March 15, 1994

Docket No. 50-317

Mr. Robert E. Denton
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

Dear Mr. Denton:

SUBJECT: ISSUANCE OF AMENDMENT FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M87690)

The Commission has issued the enclosed Amendment No. 185 to Facility Operating License No. DPR-53 for the Calvert Cliffs Nuclear Power Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated September 3, 1993, as supplemented on February 1, 1994.

The amendment revises the heatup and cooldown curves and the low-temperature overpressure protection (LTOP) controls. The changes to the LTOP controls support proposed modifications to allow a variable-setpoint (VLTOP) protection system. The VLTOP system will increase the allowable operating pressure band in the LTOP region and increase the flexibility in the use of the reactor coolant pumps.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:

Daniel G. McDonald, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 185 to DPR-53
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDI-1	PM:PDI-1	OGC - NLO	D:PDI-1		
CVogan	DMcDonald:smm	MZOBLER	RACapra		
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Subject to noted corrections

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P PDR

DATED: March 15, 1994

AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE NO. DPR-53-CALVERT CLIFFS UNIT 1

Docket File
NRC & Local PDRs
PDI-1 Reading
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J. Calvo, 14/A/4
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OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
J. Tsao, 7/D/4
L. Lois, 8/E/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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

March 15, 1994

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Baltimore Gas and Electric Company
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Office of Nuclear Reactor Regulation

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1. Amendment No. 185 to DPR-53
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cc w/enclosures:
See next page

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DF01

Mr. Robert E. Denton
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit No. 1

cc:

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated September 3, 1993, as supplemented February 1, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 185, 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Francis J. Williams for

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 15, 1994

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 185 FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Revise Appendix A as follows:

Remove Pages	Insert Pages
V	V
X1	X1
3/4 1-11	3/4 1-11
3/4 1-14	3/4 1-14
3/4 3-14	3/4 3-14
3/4 4-2	3/4 4-2
3/4 4-4	3/4 4-4
3/4 4-27	3/4 4-27
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-35	3/4 4-35
3/4 4-36 thru 4-41*	3/4 4-36 thru 4-42*
3/4 5-4	3/4 5-4
3/4 5-7	3/4 5-7
B3/4 4-1	B3/4 4-1
B3/4 4-6	B3/4 4-6
B3/4 4-7	B3/4 4-7
B3/4 4-8	B3/4 4-8
B3/4 4-9	B3/4 4-9
B3/4 4-10	B3/4 4-10
B3/4 4-11*	B3/4 4-11*
B3/4 5-2	B3/4 5-2
B3/4 5-3	B3/4 5-3

*Rollover pages - vertical line at page number only.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

Flow Paths - Shutdown

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be **OPERABLE**:

a. A flow path from the boric acid storage tank via either a boric acid pump or a gravity feed connection and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is **OPERABLE**, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump* to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b is **OPERABLE**.

APPLICABILITY: **MODES** 5 and 6.

ACTION: With none of the above flow paths **OPERABLE**, suspend all operations involving **CORE ALTERATIONS** or positive reactivity changes until at least one injection path is restored to **OPERABLE** status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one 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 is above the temperature limit line shown on Figure 3.1.2-1 when a flow path from the concentrated boric acid tanks is used.

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.

---

* At 365°F and less, the required **OPERABLE** HPSI pump shall be in pull-to-lock and will not start automatically. At 365°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

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.

* At 365°F and less, the required **OPERABLE** HPSI pump shall be in pull-to-lock and will not start automatically. At 365°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

TABLE 3.3-3 (Continued)

TABLE NOTATION

\# Containment isolation of non-essential penetrations is also initiated by SIAS (functional units 1.a and 1.c).

@ When the RCS temperature is:

(a) Greater than 385°F, the required **OPERABLE** HPSI pumps must be able to start automatically upon receipt of a SIAS signal,

(b) Between 385°F and 365°F, a transition region exists where the **OPERABLE** HPSI pump will be placed in pull-to-lock on a cooldown and restored to automatic status on a heatup.

(c) At 365°F and less, the required **OPERABLE** HPSI pump shall be in pull-to-lock and will not start automatically.

* The provisions of Specification 3.0.4 are not applicable.

** Must be **OPERABLE** only in **MODE** 6 when the valves are required **OPERABLE** and they are open.

(a) Trip function may be bypassed in this **MODE** when pressurizer pressure is < 1800 psia; bypass shall be automatically removed when pressurizer pressure is $\geq$ 1800 psia.

(c) Trip function may be bypassed in this **MODE** below 785 psia; bypass shall be automatically removed at or above 785 psia.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be **OPERABLE:**

1. Reactor Coolant Loop #11 and at least one associated reactor coolant pump.

2. Reactor Coolant Loop #12 and at least once associated reactor coolant pump.

b. At least one of the above reactor coolant loops shall be in operation*.

APPLICABILITY: **MODE 3****.

ACTION:

a. With less than the above required reactor coolant loops **OPERABLE,** restore the required loops to **OPERABLE** status within 72 hours or be in **HOT SHUTDOWN** within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and initiate corrective action to return the required loop to operation within one hour.

---

* All reactor coolant pumps may be de-energized for up to 1 hour (up to 2 hours for low flow test) provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** A reactor coolant pump shall not be started with the RCS temperature less than or equal to 365°F unless (1) the pressurizer water level is less than or equal to 170 inches, and (2) the secondary water temperature of each steam generator is less than or equal to 30°F above the RCS temperature, and (3) the pressurizer pressure is less than or equal to 300 psia.

3/4.4 **REACTOR COOLANT SYSTEM**

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

Shutdown

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be **OPERABLE**:

1. Reactor Coolant Loop #11 and its associated steam generator and at least one associated reactor coolant pump,
2. Reactor Coolant Loop #12 and its associated steam generator and at least one associated reactor coolant pump,
3. Shutdown Cooling Loop #11*,
4. Shutdown Cooling Loop #12*.

b. At least one of the above coolant loops shall be in operation**.

APPLICABILITY: **MODES** 4***# and 5***#.

ACTION:

a. With less than the above required reactor coolant loops **OPERABLE**, initiate corrective action to return the required coolant loops to **OPERABLE** status within one hour or be in **COLD SHUTDOWN** within 24 hours.

---

* The normal or emergency power source may be inoperable in **MODE** 5.

** All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

*** A reactor coolant pump shall not be started with the RCS temperature less than or equal to 365°F unless (1) the pressurizer water level is less than or equal to 170 inches, and (2) the secondary water temperature of each steam generator is less than or equal to 30°F above the RCS temperature, and (3) the pressurizer pressure is less than or equal to 300 psia.

\# See Special Test Exception 3.10.5.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

Reactor Coolant System

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4.9-1 and 3.4.9-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of:

Maximum Allowable Heatup Rate	RCS Temperature
30°F in any one hour period	70°F to 164°F
40°F in any one hour period	> 164°F to 256°F
60°F in any one hour period	> 256°F

b. A maximum cooldown of:

Maximum Allowable Cooldown Rate	RCS Temperature
100°F in any one hour period	> 270°F
20°F in any one hour period	270°F to 184°F
10°F in any one hour period	< 184°F

c. A maximum temperature change of 5°F in any one hour period, during hydrostatic testing operations above system design pressure.

APPLICABILITY: At all times.

ACTION: With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least **HOT STANDBY** within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 300 psia, respectively, within the following 30 hours.

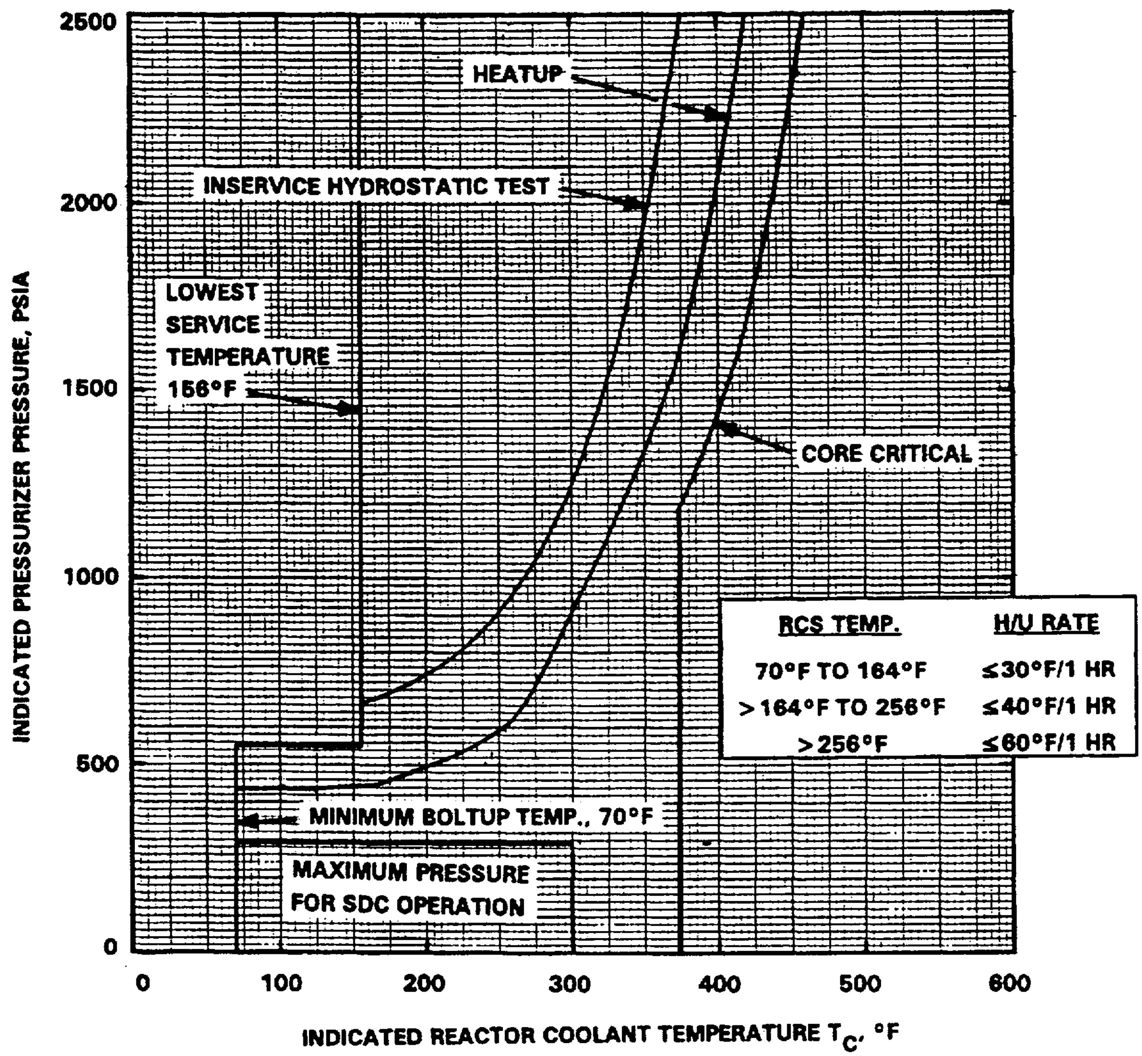

FIGURE 3.4.9-1

CALVERT CLIFFS UNIT 1 HEATUP CURVE, FOR FLUENCE ≤ 2.61×10^{19} n/cm^2
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS

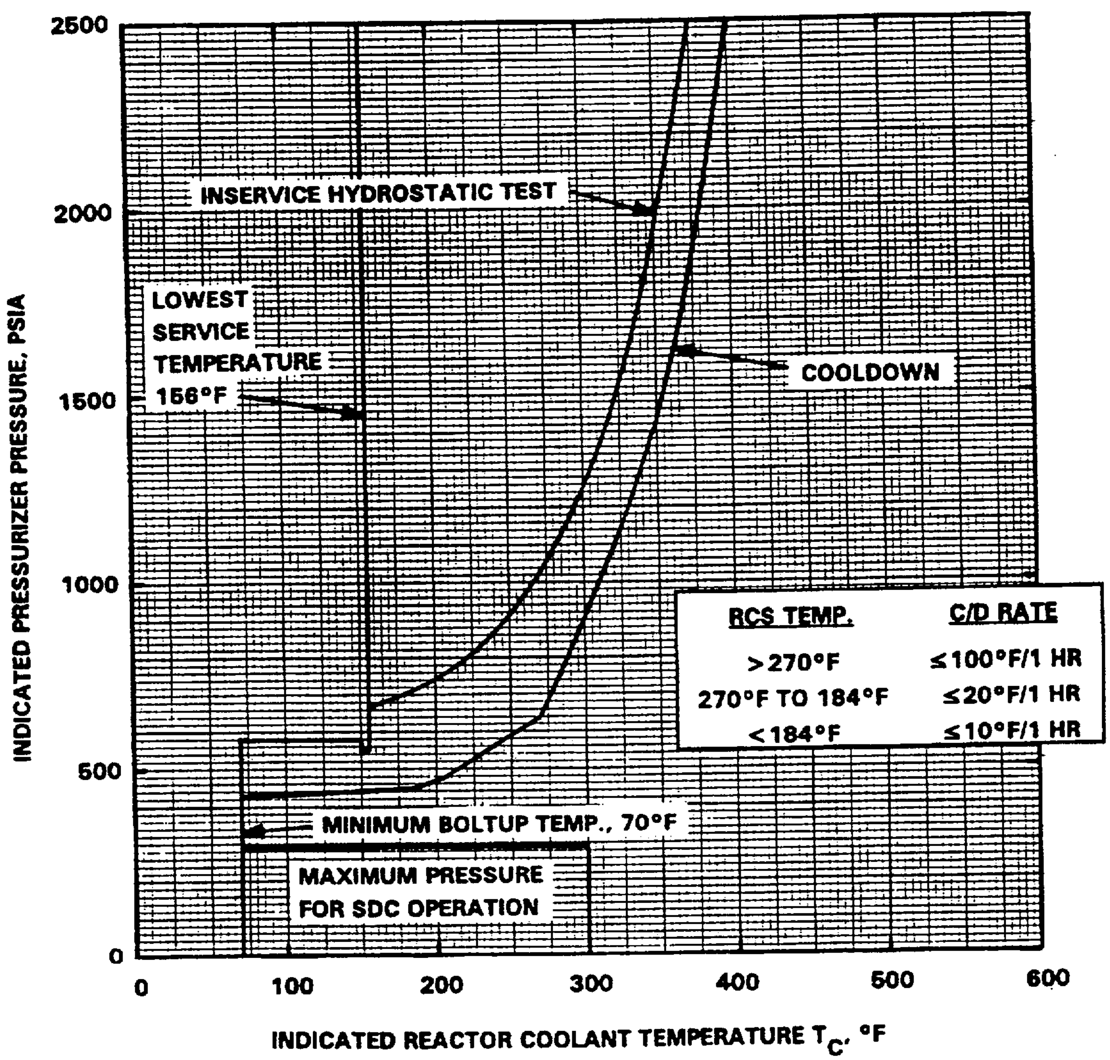

FIGURE 3.4.9-2

CALVERT CLIFFS UNIT 1 COOLDOWN CURVE, FOR FLUENCE ≤ 2.61 x 10^{19} n/cm^2
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

Overpressure Protection Systems

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following overpressure protection requirements shall be met:

a. One of the following three Overpressure Protection Systems shall be in place:

1. Two power-operated relief valves (PORVs) with a trip setpoint below the curve in Figure 3.4.9-3*, or

2. A single PORV with a trip setpoint below the curve in Figure 3.4.9-3* and a Reactor Coolant System vent of ≥ 1.3 square inches, or

3. A Reactor Coolant System (RCS) vent ≥ 2.6 square inches.

b. Two high pressure safety injection (HPSI) pumps# shall be disabled by either removing (racking out) their motor circuit breakers from the electrical power supply circuit, or by locking shut their discharge valves.

c. The HPSI loop motor operated valves (MOVs)# shall be prevented from automatically aligning HPSI pump flow to the RCS by placing their hand switches in pull-to-override.

d. No more than one **OPERABLE** high pressure safety injection pump with suction aligned to the Refueling Water Tank may be used to inject flow into the RCS and when used, it must be under manual control and one of the following restrictions shall apply:

1. The total high pressure safety injection flow shall be limited to ≤ 210 gpm, or

2. A Reactor Coolant System vent of ≥ 2.6 square inches shall exist.

e. When not in use, the above **OPERABLE** high pressure safety injection pump shall have its handswitch in pull-to-lock.

APPLICABILITY: When the RCS temperature is ≤ 365°F and the RCS is vented to < 8 square inches.

---

* When on shutdown cooling, the PORV trip setpoint shall be ≤ 429 psia.

\# EXCEPT when required for testing.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to **OPERABLE** status within 5 days or depressurize and vent the RCS through a ≥ 1.3 square inch vent(s) within the next 48 hours; maintain the RCS in a vented condition until both PORVs have been restored to **OPERABLE** status.

b. With both PORVs inoperable, depressurize and vent the RCS through a ≥ 2.6 square inch vent(s) within 48 hours; maintain the RCS in a vented condition until either one **OPERABLE PORV** and a vent of ≥ 1.3 square inches has been established or both PORVs have been restored to **OPERABLE** status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. With less than two HPSI pumps# disabled, place at least two HPSI pump handswitches in pull-to-lock within fifteen minutes and disable two HPSI pumps within the next four hours.

e. With one or more HPSI loop MOVs# not prevented from automatically aligning a HPSI pump to the RCS, immediately place the MOV handswitch in pull-to-override, or shut and disable the affected MOV or isolate the affected HPSI header flowpath within four hours, and implement the **ACTION** requirements of Specifications 3.1.2.1, 3.1.2.3, and 3.5.3, as applicable.

f. With HPSI flow exceeding 210 gpm while suction is aligned to the RWT and an RCS vent of < 2.6 square inches exists,

   1. Immediately take action to reduce flow to less than or equal to 210 gpm.

   2. Verify the excessive flow condition did not raise pressure above the maximum allowable pressure for the given RCS temperature on Figure 3.4.9-1 or Figure 3.4.9-2.

\# EXCEPT when required for testing.

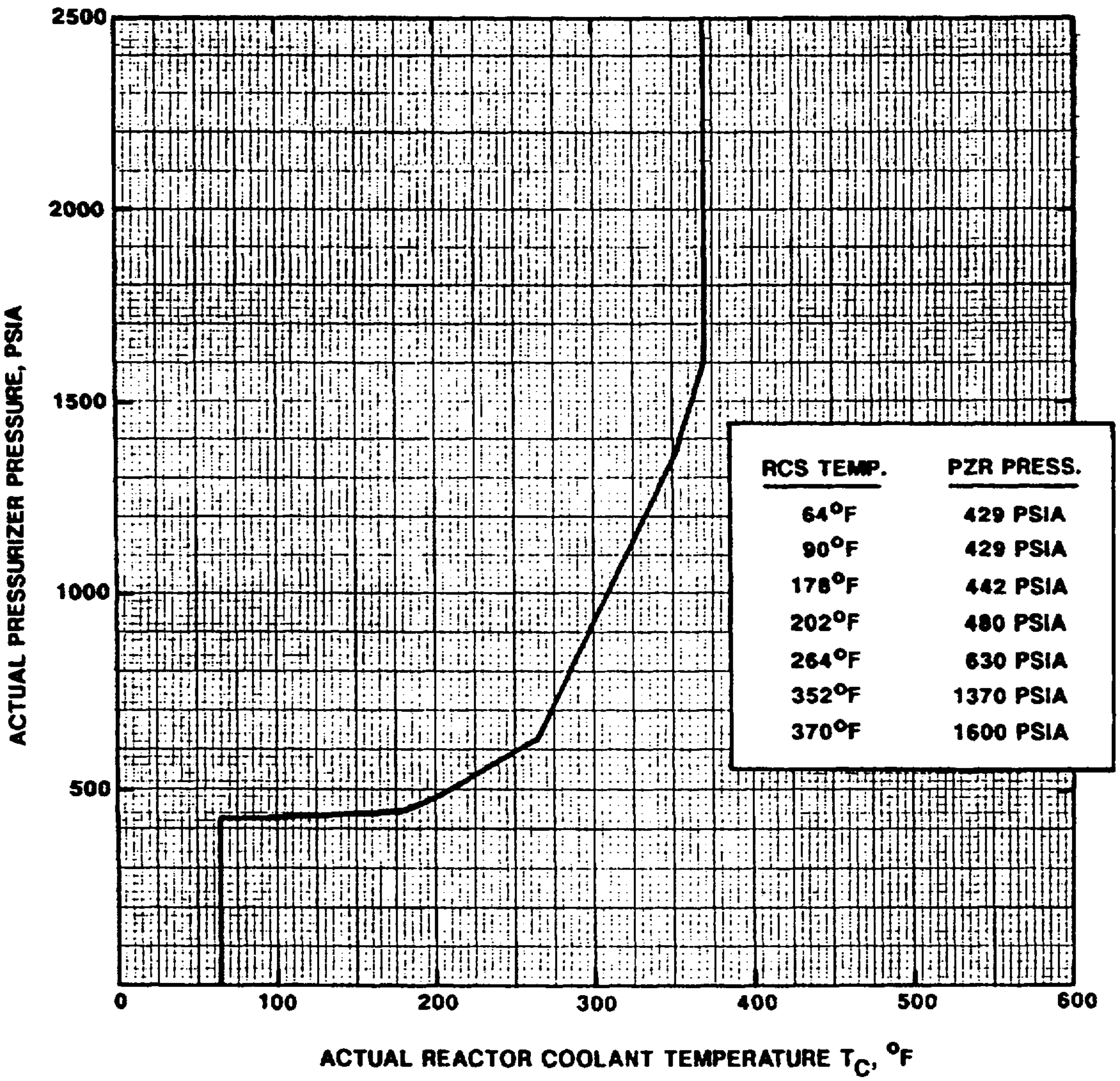

FIGURE 3.4.9-3

CALVERT CLIFFS UNIT 1, FOR FLUENCE $\leq 2.61 \times 10^{19}$ n/cm^2
MAXIMUM PORV OPENING PRESSURE vs TEMPERATURE

3/4.4.10 STRUCTURAL INTEGRITY

ASME Code Class 1, 2 and 3 Components

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

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

4.4.10.1.2 Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping - The unencapsulated welds greater than 4 inches in nominal diameter in the main steam and main feedwater piping runs located outside the containment and traversing safety related areas or located in compartments adjoining safety related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition and Addenda through Summer 1983, for Class 2 components.

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

3/4.4.11 CORE BARREL MOVEMENT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: **MODE** 1.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

3/4.4.12 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.

3/4.4.13 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.13 One Reactor Coolant System vent path consisting of two solenoid valves in series shall be **OPERABLE** and closed at each of the following locations:

a. Reactor vessel head

b. Pressurizer vapor space

APPLICABILITY: **MODES** 1 and 2

ACTION:

a. With the reactor vessel head vent path inoperable, maintain the inoperable vent path closed with power removed from the actuator of the solenoid valves in the inoperable vent path, and:

 1. If the pressurizer vapor space vent path is also inoperable, restore both inoperable vent paths to **OPERABLE** status within 72 hours or be in at least **HOT STANDBY** within 6 hours, or

 2. If the pressurizer vapor space vent path is **OPERABLE**, restore the inoperable reactor vessel head vent path to **OPERABLE** status within 30 days or be in at least **HOT STANDBY** within 6 hours.

b. With only the pressurizer vapor space vent path inoperable, maintain the inoperable vent path closed with power removed from the valve actuator of the solenoid valves in the inoperable vent path, and:

 1. Verify at least one PORV and its associated flow path is **OPERABLE** within 72 hours and restore the inoperable pressurizer vapor space vent path to **OPERABLE** status prior to entering **MODE** 2 following the next **HOT SHUTDOWN** of sufficient duration, or

 2. Restore the inoperable pressurizer vapor space vent path to **OPERABLE** status within 30 days, or be in at least **HOT STANDBY** within 6 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.13.1 Each Reactor Coolant System vent path shall be demonstrated **OPERABLE** by testing each valve in the vent path per Specification 4.0.5.

4.4.13.2 Each Reactor Coolant System vent path shall be demonstrated **OPERABLE** at least once per **REFUELING INTERVAL** by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Verifying flow through the Reactor Coolant System vent paths with the vent valves open.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated **OPERABLE***:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
1. MOV-659	Mini-flow Isolation	Open
2. MOV-660	Mini-flow Isolation	Open
3. CV-306	Low Pressure SI Flow Control	Open

b. At least once per 31 days by:

1. Verifying that upon a Recirculation Actuation Test Signal, the containment sump isolation valves open.

2. 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. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing **CONTAINMENT INTEGRITY**, and

2. Of the areas affected within containment at the completion of containment entry when **CONTAINMENT INTEGRITY** is established.

d. Within 4 hours prior to increasing the RCS pressure above 1750 psia by verifying, via local indication at the valve, that CV-306 is open.

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* Whenever flow testing into the RCS is required at RCS temperatures of 365°F and less, the high pressure safety injection pump shall recirculate RCS water (suction from RWT isolated) or the controls of Technical Specification 3.4.9.3 shall apply.

3/4.5 **EMERGENCY CORE COOLING SYSTEMS (ECCS)**

3/4.5.3 ECCS SUBSYSTEMS - **MODES** 3 (< 1750 PSIA) AND 4

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be **OPERABLE:**

a. One# **OPERABLE** high-pressure safety injection pump, and

b. An **OPERABLE** flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: **MODES** 3* and 4.

ACTION:

a. With no ECCS subsystem **OPERABLE**, restore at least one ECCS subsystem to **OPERABLE** status within 1 hour or be in **COLD SHUTDOWN** within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated **OPERABLE** per the applicable Surveillance Requirements of 4.5.2.

Between 385°F and 365°F, a transition region exists where the **OPERABLE** HPSI pump will be placed in pull-to-lock on a cooldown and restored to automatic status on a heatup. At 365°F and less, the required **OPERABLE** HPSI pump shall be in pull-to-lock and will not start automatically. At 365°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

* With pressurizer pressure < 1750 psia.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.195 during all normal operations and anticipated transients.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in **MODES** 2 and 3; however, single failure considerations require plant shutdown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

In **MODES 4** and 5, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be **OPERABLE**. Thus, if the reactor coolant loops are not **OPERABLE**, this specification requires two shutdown cooling loops to be **OPERABLE**.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevents stratification and produces gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump during **MODES** 3, 4 and 5 with the RCS temperature $\leq$ 365°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary System, which could exceed the limits of 10 CFR Part 50, Appendix G (see Bases 3/4.4.9). For starting the reactor coolant pumps the following criteria apply; (1) restricting the water volume in the pressurizer (170 inches) and thereby providing a volume for the primary coolant to expand into and (2) by restricting starting of the RCPs to when the indicated secondary water temperature of each steam generator is less than or equal to 30°F above the Reactor Coolant System temperature, (3) limit the initial indicated pressure of the pressurizer to less than or equal to 300 psia. The limit on initial pressurizer pressure will prevent the PORV from lifting during the pressure transient.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS form being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve approximately 3 x 10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are **OPERABLE**, an operating

Figure 3.4.8-1 increase the 2 hour thyroid dose at the **SITE BOUNDARY** by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and **STARTUP** and shutdown operation. The various categories of load cycles used for design purposes are provided in Section 4.1.1 of the UFSAR. During **STARTUP** and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

Operation within the appropriate heatup and cooldown curves assures the integrity of the reactor vessel against fracture induced by combinative thermal and pressure stresses. As the vessel is subjected to increasing fluence, the toughness of the limiting material continues to decline, and ever more restrictive Pressure/Temperature limits must be observed. The current limits, Figures 3.4.9-1 and 3.4.9-2, are for a peak neutron fluence to the inner surface of the reactor vessel of $\leq 2.61 \times 10^{19} N/cm^2$ (E > 1 MeV). This fluence corresponds to the Pressurized Thermal Shock Screening Criteria defined in 10 CFR 50.61 for weld 2-203 A, B, C.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Section 4.1.5 of the UFSAR. Reactor operation and resultant fast neutron (E > 1 MeV) irradiation will cause an increase in the RT_{NDT}. The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4-13 and are approved by the NRC prior to implementation in compliance with the requirements of 10 CFR Part 50, Appendix H.

The shift in the material fracture toughness, as represented by RT_{NDT}, is calculated using Regulatory Guide 1.99, Revision 2. For a fluence of $2.61x10^{19}N/cm^2$, the adjusted reference temperature (ART) value at the 1/4 T position is 241.4°F. At the 3/4 T position the ART value is 181.0°F.

These values are used with procedures developed in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G to calculate heatup and cooldown limits in accordance with the requirements of 10 CFR Part 50, Appendix G.

To develop composite pressure-temperature limits for the heatup transient, the isothermal, 1/4 T heatup, and 3/4 T heatup pressure-temperature limits are compared for a given thermal rate. Then the most restrictive pressure-temperature limits are combined over the complete temperature interval resulting in a composite limit curve for the reactor vessel beltline for the heatup event. The composite pressure-temperature limit for the cooldown transient is developed similarly. The Appendix G limits in Figures 3.4.9-1 and 3.4.9-2 assume the following number of RCPs are running:

HEATUP

Indicated RCS Temperature	Maximum Number of RCPs Operating
70°F to 330°F	2
> 330°F	4

COOLDOWN

Indicated RCS Temperature	Maximum Number of RCPs Operating
> 350°F	4
350°F to 150°F	2
< 150°F	0

Both 10 CFR Part 50, Appendix G and ASME, Code Appendix G require the development of pressure-temperature limits which are applicable to inservice hydrostatic tests. The minimum temperature for the inservice hydrostatic test pressure can be determined by entering the curve at the test pressure (1.1 times normal operating pressure) and locating the corresponding temperature. This curve is shown for a fluence of $\leq 2.61x10^{19}N/cm^2$ on Figures 3.4.9-1 and 3.4.9-2.

Similarly, 10 CFR Part 50 specifies that core critical limits be established based on material considerations. This limit is shown on the heatup curve, Figure 3.4.9-1. Note that this limit does not consider the core reactivity safety analyses that actually control the temperature at which the core can be brought critical.

The Lowest Service Temperature is the minimum allowable temperature at pressures above 20% of the pre-operational system hydrostatic test pressure (625 psia). This temperature is defined as equal to the most limiting RT_{NDT} for the balance of the Reactor Coolant System components plus 100°F, per Article NB 2332 of Section III of the ASME Boiler and Pressure Vessel Code.

The horizontal line between the minimum boltup temperature and the Lowest Service Temperature is defined by the ASME Boiler and Pressure Vessel Code as 20% of the pre-operational hydrostatic test pressure. The change in the line at 150°F on Figure 3.4.9-2 is due to a cessation of RCP flow induced pressure deviation, since no RCPs are permitted to operate during a cooldown below 150°F.

The minimum boltup temperature is the minimum allowable temperature at pressures below 20% of the pre-operational system hydrostatic test pressure. The minimum is defined as the initial RT_{NDT} for the material of the higher stressed region of the reactor vessel plus any effects for irradiation per Article G-2222 of Section III of the ASME Boiler and Pressure Vessel Code. The initial reference temperature of the reactor vessel and closure head flanges was determined using the certified material test reports and Branch Technical Position MTEB 5-2. The maximum initial RT_{NDT} associated with the stressed region of the closure head flange is -10°F. However, in order to comply with the 10 CFR 50, Appendix G limits, the minimum allowable reactor vessel temperature with the reactor head attached is 70°F. Hence, the minimum boltup temperature used in Figures and 3.4.9-1 and 3.4.9-2.

The Low-Temperature Overpressure Protection (LTOP) System consists of administrative controls coupled with low-pressure setpoint PORVs. The administrative controls provide the first line of defense against overpressurization events; the PORVs provide a backup to the administrative controls. The following section discusses the bases for the PORV setpoint and administrative controls.

Low-Temperature Overpressure Protection uses a variable PORV setpoint to take advantage of the increased Appendix G limits at higher RCS temperatures. Reactor Coolant System temperature is measured at the cold leg RTDs. This provides an accurate temperature indication during forced circulation, and is also adequate for natural circulation. However, the T_{cold} RTDs are not accurate when on shutdown cooling because they are not in the flow stream. For this reason, the lowest PORV setpoint is maintained whenever on shutdown cooling. This setpoint, which is independent of RCS temperature, is manually set when shutdown cooling is initiated and maintained until forced circulation is established after the RCPs are started.

The PORV setpoint is chosen to protect the most limiting of the heatup or cooldown Appendix G limits. Figure 3.4.9-3 shows the maximum PORV opening pressure. This includes corrections for static and dynamic head, and pressure overshoot to account for PORV response time and the maximum pressurization rate. The actual PORV setpoint is controlled by procedure and accounts for device uncertainty, calibration uncertainty and loop drift.

The design basis events in the low temperature region are:

- An RCP start with hot steam generators; and,
- An inadvertent HPSI actuation with concurrent charging.

These transients are most severe when the RCS is initially water solid. Any measures which will prevent or mitigate the design basis events are sufficient for any less severe incidents. Therefore, this section will discuss the results of the RCP start and mass addition transient analyses. Also discussed is the effectiveness of a pressurizer steam bubble and a single PORV relative to mitigating the design basis events.

The RCP start transient is a severe LTOP challenge that can quickly exceed the Appendix G limits for a water solid RCS. Therefore, during water solid operations all four RCPs are tagged out of service and their motor circuit breakers are disabled. However, the transient is adequately mitigated by restricting three parameters: 1) the initial water volume in the pressurizer to 170 inches (indicated), thereby providing a volume for the primary coolant to expand into; 2) the indicated secondary water temperature for each steam generator to 30°F above the RCS temperature; and 3) the initial pressure of the pressurizer to 300 psia. With these restrictions in place, the transient is adequately controlled without the assistance of the PORVs. Failure to maintain one of the initial conditions could cause the PORVs to open following an RCP start.

The mass addition transient from HPSI or multiple charging pumps is a severe LTOP challenge for a water solid system due to PORV response time. To preclude this event from happening while water solid, all HPSI pumps and two charging pumps are tagged out-of-service during water solid operations.

Analyses were performed for a HPSI mass addition transient with concurrent charging and the expansion of the RCS water volume following loss of decay heat removal, assuming one PORV available (due to single-failure criteria). This mass addition, determined at the point when the RCS reached water solid conditions, must be less than the capability of a single PORV to limit the LTOP event. Sufficient overpressure protection results when the equilibrium pressure does not exceed the limiting Appendix G curve pressure. Because the equilibrium pressure exceeds the minimum Appendix G limit for full HPSI flow, HPSI flow is throttled to no more than 210 gpm indicated when the HPSI pump is used for mass addition. The HPSI flow limit includes allowances for instrumentation uncertainty, charging pump flow addition and RCS expansion following loss of decay heat removal. The HPSI flow is injected through only one HPSI loop MOV to limit instrumentation uncertainty. No more than one charging pump (44 gpm) is allowed to operate during the HPSI mass addition.

Three 100% capacity HPSI pumps are installed at Calvert Cliffs. Procedures will require that two of the three HPSI pumps be disabled (breakers racked out) at RCS temperatures less than or equal to 365°F and that the remaining HPSI pump handswitch be placed in pull-to-lock. Additionally, the HPSI pump normally in pull-to-lock shall be throttled to less than or equal to 210 gpm when used to add mass to the RCS. Exceptions are provided for ECCS testing and for response to LOCAs.

To provide single failure protection against a HPSI pump mass addition transient when in MPT enable, the HPSI loop MOV handswitches must be placed in pull-to-override so the valves do not automatically actuate upon receipt of a SIAS signal. Alternative actions, described in the **ACTION** statement, are to disable the affected MOV (by racking out its motor circuit breaker or equivalent), or to isolate the affected HPSI header. Examples of HPSI header isolation actions include; (1) de-energizing and tagging shut the HPSI header isolation valves; (2) locking shut and tagging all three HPSI pump discharge valves; and (3) disabling all three HPSI pumps.

RCS temperature, as used in the applicability statement, is determined as follows: (1) with the RCPs running, the RCS cold leg temperature is the appropriate indication, (2) with the Shutdown Cooling System in operation, the shutdown cooling temperature indication is appropriate, (3) if neither the RCPs or shutdown cooling is in operation, the core exit thermocouples are the appropriate indicators of RCS temperature.

3/4.4.10 STRUCTURAL INTEGRITY

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

3/4.4.11 CORE BARREL MOVEMENT

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

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

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

3/4.4.12 LETDOWN LINE EXCESS FLOW

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

3/4.4.13 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the Primary System that could inhibit natural circulation core cooling. The **OPERABILITY** of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer vapor space ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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 ensure that as a minimum, the assumptions used in the safety analyses are met and the subsystem **OPERABILITY** is maintained. The surveillance requirement for flow balance testing provides 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 for temperatures above 365°F 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.

At indicated RCS temperatures of 365°F and less, HPSI injection flow is limited to less than or equal to 210 gpm except in response to excessive reactor coolant leakage. With excessive RCS leakage (LOCA), make-up requirements could exceed an HPSI flow of 210 gpm. Overpressurization is prevented by controlling other parameters, such as RCS pressure and subcooling. This provides overpressure protection in the low temperature region. An analysis has been performed which shows this flow rate is more than adequate to meet core cooling safety analysis assumptions. HPSI pumps are not required to auto-start when the RCS is in the MPT enable condition. The Safety Injection Tanks provide immediate injection of borated water into the core in the event of an accident, allowing adequate time for an operator to take action to start a HPSI pump.

Surveillance testing of HPSI pumps is required to ensure pump **OPERABILITY**. Some surveillance testing requires that the HPSI pumps deliver flow to the RCS. To allow this testing to be done without increasing the potential for overpressurization of the RCS, either the RWT must be isolated or the HPSI pump flow must be limited to less than or equal to 210 gpm or an RCS vent greater than 2.6 square inches must be provided.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE NO. DPR-53

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-317

1.0 INTRODUCTION

By letter dated September 3, 1993, as supplemented February 1, 1994, the Baltimore Gas and Electric Company (the licensee) submitted a request for changes to the Calvert Cliffs Nuclear Power Plant, Unit No. 1 Technical Specifications (TSs). The requested changes would revise the heatup and cooldown curves and the low-temperature overpressure protection (LTOP) controls for Calvert Cliffs, Unit 1, to support modifications to the LTOP system that are scheduled for the spring 1994 refueling outage. The current design utilizes administrative controls and hardware to protect the 10 CFR Part 50, Appendix G, pressure temperature (P-T) limits from an LTOP event for reactor pressure vessel irradiation (accumulated neutron fluence) up to 22 effective full-power years (EFPY). The February 1, 1994, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The proposed heatup and cooldown curves and rates are based on projected fluence with no reference to the corresponding EFPY due to the fact that vessel embrittlement calculations are actually based on fluence and not EFPY. Therefore, it is more appropriate to base the heatup and cooldown curves on fluence.

To evaluate the P-T limits, the NRC staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; Generic Letter (GL) 88-11; Regulatory Guide (RG) 1.99, Revision 2; and Standard Review Plan (SRP) Section 5.3.2.

Appendix G to 10 CFR Part 50 requires that "...when the core is not critical pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code..." Appendix G also imposes requirements on the minimum temperature for criticality, the closure head flange, and hydrostatic pressure tests or leak tests.

Appendix H of 10 CFR Part 50 requires licensees to establish a surveillance program to monitor embrittlement of reactor vessel materials. The program includes capsules that contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline. Appendix H refers to the American Society for Testing and Materials Standards which, in turn, require that the capsules be installed in the vessel before startup and be removed from the reactor vessel periodically for testing. The test results may be used in calculating P-T limits.

GL 88-11 indicates that licensees may use the methods in RG 1.99, Revision 2, to predict the embrittlement effect of neutron irradiation on reactor vessel materials. The embrittlement effect is defined in terms of adjusted reference temperatures (ART), which is the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the unirradiated reference temperature, copper (Cu) and nickel (Ni) contents, fluence, and the calculational procedures.

SRP 5.3.2 describes a calculation of the P-T limit curves based on the principles of linear elastic fracture mechanics. SRP 5.3.2 calculation follows the methodology specified in Appendix G to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III.

In relation to the LTOP controls, the current LTOP system utilizes two pressurizer power operated relief valves (PORVs). When the LTOP system is enabled, each of the two PORVs is set to open at a reduced pressure. The present configuration of the minimum pressure and temperature (MPT) Enable temperature circuitry (with the single setpoint PORV), combined with the reactor coolant pump (RCP) operating curves, gives a small operating window with a "knee" at the MPT Enable temperature. The MPT Enable temperature is the reactor coolant system (RCS) temperature below which the LTOP controls are required to be in place to protect the Appendix G limits. In addition, the current LTOP system does not allow the use of one RCP in each coolant loop which could be used for recovery from certain postulated accidents.

A variable-setpoint low temperature overpressure protection (VLTOP) system will be installed to increase the allowable operating pressure band in the LTOP region and to increase flexibility in the use of RCPs. The VLTOP system uses a variable PORV setpoint to take advantage of increased Appendix G pressure limits at increased RCS temperatures. The new system will allow operators to cooldown to shutdown cooling (SDC) conditions while running one RCP in each loop. This system will significantly increase the operating window in the LTOP region.

2.0 EVALUATION

2.1 Appendix G Heatup and Cooldown Curves and Rates

The licensee determined that intermediate shell axial weld, 2-203 A, B, and C, was the limiting material. The chemistry for weld 2-203 used in the licensee's calculation was 0.21% Cu and 0.88% Ni. The licensee used a margin of 56 °F and an initial RT_{ndt} of -50 °F. The licensee calculated the limiting ARTs of 241.4 °F at the 1/4T location and 181 °F at the 3/4T location based on Position C.1 of RG 1.99.

The NRC staff has identified the same material, weld 2-203 A, B, and C, as limiting. The NRC staff verified that the Cu content, Ni content, initial RT_{ndt}, and margin used for weld 2-203 in the licensee's calculation are acceptable. Based on the above data and a neutron fluence of 2.61E19 n/cm^2 on the inside surface of the reactor, the NRC staff calculated the same ARTs as the licensee.

Based on SRP 5.3.2, the NRC staff verified that the proposed P-T limits for heatup, cooldown, criticality, and inservice hydrostatic test meet the requirements in Paragraphs IV.A.2 and IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50, also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 10 °F, the NRC staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation, hydrostatic pressure test and leak test.

The licensee has removed surveillance capsules 97° and 263° from Calvert Cliffs Unit 1 and has performed the required tests. The NRC staff has determined that the surveillance program has satisfied Appendix H to 10 CFR Part 50.

The NRC staff performed an independent analysis of the P-T limits to verify the licensee's proposed limits. The NRC staff has determined that the proposed P-T limits for heatup, cooldown, inservice hydrostatic test, and criticality are valid for neutron fluences equal or less than 2.61E19 n/cm$_2$, because the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11. Hence, the proposed P-T limits may be incorporated in the Calvert Cliffs Unit 1 TSs.

2.2 Fast Neutron Flux

The proposed heatup and cooldown curves and rates are based on the projected 10 CFR 50.61 limit on the fluence for the critical elements with no reference to corresponding EFPY. For the Unit 1 pressure vessel, the critical elements are the intermediate shell axial welds 2-203 A, B, and C, as noted above. Their estimated fluence value is 2.61×10^{19} n/cm^2. This value was estimated using the methods, approximations, and cross sections described in the 97° surveillance capsule report. The method is based on the discrete ordinates DOT-IV code using an S_8 geometrical quadrature approximation and a P_3 cross section scattering approximation. The nuclear cross sections were from the CASK set which is ENDF/B-IV based; and, the effective dosimeter reaction cross sections were based on the ENDF/B-V. The source distribution was computed using the Asea Brown Boves-Combustion Engineering (ABB-CE), ROCS/DIT/MC codes. DOT-IV has been benchmarked by ABB-CE. The above methodology, the cross sections and the approximations used, are in accordance with above mentioned NRC staff recommendations and, therefore, are acceptable.

2.3 Technical Specification Changes

As noted, the proposed TSs are based on projected fluence with no reference to EFPY. The reason for this change is that the licensee is planning further fast neutron leakage reduction in future cycles, thus, an estimate of the EFPY required to reach the limiting fluence is not feasible. We find that the change in the titles of Figures 3.4.9-1, 2, and 3 and in the Bases of TS 3/4.4 are essentially editorial changes, and therefore, are acceptable.

The changes involving the heatup and cooldowns curves and rates in TSs 3.4.9.1.a and 3.4.9.1.b, and the RCS P-T limits in Figures 3.4.9.1 and 3.4.9.2, provide the necessary protection for the Appendix G P-T limits as previously discussed in this evaluation and are, therefore, acceptable.

Additional TS changes are proposed to account for the revised heatup and cooldown rates and support a VLTOP system. Specifically, TSs 3.4.9.3.a.1 and 3.4.9.3.a.2 change the pressure limit and the footnote to Figure 3.4.9-3*, TSs 3.1.2.1, and 3.1.2.3. Table 3.3-3, TSs 3.4.1.2, 3.4.1.3, 3.4.9.3, 4.5.2, 3.5.3, Bases 3/4.4.1, Bases 3/4.4.9 and Bases 3/4.5.2 change the minimum Enable temperature from 355 °F to 365 °F. TS 3.5.3 and Table 3.3-3 change the higher Enable temperature when the high-pressure safety injection (HPSI) pumps are placed under manual control on cooldown (and return to automatic on heatup) from 355 °F - 375 °F to 365 °F - 385 °F, TS 3.4.9.3 Bases 3/4.4.9 and Bases 3/4.5.2 change the maximum allowable HPSI pump flow from 200 gpm to 210 gpm. Bases 3/4.4.1 coolant loop and coolant circulation, Bases 3/4.4.9 P-T limits and Bases 3/4.5.2 emergency core cooling system subsystems were changed to be consistent with the above.

These proposed changes are consistent with the revised heatup and cooldown rates and RCS P-T limits which assure that the Appendix G P-T limits are met and are, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Maryland State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 50963). 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.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) 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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