm sum of the three lowest flow legs. This proposed TS has a footnote which indicates that this flow includes allowances for instrument error, drift or fluctuation. The upper limit on the flow was also deleted from this section.

With the assumed reduction in the HPSI pump flow capacity and crediting a minimum of 13 gpm flow from one charging pump, reanalysis of the ECCS performance for the worst Small Break LOCA (a 0.1 ft^2 break) demonstrated a peak clad temperature of 1877°F and a peak local clad oxidation percentage of 4.91 percent and corewide clad oxidation percentage of less than 0.632 which are less than the acceptance criteria of 2200°F, 17 percent and 1 percent, respectively.

With the reduction in the HPSI flow, reanalysis for both the full and zero power excess load events demonstrated a DNBR value which is greater than the design limit of 1.23. BG&E stated[1] that 0.1 ft^2 remains as the limiting small break with the modified HPSI and charging pump flows. The references and evaluation models used in the analysis were approved by the Nuclear Regulatory Commission (NRC).

BG&E submitted pump test results[6] which show that, with the existing piping arrangement, the HPSI pumps flow will not exceed the runout flow conditions, therefore obviating the need for a TS requirement on the HPSI flow upper limit. In addition, a new TS 4.5.2i would be added to require verification of HPSI flow at a total head of 2900 ft. on recirculation flow.

We conclude that these proposed TS changes are acceptable.

TS 3/4.10.1, Shutdown Margin:

In order to facilitate performance of startup testing, the licensee proposes to alter Specification 4.10.1.2 for both Units 1 and 2 to permit scram testing within 7 days prior to the performance of the CEA worth tests instead of the present 24 hours. This extension would eliminate the need for performance of a special scram test and would save both time and wear and tear on equipment.

We find the proposed change to be acceptable. Since those core changes which might result in a stuck rod occur slowly, the probability of having a stuck rod during the rod worth measurement is not significantly increased by the larger interval between scram testing and measurement. The probability of an overcooling event is not changed and no new accident would be created by the change. We thus conclude that the margin of safety for the core is not significantly changed.

Bases:

The bases for the various Specifications have been revised to make them consistent with the TS. This is acceptable.

4.0 Startup Testing

The startup testing program is not altered from that in the previous cycle and is acceptable.

5.0 Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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5. Letter, R. E. Denton (BG&E) to J. R. Miller (NRC), dated April 22, 1985

6. Letter, R. F. Ash (BG&E) to J. R. Miller (NRC), dated April 23, 1985

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