



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

December 18, 1990

Docket No. 50-318

Mr. G. C. Creel
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
MD Rts. 2 & 4
P. O. Box 1535
Lusby, Maryland 20657

Dear Mr. Creel:

SUBJECT: ISSUANCE OF AMENDMENT FOR CALVERT CLIFFS NUCLEAR POWER PLANT
UNIT 2 (TAC NO. 77840)

The Commission has issued the enclosed Amendment No. 131 to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated October 22, 1990.

This amendment replaces the existing 0-10 effective full power years (EFPY) and 10-40 EFPY heatup and cooldown curves with 0-12 EFPY heatup and cooldown curves. These curves are based on the final version of Regulatory Guide 1.99, Revision 2, and use Combustion Engineering methodology, which has been previously reviewed and approved. These new calculations also resulted in Technical Specification changes to the low temperature overpressure protection (LTOP) system; including the pressure operated relief valves (PORV) lift setpoint, the reactor coolant pump (RCP) controls, the high pressure safety injection (HPSI) operability and the HPSI controls which are reflected in this amendment. The supporting TS Bases were modified to reflect the changes.

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Mr. G. C. Creel

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

Sincerely,

A handwritten signature in black ink, appearing to read "Daniel G. McDonald". The signature is fluid and cursive, with a large initial "D" and "M".

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.131 to DPR-69
2. Safety Evaluation

cc: w/enclosures
See next page

Mr. G. C. Creel
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

cc:

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

December 18, 1990

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. DPR-69

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated October 22, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-69 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 131, 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


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: December 18, 1990

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 131 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Revise Appendix A as follows:

<u>Remove Pages</u>	<u>Insert Pages</u>
IX*	IX*
X	X
3/4 1-8	3/4 1-8
3/4 1-9*	3/4 1-9*
3/4 1-10	3/4 1-10
3/4 3-11	3/4 3-11
3/4 3-12*	3/4 3*-12*
3/4 4-1*	3/4 4-1*
3/4 4-2	3/4 4-2
3/4 4-2a	3/4 4-2a
3/4 4-2b	3/4 4-2b
3/4 4-23	3/4 4-23
3/4 4-24*	3/4 4-24*
3/4 4-25*	3/4 4-25*
3/4 4-25a	3/4 4-25a
3/4 4-25b	3/4 4-25b
3/4 4-26*	3/4 4-26*
3/4 4-27	3/4 4-27
3/4 4-27a	3/4 4-27a
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- - -	3/4 4-27c
3/4 4-28*	3/4 4-28*
3/4 5-3*	3/4 5-3*
3/4 5-4	3/4 5-4
3/4 5-6	3/4 5-6
3/4 5-7*	3/4 5-7*
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2*	B 3/4 4-2*
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12*	B 3/4 4-12*
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*Pages that did not change, but are overleaf.

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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 3.1-1 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 305°F and less, the required OPERABLE HPSI pump shall be in pull-to-lock and will not start automatically. At 305°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

REACTIVITY CONTROL SYSTEMS

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 refueling interval 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.

REACTIVITY CONTROL 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 305°F and less, the required OPERABLE HPSI pump shall be in pull-to-lock and will not start automatically. At 305°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
1. SAFETY INJECTION (SIAS) [@]					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Containment Pressure - High	4	2	3	1, 2, 3	7*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	7*
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Containment Pressure - High	4	2	3	1, 2, 3	11
3. CONTAINMENT ISOLATION (CIS) [#]					
a. Manual CIS (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Containment Pressure - High	4	2	3	1, 2, 3	7*

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 350°F, the required OPERABLE HPSI pumps must be able to start automatically upon receipt of a SIAS signal,
- (b) Between 350°F and 305°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 305°F and less, the required OPERABLE HPSI pump shall be in pull-to-lock and will not start automatically.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
4. MAIN STEAM LINE ISOLATION					
a. Manual (MSIV Hand Switches and Feed Head Isolation Hand Switches)	1/valve	1/valve	1/valve	1, 2, 3, 4	6
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	7*
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	7*

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2*.

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

* See Special Test Exception 3.10.3.

REACTOR COOLANT SYSTEM

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 #21 and at least one associated reactor coolant pump.
2. Reactor Coolant Loop #22 and at least one 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.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

* 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 305°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 320 psia.

REACTOR COOLANT SYSTEM

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 #21 and its associated steam generator and at least one associated reactor coolant pump,
 2. Reactor Coolant Loop #22 and its associated steam generator and at least one associated reactor coolant pump,
 3. Shutdown Cooling Loop #21*,
 4. Shutdown Cooling Loop #22*.
- 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 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.
- b. With no 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 coolant loop to operation within one hour.

* 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 305°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 320 psia.

See Special Test Exception 3.10.5.

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

SHUTDOWN

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required shutdown cooling loop(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability for pumps and shutdown cooling loop valves.

4.4.1.3.2 The required steam generator(s), if it is being used to meet 3.4.1.3.a, shall be determined OPERABLE by verifying the secondary side water level to be above -50 inches at least once per 12 hours.

4.4.1.3.3 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

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 Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

- a. A maximum heatup of 75°F in any one hour period,
- b. A maximum cooldown of:

Maximum Allowable Cooldown Rate

RCS Temperature

100°F in any one hour period
40°F in any one hour period
15°F in any one hour period

> 180°F
180°F to 140°F
< 140°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.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

DELETED

FIGURE 3.4-2b
CALVERT CLIFFS UNIT 2 HEATUP CURVE, 12 EFPPY
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS

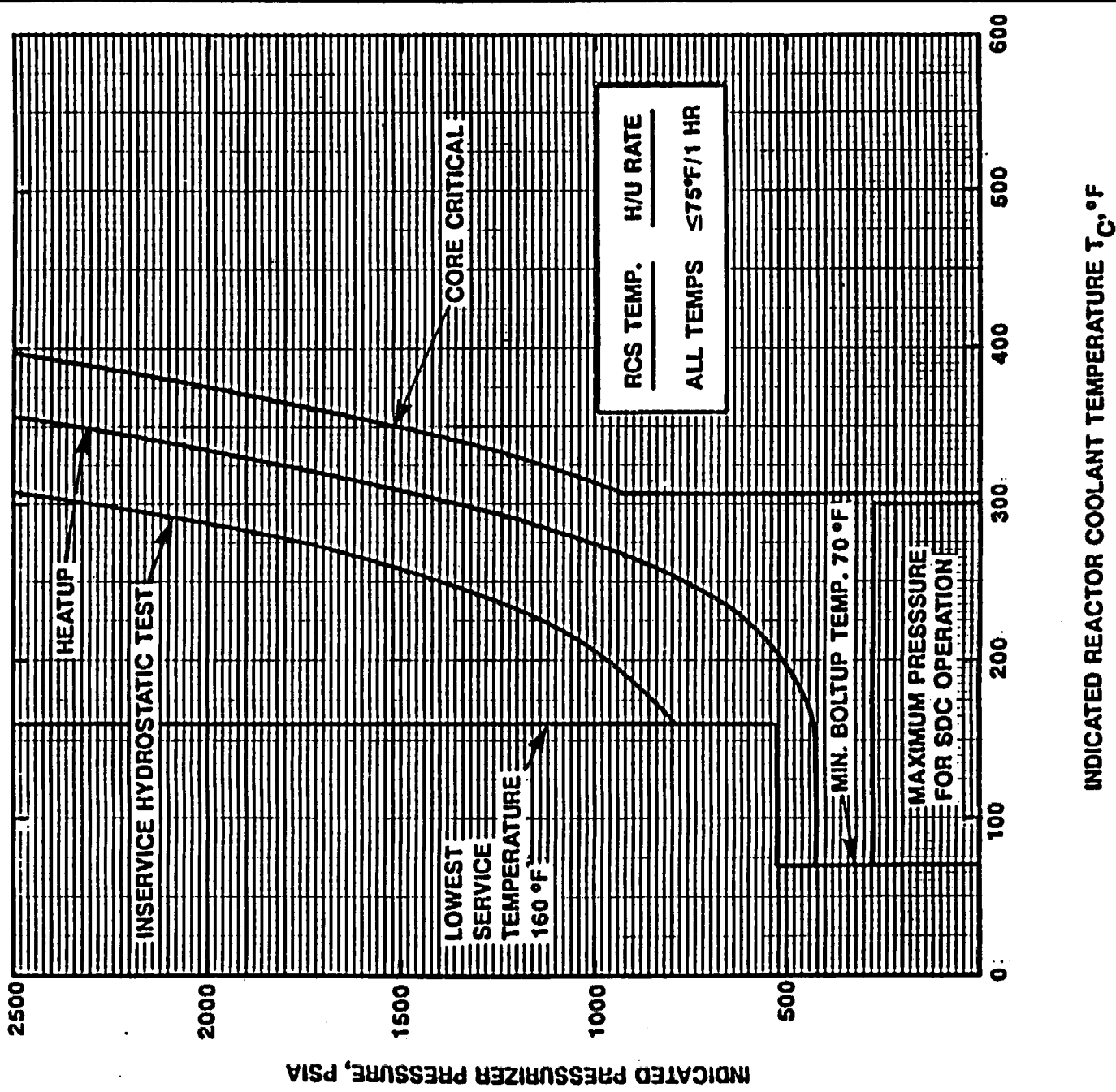


FIGURE 3.4-2c
CALVERT CLIFFS UNIT 2 COOLDOWN CURVE, 12 EFY
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS

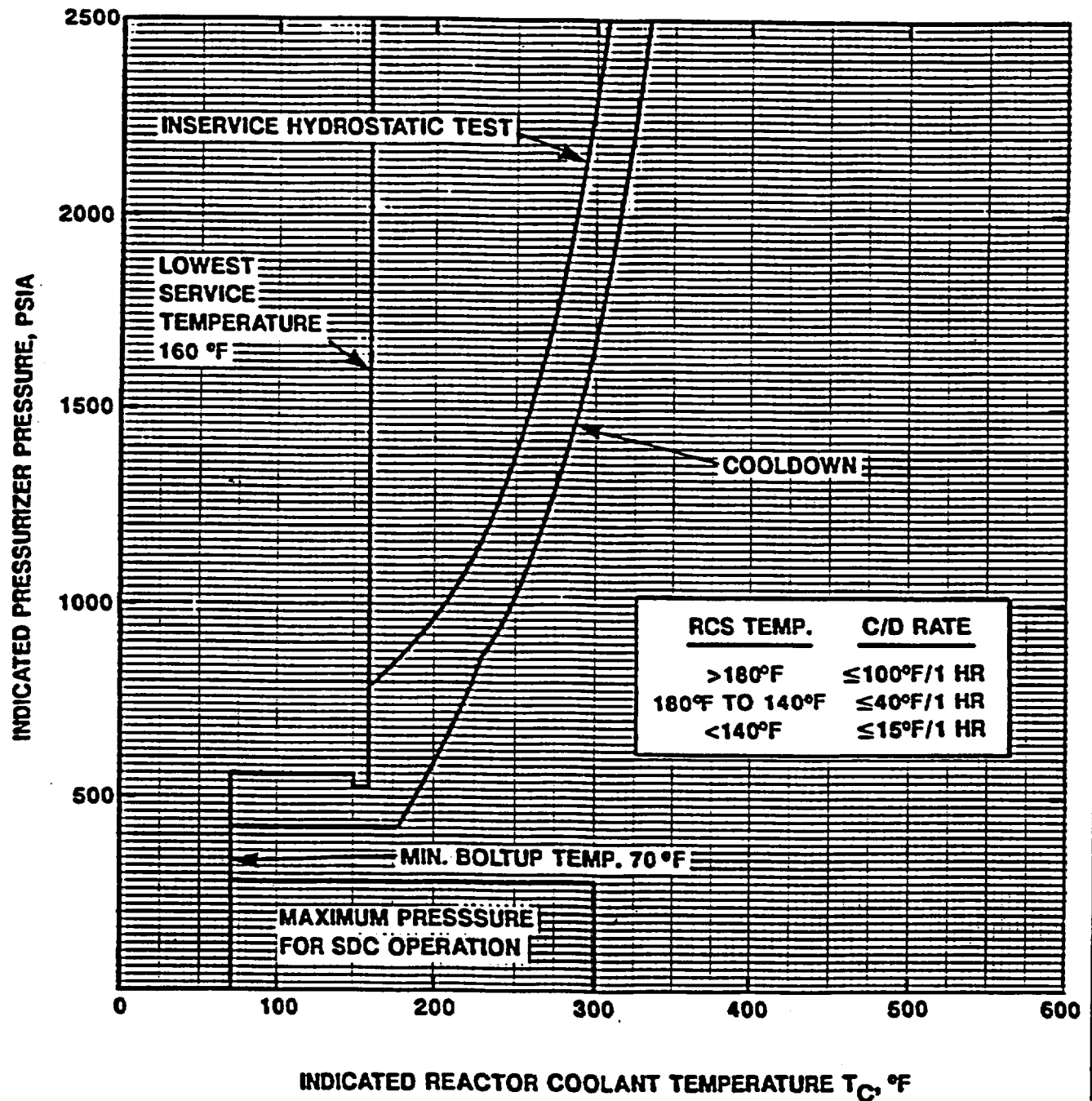


TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

<u>SPECIMEN</u>	<u>REMOVAL INTERVAL</u>
1. Capsule No. 1	5 years
2. Capsule No. 2	14 years
3. Capsule No. 3	23 years
4. Capsule No. 4	30 years
5. Capsule No. 5	35 years
6. Capsule No. 6	40 years

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

- a. A maximum heatup of 100°F in any one hour period,
- b. A maximum cooldown of 200°F in any one hour period, and
- c. A maximum spray water temperature differential of 400°F.

APPLICABILITY: At all times

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 300 psia within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

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 lift setting of ≤ 430 psia, or
 2. A single PORV with a lift setting of ≤ 430 psia 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 (MOV)s# shall be prevented from automatically aligning HPSI pump flow to the RCS by placing their handswitches 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.

APPLICABILITY: When the RCS temperature is $\leq 305^{\circ}\text{F}$ and the RCS is vented to < 8 square inches.

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.

Except when required for testing.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

- c. In the event either the PORVs or the RCS vent(s) are used to mitigate an 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 3.4-2b or Figure 3.4-2c.
 - 3. If a pressure limit was exceeded, take action in accordance with Specification 3.4.9.1.
- g. The provisions of Specification 3.0.4 are not applicable.

[#] Except when required for testing.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

- a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.
- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.
- c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.
- d. Testing in accordance with the inservice test requirements for ASME Category C valves pursuant to Specification 4.0.5.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

4.4.9.3.3 All high pressure safety injection pumps, except the above OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours by verifying that the motor circuit breakers have been removed from their electrical power supply circuits or by verifying their discharge valves are locked shut. The automatic opening feature of the high pressure safety injection loop MOVs shall be verified disabled at least once per 12 hours.

* Except when the vent pathway is locked, sealed, or otherwise secured in the open position, then verify these vent pathways open at least once per 31 days.

REACTOR COOLANT SYSTEM

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.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 300^{\circ}\text{F}$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

- a. One OPERABLE high-pressure safety injection pump,
- b. One OPERABLE low-pressure safety injection pump, and
- c. 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 1, 2, and 3*.

ACTION:

- a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 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.

* With pressurizer pressure ≥ 1750 psia.

EMERGENCY CORE COOLING SYSTEMS

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:

<u>Valve Number</u>	<u>Valve Function</u>	<u>Valve Position</u>
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 suction 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.

* Whenever flow testing into the RCS is required at RCS temperatures of 305°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.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} < 300^{\circ}\text{F}$

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 Specification 4.5.2.

- * With pressurizer pressure < 1750 psia.
- # Between 350°F and 305°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 305°F and less, the required OPERABLE HPSI pump shall be in pull-to-lock and will not start automatically. At 305°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

- a. A minimum contained borated water volume of 400,000 gallons,
- b. A boron concentration of between 2300 and 2700 ppm,
- c. A minimum water temperature of 40°F, and
- d. A maximum solution temperature of 100°F in MODE 1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

- a. At least once per 7 days by:
 1. Verifying the contained borated water volume in the tank, and
 2. Verifying the boron concentration of the water.
- b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 40°F.

3/4.4 REACTOR COOLANT SYSTEM

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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, prevent stratification and produce 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 $\leq 305^{\circ}\text{F}$ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50 (see Bases 3/4.4.9). For operation of the reactor coolant pumps, the following criteria apply: (1) restrict the water volume in the pressurizer (170 inches) and thereby providing a volume for the primary coolant to expand into, and (2) restrict starting of the RCPs to when the indicated secondary water temperature of each steam generator is less than or equal to 30°F above Reactor Coolant System temperature, and (3) limit the initial indicated pressure of the pressurizer to less than or equal to 320 psia.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve approximately 3×10^5 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 shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to

REACTOR COOLANT SYSTEM

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Limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer with the level as programmed ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The operating band for pressurizer level bounds the programmed level and ensures that RCS pressure remains within the bounds of an analyzed condition during the excessive charging event as well as during the limiting depressurization event, Excess Load. The operating band also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valves function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer-Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to

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steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Calvert Cliffs site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity >1.0 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4-1 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 $< 500^{\circ}\text{F}$ 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

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 even more restrictive Pressure/Temperature limits must be observed. The current limits, Figures 3.4-2b and 3.4-2c, are for up to and including 12 Effective Full Power Years (EFPY) of operation.

The shift in the material fracture toughness, as represented by RT_{NDT} , is calculated using Regulatory Guide 1.99, Revision 2. For 12 EFPY, at the $1/4$ T position, the adjusted reference temperature (ART)

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value is 171°F. At the 3/4 T position the ART value is 125°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.

To develop a composite pressure-temperature limit for the cooldown event, the isothermal pressure-temperature limit must be calculated. The isothermal pressure-temperature limit is then compared to the pressure-temperature limit associated with a cooling rate and the more restrictive allowable pressure-temperature limit is chosen resulting in a composite limit curve for the reactor vessel beltline.

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 12 EFPY on Figures 3.4-2b and 3.4-2c.

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 3.4-2b. 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 the cooldown curve is due to a cessation of RCP flow induced pressure deviation, since no RCPs are permitted to operate during a cooldown below 150°F.

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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 30°F . The minimum boltup temperature including temperature instrument uncertainty is $30^{\circ}\text{F} + 10^{\circ}\text{F} = 40^{\circ}\text{F}$. However, for conservatism, a minimum boltup temperature of 70°F is utilized in the analysis to establish the low temperature PORV lift setpoint.

The design basis events in the low temperature region assuming a water solid system are:

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

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 for a water solid RCS. Therefore, during water solid operations all four RCPs are tagged out of service. Analysis indicates the transient is adequately controlled by placing restrictions on three parameters: initial pressurizer pressure and level, and the secondary-to-primary temperature difference. With these restrictions in place, the transient is adequately controlled without the assistance of the PORVs.

The inadvertent actuation of one HPSI pump in conjunction with one charging pump is the most severe mass addition overpressurization event. Analyses were performed for a single HPSI pump and one charging pump assuming one PORV available with the existing orifice area of 1.29 in^2 . For the limiting case, only a single PORV is considered available due to single failure criteria. A figure was developed which shows the calculated RCS pressures versus time that will occur assuming HPSI and charging pump mass inputs, and the expansion of the RCS following loss of decay heat removal. Sufficient overpressure protection results when the

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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.

Comparison of the PORV discharge curve with the critical pressurizer pressure of 471.2 psia indicates that adequate protection is provided by a single PORV for RCS temperatures of 70°F or above when all mass input is limited to 380 gpm. HPSI discharge is limited to 210 gpm to allow for one charging pump and system expansion due to loss of decay heat removal. The low temperature PORV pressure lift setpoint is set to protect the most restrictive Appendix G pressure limit (471.2 psia). A PORV setpoint of 430 psia, which includes instrumentation uncertainties and sufficient margins for PORV response time requirements necessary for the protection of 471.2 psia, was selected.

To provide single failure protection against a HPSI pump mass addition transient, 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 MOVs; and (3) disabling all three HPSI pumps.

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 305°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.

A pressurizer steam volume and a single PORV will provide satisfactory control of all mass addition transients with the exception of a spurious actuation of full flow from a HPSI pump. Overpressurization due to this transient will be precluded for temperatures 305°F and less by disabling two HPSI pumps, placing the third in pull-to-lock, and by throttling the third pump to less than or equal to 210 gpm flow when it is used to add mass to the RCS.

Note that only the design bases events are discussed in detail since the less severe transients are bounded by the RCP start and inadvertent HPSI actuation analysis.

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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.

DELETED

DELETED

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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 computer from the ADP 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 contained 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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

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 protection 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 ensure 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.

EMERGENCY CORE COOLING SYSTEMS

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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 ≥ 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 at 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 305°F are based upon small break LOCA calculations which credit charging pump flow following a 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 Specifications 3/4.1.2. Specification of safety injection pump total developed head ensures pump performance is consistent with safety analysis assumptions.

At temperatures of 305°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 a 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 or equal to 2.6 square inches must be provided.

EMERGENCY CORE COOLING SYSTEMS

BASES

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

DOCKET NO. 50-318

1.0 INTRODUCTION

By letter dated October 22, 1990, the Baltimore Gas and Electric Company (the licensee) proposed to amend the Technical Specifications of the Calvert Cliffs Nuclear Power Plant, Unit 2. In its submittal, the licensee provided Technical Specification changes to support 10 CFR Part 50, Appendix G, heatup and cooldown Pressure/Temperature (P/T) limits applicable to the Unit 2 reactor vessel for a period up to 12 effective full power years (EFPY).

The proposed P/T limits were developed based on Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system (RCS) during heatup, cooldown, criticality, and inservice hydrostatic testing. In addition, the proposed changes included revised heatup and cooldown rates, a change in the Power Operated Relief Valve (PORV) pressure setpoint for Low Temperature Overpressure Protection (LTOP), a change in the LTOP enable temperature, a modification to Reactor Coolant Pump (RCP) controls when in LTOP conditions, a modification to High Pressure Safety Injection (HPSI) pump controls when in LTOP conditions, and changes to the Bases for the affected Limiting Conditions for Operation (LCOs) to reflect the proposed changes.

To evaluate the P/T limits and supporting changes, the staff used the following NRC regulations and guidance: Appendices G and H to 10 CFR Part 50; the American Society of Testing Materials (ASTM) Standards and the American Society of Mechanical Engineers (ASME) Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Sections 5.2.2 and 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for

all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements of reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires that the licensee establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before initial plant startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

LTOP is provided by the PORVs on the pressurizer. These PORVs are set at a pressure low enough to prevent violation of the 10 CFR Part 50, Appendix G, P/T limits during heatup and cooldown should a RCS pressure transient occur during low temperature operations. The potential for overpressurization of the RCS can be minimized by a combination of administrative procedures and operator actions. However, because operator action cannot always be assumed, and because possible equipment malfunctions must be considered, additional controls must be in place to ensure adequate protection exists for all postulated events.

The two major concerns for LTOP protection are the mass addition and energy addition transients. The proposed amendment provides restrictions on the use of HPSI pumps to provide protection for mass addition transients. Restrictions are also imposed on the starting and use of the RCPs to provide protection for energy addition transients.

The revised Regulatory Guide 1.99 results in more restrictive P/T limits. To meet the revised requirements a new LTOP pressure setpoint and new heatup and cooldown rates are proposed. These new values are such as to ensure that (1) given a limiting mass or energy input to the RCS during normal operation, anticipated operational occurrences, and hydrostatic testing; the Appendix G pressure-temperature limits are not challenged, and (2) operational flexibility is maintained.

2.0 EVALUATION - APPENDIX G-HEATUP AND COOLDOWN P/T LIMITS

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Calvert Cliffs 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 12 EFPY was the intermediate shell longitudinal welds 2-203A, B, and C with 0.12% copper (Cu), 1.01% nickel (Ni), and an initial RT_{ndt} of -56°F.

The licensee has removed one surveillance capsule from Calvert Cliffs 2. The results from capsule 263 were published in Southwest Research Institute Report SWRI-7524. The surveillance capsule contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline materials, intermediate shell longitudinal weld 2-203A, B, and C, the staff calculated the ART to be 170.8°F at 1/4T (T = reactor vessel beltline thickness) and 124.8°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 1.007E19 n/cm² at 1/4T and 3.58E18 n/cm² at 3/4T. The ART was determined by the Section 1 of RG 1.99, Revision 2, because only one surveillance capsule has been removed from the Calvert Cliffs 2 reactor vessel.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 171°F at 12 EFPY at 1/4T for the same limiting weld metal. The staff judges that the licensee's ART 171°F is more conservative than the staff's ART of 170.8°F, and it is acceptable. Substituting the ART of 170.8°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% 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 30°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The intermediate shell plate D8906-1 (Heat No. C-4463-1) has the lowest (limiting) unirradiated USE of 76.7 ft-lb among all beltline materials. Using an end of life peak fluence of $2.72E19$ n/cm² at 1/4T, the staff calculated an USE of 53.7 ft-lb for plate D8906-1 at end of life. This is above the 50 ft-lb requirement, and is acceptable.

The staff has determined that the proposed P/T limits for the reactor coolant system for heatup, cooldown, inservice hydrostatic test, leak test, and criticality are valid through 12 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Revision 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Calvert Cliffs 2 Technical Specifications.

3.0 EVALUATION - LTOP CONTROLS

The PORV lift setpoint is estimated at 430.0 psia to protect the most restrictive pressure of 471.2 psia which corresponds to a rate of 15°F/hr at 70°F in the RCS. The difference in the setting and the protect pressure is due to instrumentation uncertainty and PORV response time allowances. The LTOP enable temperature is 305°F and was estimated using the Standard Review Plan 5.2.2, Revision 2, for heatup rates to 75°F/hr. Based on the conservative assumptions and approved methods used, the PORV lift setpoint and the LTOP enable temperature are acceptable.

3.1 HPSI Pump Controls

Overpressurization events due to mass addition, in their most limiting case include: HPSI pump flow, charging pump flow and the coolant expansion due to loss of decay heat removal. The only controllable component in this case is the HPSI flow. Thus, the maximum PORV flow determines the HPSI flow after the charging pump and the expansion equivalent have been subtracted and the instrumentation uncertainty been accounted. In this manner a total flow limit of 380 gpm yields an HPSI indicated flow of 210 gpm to ensure an Appendix G pressure limit of 471.2 psia in the pressurizer. The HPSI flow rate was then compared to the requirements of other design basis events. The most limiting such event is the loss of shutdown cooling which requires an actual flow rate of 175 gpm to prevent core uncovering. The proposed flow of 210 gpm meets this limiting design requirement and is, therefore, acceptable.

3.2 RCP Controls

RCP start is the primary concern for the limiting energy addition LTOP transient. In this case we assume RCP start with: letdown isolation, energy addition from two RCPs, energy addition from the pressurizer heaters, and loss of decay heat removal. Mitigation of such transients is provided by the initial pressurizer pressure, pressurizer level, and the steam generator primary-to-secondary change in temperature (ΔT). For two RCPs starting assuming; an initial pressurizer level of 170 inches, a steam generator ΔT of 30°F,

initial pressurizer pressure of 320 psia, decay heat at a level of 2 hours after shutdown, and no operator action, the pressurizer insurge peak pressure will be below the PORV setting, thus within the Appendix G acceptance criteria and is acceptable.

3.4 PORV Response Time

The PORV response time is part of the estimated assumptions for the Appendix G limits. For Unit 2 this response time has not been directly measured but assumed to be the same as for Unit 1. The justification for this assumption is that the designs are identical. For Unit 1 the maximum total response time is 0.49 seconds based on confirmatory analysis and testing. The licensee's confirmatory test results are consistent with results of similar tests performed by other utilities and with bench tests performed by EPRI. Based on the results of the Unit 1 tests and the other industry tests, the response time assumed for Unit 2 PORVs is acceptable.

4.0 TECHNICAL SPECIFICATION CHANGES

The licensee provided updated P/T curves in the proposed Technical Specification Figures 3.4-2b and 3.4-2c for heatup and cooldown, respectively. Technical Specification 3.4.9.1.a provides a new heatup rate of 75°F/hr for all temperatures. Technical Specification 3.4.9.1.b provides new cooldown rates as follows:

100°F/hr	for T_{ave} greater than 180°F
40°F/hr	for T_{ave} between 180°F and 140°F
15°F/hr	for T_{ave} less than 140°F

The Action Statements for Technical Specifications 3.4.9.1 and 3.4.9.2 are changed to reflect the proposed cooldown rates. The Bases for Technical Specification 3.4.9.1 have been changed to reflect the revisions in the heatup and cooldown rates. Technical Specification 3.4.9.3 proposes to lower the PORV lift setting to less than 430 psia, and require system vents equivalent to the PORVs for RCS temperatures less than 305°F (for system testing). In addition, two of the three HPSI pumps will be disabled and the HPSI loop motor operated valves be prevented from aligning pump flow to the RCS for RCS temperatures less than 305°F. For one HPSI pump operable, the total flow will be throttled to 210 gpm. The above are not applicable if a system vent greater than 8 square inches exist. The current action times are 7 days to restore a PORV to "Operable" status or must be vented (depressurized) within 8 hours. This is changed to 5 days to restore the PORV to operable and 48 hours to vent. The total time is still the same (7 days) but the venting time is increased for ease of operation. Surveillance requirements are added to verify system operability conditions. The Technical Specification 3.4.9.3 Bases have also been changed to reflect the proposed set of conditions.

Several Technical Specifications have been changed to reflect the new requirements on the HPSI pumps. A footnote is added to Technical Specifications 3.1.2.1 and 3.1.2.3 which defines an "operable" HPSI pump. A footnote is added to Technical Specification 3.5.3 which states that a maximum of one HPSI pump shall be operable when the RCS temperature is less or equal to 305°F. A footnote is added to the surveillance requirements of Technical Specification 4.5.2 allowing full flow testing of a HPSI pump. A footnote is also added to Table 3.3-3 providing information on HPSI pump operation. The Technical Specification 3.5.3 Bases have also been changed to reflect the new requirements on the operation of the HPSI pumps.

The following changes reflect the new restrictions on the RCP operation. A footnote to Technical Specification 3.4.1.3 is changed to require that an RCP not be started if the RCS temperature is less (or equal) to 305°F unless; (1) the pressurizer level is less or equal to 170 inches, (2) the primary to secondary steam generator delta-T is less than or equal to 30°F, and (3) the pressurizer pressure is less than or equal to 320 psia. A footnote is added to Technical Specification 3.4.1.2 to provide RCP start control consistent with that of 3.4.1.3. Finally, the bases of 3/4.4.1 and 3/4.4.9 are changed to reflect the new requirements.

5.0 SUMMARY

The proposed P/T limits for the RCS for heatup, cooldown, inservice hydrostatic testing, and criticality are valid through 12 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The P/T limits also satisfy Generic Letter 88-11 because the licensee used the methods in RG 1.99, Revision 2, to calculate the ART. Conservative assumptions and approved methodology were used for the LTOP analyses. The analyses defined the HPSI pump and RCP control limitations. We found these proposed revisions acceptable. In addition, Technical Specification changes were defined which correctly reflect the new limitations and restrictions.

The staff has concluded, based on the above and details provided in Sections 2, 3, and 4, that the proposed Technical Specifications and Bases supporting the new 12 EFPY P/T limits and LTOP controls are acceptable.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of the facilities' components located within the restricted areas as defined in 10 CFR Part 20 and to a surveillance requirement. The staff has determined that this 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

7.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 18, 1990

PRINCIPAL CONTRIBUTORS:

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December 18, 1990

A copy of the related Safety Evaluation is enclosed, Notice of Issuance will be included in the Commission's next regular bi-weekly 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. 131 to DPR-69
2. Safety Evaluation

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