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. 188
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 1, 1992, as supplemented March 17, 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) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 188, 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

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: April 20, 1994

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 165
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 September 1, 1992, as supplemented March 17, 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-69 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 165 , 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

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: April 20, 1994

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 188 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 165 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Revise Appendix A, DPR-53, as follows:

Remove Pages	Insert Pages
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Revise Appendix A, DPR-69, as follows:

Remove Pages	Insert Pages
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*These pages are text rollover pages with no changes as the result of this amendment.

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.3 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3 Two power-operated relief valves (PORVs) and their associated block valves shall be **OPERABLE**.

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

ACTION:

a. If one or both PORV(s) has excessive seat leakage, within 1 hour close the associated block valve(s) and maintain power to the block valve(s).

b. With one PORV inoperable due to causes other than excessive PORV seat leakage, within 1 hour either restore the PORV to **OPERABLE** status or close the associated block valve and remove power from the block valve; restore the PORV to **OPERABLE** status within the following 5 days or be in **HOT STANDBY** within the next 12 hours and at or below 365°F within the following 24 hours.

c. With both PORVs inoperable due to causes other than excessive PORV seat leakage, within 1 hour either restore one PORV to **OPERABLE** status or close the associated block valve and remove power from the block valve; restore one PORV to **OPERABLE** status within the following 72 hours or be in **HOT STANDBY** within the next 12 hours and at or below 365°F within the following 24 hours.

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to **OPERABLE** status or place its associated PORV(s) in override closed. Restore at least one block valve to **OPERABLE** status within the next 72 hours if both block valves are inoperable; restore any remaining inoperable block valve to **OPERABLE** status within the following 5 days; otherwise, be in at least **HOT STANDBY** within the next 12 hours and at or below 365°F within the following 24 hours.

e. The provisions of Specification 3.0.4 are not applicable.

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* Above 365°F. At or below 365°F, Specification 3/4.4.9.3 applies.

SURVEILLANCE REQUIREMENTS

4.4.3.1 Each PORV shall be demonstrated **OPERABLE:**

a. At least once per 31 days by performance of a **CHANNEL FUNCTIONAL TEST**, in accordance with Table 4.3-1, Item 4.

b. At least once per 18 months by performance of a **CHANNEL CALIBRATION.**

4.4.3.2 Each block valve shall be demonstrated **OPERABLE** at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed to meet the requirements of Action a, b, or c in Specification 3.4.3.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The Pressurizer shall be **OPERABLE** with a steam bubble and with at least 150 kw of pressurizer heater capacity capable of being supplied by emergency power. The pressurizer level shall be maintained within an operating band between 133 and 225 inches except when three charging pumps are operating and letdown flow is less than 25 GPM. If three charging pumps are operating and letdown flow is less than 25 GPM pressurizer level shall be limited to between 133 and 210 inches.

APPLICABILITY: **MODES** 1 and 2.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least **HOT STANDBY** within the next 6 hours and in **HOT SHUTDOWN** within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least **HOT STANDBY** with the reactor trip breakers open within 6 hours and in **HOT SHUTDOWN** within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within the above band at least once per 12 hours.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be **OPERABLE**.

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

ACTION: With one or more steam generators inoperable, restore the inoperable generator(s) to **OPERABLE** status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated **OPERABLE** by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined **OPERABLE** during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (> 20%), and

2. Tubes in those areas where experience has indicated potential problems.

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If at least 20 percent of the tubes were inspected and the results were in the C-1 Category or if at least 40 percent of the tubes were inspected and were in the C-2 Category during the previous inspection, the next inspection may be extended up to a maximum of 30 months in order to correspond with the next refueling outage if the results of the two previous inspections were not in the C-3 Category. However, if the results of either of the previous two inspections were in the C-2 Category, an engineering assessment shall be performed before operation beyond 24 months and shall provide assurance that all tubes will retain adequate structural margins against burst throughout normal operating, transient, and accident conditions until the end of the fuel cycle or 30 months, whichever occurs first. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, theinspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 30 or 40 months, as applicable.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   4. A main steam line or feedwater line break.

d. The provisions of Specification 4.0.2 do not apply for extending the frequency for performing inservice inspections as specified in Specifications 4.4.5.3.a and b.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

b. The steam generator shall be determined **OPERABLE** after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed (pursuant to Specification 6.9.1.5.b). This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operations. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days pursuant to Specification 6.9.2.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATION:

1 The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2 The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3 Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. 24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	All other S. G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. 24 hour verbal notification to NRC with written followup pursuant to specification 6.9.2.	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be **OPERABLE**:

a. A Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Sump Level Alarm System, and

c. A Containment Atmosphere Gaseous Radioactivity Monitoring System.

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

ACTION:

a. With only two of the above required Leakage Detection Systems **OPERABLE**, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when either the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

b. With only one of the above required Leakage Detection Systems **OPERABLE**, operation may continue for up to 7 days provided that:

   1. Grab samples of the containment atmosphere are obtained and analyzed at least once per 12 hours, and

   2. The Reactor Coolant System water inventory balance of Surveillance Requirement 4.4.6.2.c is performed at least once per 24 hours.

   Otherwise be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated **OPERABLE** by:

a. Containment Atmosphere Gaseous and Particulate Monitoring Systems-performance of **CHANNEL CHECK**, **CHANNEL CALIBRATION** and **CHANNEL FUNCTIONAL TEST** at the frequencies specified in Table 4.3-3, and

b. Containment Sump Level Alarm System-performance of **CHANNEL CALIBRATION** at least once per 18 months.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

Reactor Coolant System Leakage

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No **PRESSURE BOUNDARY LEAKAGE**,

b. 1 GPM **UNIDENTIFIED LEAKAGE**,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 100 gallons-per-day through any one steam generator, and

d. 10 GPM **IDENTIFIED LEAKAGE** from the Reactor Coolant System.

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

ACTION:

a. With any **PRESSURE BOUNDARY LEAKAGE**, be in at least **HOT STANDBY** within 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding **PRESSURE BOUNDARY LEAKAGE**, reduce the leakage rate to within limits 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

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Either:

1. Monitoring the containment atmosphere particulate or gaseous radioactivity at least once per 12 hours, or

2. With the gaseous and particulate monitors inoperable, conducting the containment atmosphere grab sample analysis in accordance with the **ACTION** requirements of Technical Specification 3.4.6.1.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump discharge frequency at least once per 12 hours, when the Containment Sump Level Alarm System is OPERABLE,

c. Determining Reactor Coolant System leakage at least once per 72 hours during steady state operation and at least once per 24 hours when required by ACTION 3.4.6.1.b, except when operating in the shutdown cooling mode, and

d. Monitoring the reactor vessel head closure seal Leakage Detection System at least once per 24 hours.

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least **HOT STANDBY** within 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

MODES 5 and 6

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to $\leq$ 500 psia, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to **MODE** 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

* Limit not applicable with $T_{avg} \leq 250°F$.

TABLE 4.4-3

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

* Not required with $T_{avg} \leq 250°F$.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 µCi/gram **DOSE EQUIVALENT I-131**, and

b. $\leq 100/\bar{E}$ µCi/gram.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant > 1.0 µCi/gram **DOSE EQUIVALENT I-131** but within the allowable limit (below and to the left of the line) shown on Figure 3.4.8-1, operation may continue for up to 100 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. The provisions of specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 µCi/gram **DOSE EQUIVALENT I-131** for more than 100 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4.8-1, be in at least **HOT STANDBY** with T_{avg} < 500°F within 6 hours.

c. With the specific activity of the primary coolant > $100/\bar{E}$ µCi/gram, be in at least **HOT STANDBY** with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant > 1.0 µCi/gram **DOSE EQUIVALENT I-131** or > $100/\bar{E}$ µCi/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. Whenever the specific activity of the primary coolant exceeds 1.0 µCi/gram **DOSE EQUIVALENT I-131** for in excess of 50 hours for one continuous time interval or 5 percent of the unit's total yearly operating time pursuant to **ACTION** a above, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within the next 30 days. This

* With $T_{avg} \geq$ 500°F.

LIMITING CONDITION FOR OPERATION (Continued)

report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the DOSE EQUIVALENT I-131 exceeds 1.0 µCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

\# Until the specific activity of the Primary Coolant System is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

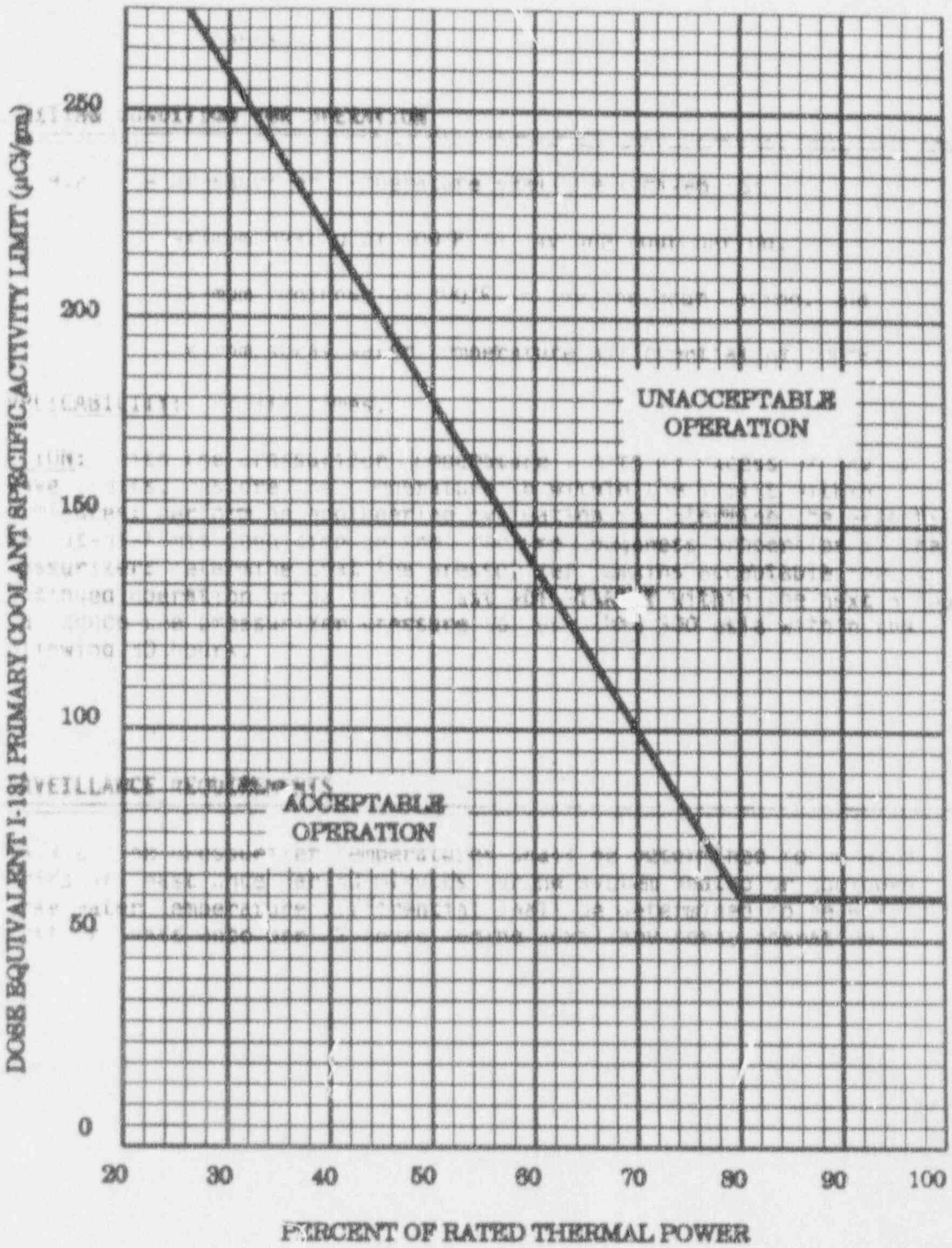

FIGURE 3.4.8-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY >1.0µCi/GRAM DOSE EQUIVALENT I-131

3/4.4 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 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.

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, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4.9-1 and 3.4.9-2.

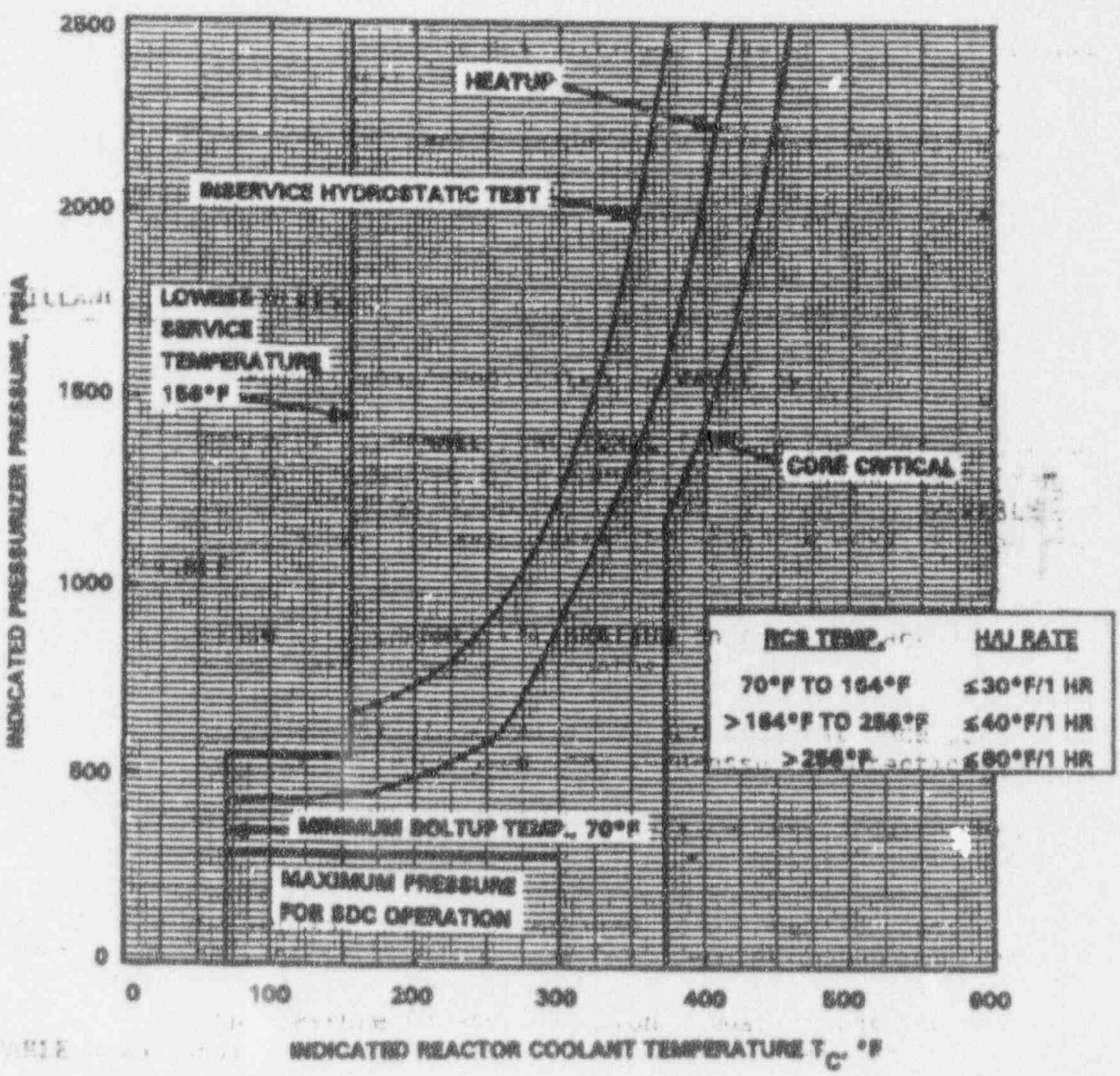

FIGURE 3.4.9-1

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

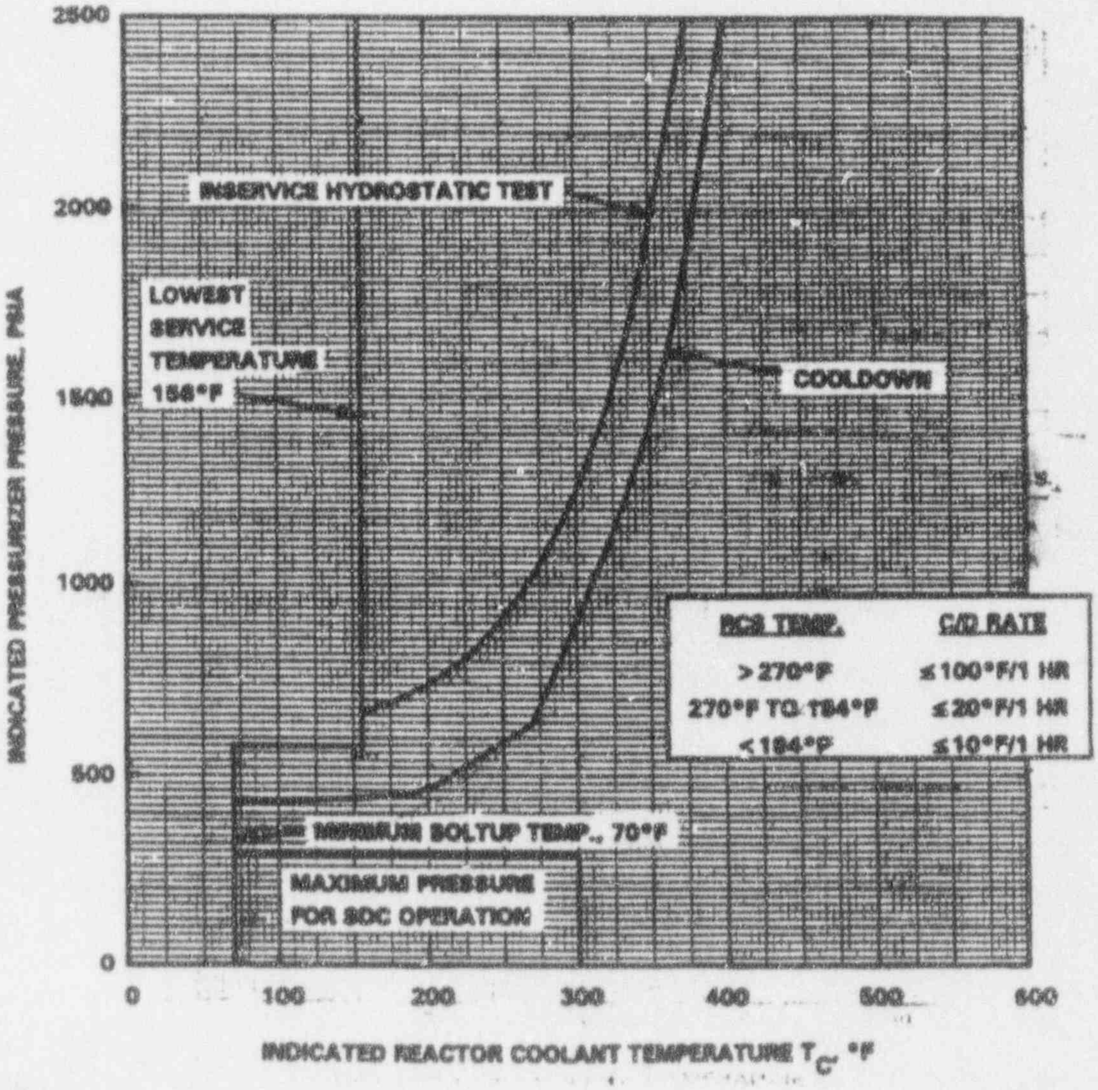

FIGURE 3.4.9-2

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

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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.

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* with their associated block valves open, or

2. A single PORV with a trip setpoint below the curve in Figure 3.4.9-3* with its associated block valve open 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.

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* 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 in **MODE** 3 with the RCS temperature $\leq$ 365°F or in **MODE** 4, either restore the inoperable PORV to **OPERABLE** status within 5 days or depressurize and vent the RCS through a $\geq$ 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 one PORV inoperable in **MODES** 5 or 6, either restore the inoperable PORV to **OPERABLE** status within 24 hours, or depressurize and vent the RCS through a $\geq$ 1.3 square inch vent(s) within the next 48 hours; and maintain the RCS in this vented condition until both PORVs have been restored to **OPERABLE** status.

c. With both PORVs inoperable, depressurize and vent the RCS through a $\geq$ 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 $\geq$ 1.3 square inches has been established or both PORVs have been restored to **OPERABLE** status.

d. 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.

e. 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.

f. 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.

g. 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.

\# EXCEPT when required for testing.

LIMITING CONDITION FOR OPERATION (Continued)

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.

3. If a pressure limit was exceeded, take action in accordance with Specification 3.4.9.1.

h. The provisions of Specification 3.0.4 are not applicable.

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 block 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 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. The above **OPERABLE** pump shall be verified to have its handswitch in pull-to-lock 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.

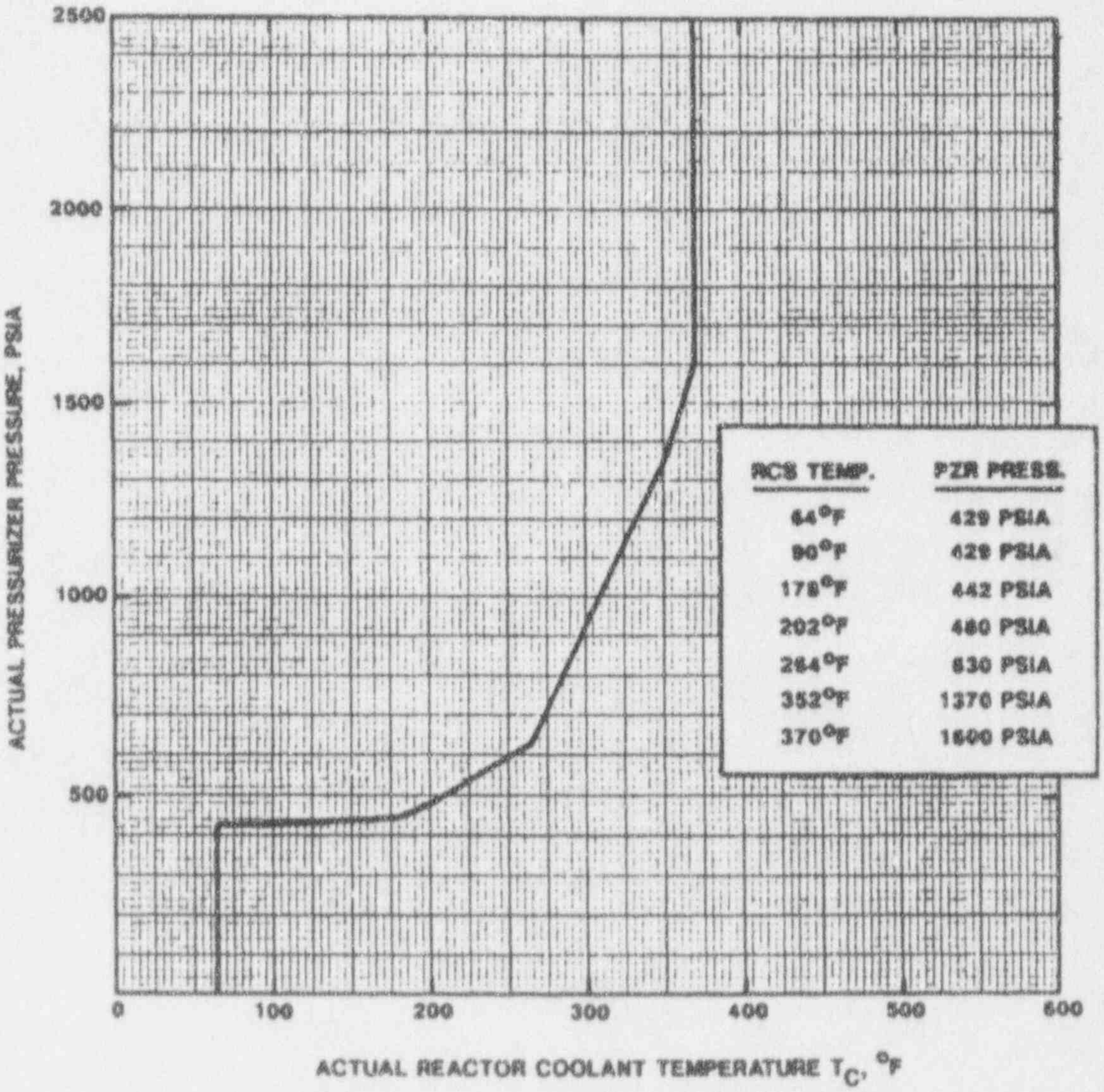

FIGURE 3.4.9-3

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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 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. However, the PORVs and their circuitry do not perform a safety-related function and, therefore, do not need emergency power as part of their operability requirements.

The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the **ACTION** requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to **OPERABLE** status.

Power is maintained to the block valve when it is closed to control excessive PORV seat leakage. This allows the PORV and block valve to remain **OPERABLE** should the PORV be needed to control reactor pressure and facilitate decay heat removal during certain accident conditions. The removal of power from a closed block valve for a PORV inoperable due to causes other than excessive PORV seat leakage provides additional assurance that the block valve will not be inadvertently opened when the condition of the PORV is uncertain.

RCS temperature, as used in the applicability statement, is determined as follows: (1) with the RCPs running, the RCS cold leg temperature (Tc) 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.

The testing for transferring motive and control power for the PORVs and block valves from the normal to emergency power bus is done under Technical Specification 4.8.1.1.2.d.3.

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 maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An engineering assessment of steam generator tube integrity will confirm that no undue risk is associated with plant operation beyond 24 months of the previous steam generator tube inspection. To provide this confirmation, the assessment would demonstrate that all tubes will retain adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the previous inspections.

2. An assessment of the structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," that can be expected before the end of the fuel cycle or 30 months, whichever comes first.

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the steam generator tube integrity assessment with actual inspection results from previous inspections.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon perminute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specifications 6.9.2 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 Leakage Detection Systems

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 Reactor Coolant System Leakage

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM **IDENTIFIED LEAKAGE** limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of **UNIDENTIFIED LEAKAGE** by the Leakage Detection Systems.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 100 gallon per day leakage limit per steam generator ensures that steam generator tube integrity is maintained in accordance with the recommendations of Generic Letter 91-04.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any **PRESSURE BOUNDARY LEAKAGE** requires the unit to be promptly placed in **COLD SHUTDOWN**.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the **SITE BOUNDARY** will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to- econdary 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 μCi/gram **DOSE EQUIVALENT I-131**, but within the allowable limit shown on Figure 3.4.8-1, accommodates possible iodine spiking phenomenon which may occur following changes in **THERMAL POWER**. Operation with specific activity levels exceeding 1.0 μCi/gram **DOSE EQUIVALENT I-131** but within the limits shown on Figure 3.4.8-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4.8-1 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 $2.61 \times 10^{19} N/cm^2$, 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 $\leq 2.61 \times 10^{19} N/cm^2$ 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.

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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 -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.

The allowed out-of-service times for degraded low temperature overpressure protection system in **MODES** 5 and 6 are based on the guidance provided in Generic Letter 90-06 and the time required to conduct a controlled, deliberate cooldown, and to depressurize and vent the RCS under the **ACTION** statement entry conditions.

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.

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.3 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3 Two power-operated relief valves (PORVs) and their associated block valves shall be **OPERABLE**.

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

ACTION:

a. If one or both PORV(s) has excessive seat leakage, within 1 hour close the associated block valve(s) and maintain power to the block valve(s).

b. With one PORV inoperable due to causes other than excessive PORV seat leakage, within 1 hour either restore the PORV to **OPERABLE** status or close the associated block valve and remove power from the block valve; restore the PORV to **OPERABLE** status within the following 5 days or be in **HOT STANDBY** within the next 12 hours and at or below 305°F within the following 24 hours.

c. With both PORVs inoperable due to causes other than excessive PORV seat leakage, within 1 hour either restore the PORVs to **OPERABLE** status or close its associated block valve and remove power from the block valve; restore one PORV to **OPERABLE** status within the following 72 hours or be in **HOT STANDBY** within the next 12 hours and at or below 305°F within the following 24 hours.

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to **OPERABLE** status or place its associated PORV(s) in override closed. Restore at least one block valve to **OPERABLE** status within the next 72 hours if both block valves are inoperable; restore any remaining inoperable block valve to **OPERABLE** status within the following 5 days; otherwise, be in at least **HOT STANDBY** within the next 12 hours and at or below 305°F within the following 24 hours.

e. The provisions of Specification 3.0.4 are not applicable.

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* Above 305°F. At or below 305°F, Specification 3/4.4.9.3 applies.

SURVEILLANCE REQUIREMENTS

4.4.3.1 Each PORV shall be demonstrated **OPERABLE**:

a. At least once per 31 days by performance of a **CHANNEL FUNCTIONAL TEST**, in accordance with Table 4.3-1, Item 4.

b. At least once per 18 months by performance of a **CHANNEL CALIBRATION**.

4.4.3.2 Each block valve shall be demonstrated **OPERABLE** at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed to meet the requirements of Action a, b, or c in Specification 3.4.3.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be **OPERABLE** with a steam bubble and with at least 150 kw of pressurizer heater capacity capable of being supplied by emergency power. The pressurizer level shall be maintained within an operating band between 133 and 225 inches except when three charging pumps are operating and letdown flow is less than 25 GPM. If three charging pumps are operating and letdown flow is less than 25 GPM pressurizer level shall be limited to between 133 and 210 inches.

APPLICABILITY: **MODES** 1 and 2.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least **HOT STANDBY** within the next 6 hours and in **HOT SHUTDOWN** within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least **HOT STANDBY** with the reactor trip breakers open within 6 hours and in **HOT SHUTDOWN** within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within the above band at least once per 12 hours.

3/4.4 **REACTOR COOLANT SYSTEM**

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be **OPERABLE**.

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

ACTION: With one or more steam generators inoperable, restore the inoperable generator(s) to **OPERABLE** status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated **OPERABLE** by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined **OPERABLE** during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

2. Tubes in those areas where experience has indicated potential problems.

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If at least 20 percent of the tubes were inspected and the results were in the C-1 Category (See Note*) or if at least 40 percent of the tubes were inspected and were in the C-2 Category during the previous inspection, the next inspection may be extended up to a maximum of 30 months in order to correspond with the next refueling outage if the results of the two previous inspections

*NOTE: For Cycle 9, an inspection of 15% of the steam generator tubes with inspection results in the C-1 Category shall be acceptable to extend the next inspection up to 30 months to coincide with the next refueling outage.

were not in the C-3 Category. However, if the results of either of the previous two inspections were in the C-2 Category, an engineering assessment shall be performed before operation beyond 24 months and shall provide assurance that all tubes will retain adequate structural margins against burst throughout normal operating, transient, and accident conditions until the end of the fuel cycle or 30 months, whichever occurs first. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 30 months or 40 months, as applicable.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

   2. A seismic occurrence greater than the Operating Basis Earthquake,

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   4. A main steam line or feedwater line break.

d. The provisions of Specificatin 4.0.2 do not apply for extending the frequency for performing inservice inspections as specified in Specifications 4.4.5.3.a and b.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

b. The steam generator shall be determined **OPERABLE** after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed (pursuant to Specification 6.9.1.5.b). This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days pursuant to Specification 6.9.2.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATION:

1 The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2 The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3 Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G.	All other S. G.s are C-1	None	N/A	N/A
		24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. 24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	N/A	N/A

$S = 3 \frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be **OPERABLE**:

a. A Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Sump Level Alarm System, and

c. A Containment Atmosphere Gaseous Radioactivity Monitoring System.

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

ACTION:

a. With only two of the above required Leakage Detection Systems **OPERABLE**, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when either the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

b. With only one of the above required Leakage Detection Systems **OPERABLE**, operation may continue for up to 7 days provided that:

1. Grab samples of the containment atmosphere are obtained and analyzed at least once per 12 hours, and

2. The Reactor Coolant System water inventory balance of Surveillance Requirement 4.4.6.2.c is performed at least once per 24 hours.

Otherwise be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated **OPERABLE** by:

a. Containment Atmosphere Gaseous and Particulate Monitoring Systems-performance of **CHANNEL CHECK, CHANNEL CALIBRATION** and **CHANNEL FUNCTIONAL TEST** at the frequencies specified in Table 4.3-3, and

b. Containment Sump Level Alarm System-performance of **CHANNEL CALIBRATION** at least once per 18 months.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

Reactor Coolant System Leakage

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No **PRESSURE BOUNDARY LEAKAGE**,

b. 1 GPM **UNIDENTIFIED LEAKAGE**,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 100 gallons-per-day through any one steam generator, and

d. 10 GPM **IDENTIFIED LEAKAGE** from the Reactor Coolant System.

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

ACTION:

a. With any **PRESSURE BOUNDARY LEAKAGE**, be in at least **HOT STANDBY** within 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding **PRESSURE BOUNDARY LEAKAGE**, reduce the leakage rate to within limits 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

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Either:

1. Monitoring the containment atmosphere particulate or gaseous radioactivity at least once per 12 hours, or

2. With the gaseous and particulate monitors inoperable, conducting the containment atmosphere grab sample analysis in accordance with the **ACTION** requirements of Technical Specification 3.4.6.1.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump discharge frequency at least once per 12 hours, when the Containment Sump Level Alarm System is OPERABLE,

c. Determining the Reactor Coolant System water leakage at least once per 72 hours during steady state operation and at least once per 24 hours when required by ACTION 3.4.6.1.b, except when operating in the shutdown cooling mode, and

d. Monitoring the reactor vessel head closure seal Leakage Detection System at least once per 24 hours.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least **HOT STANDBY** within 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

MODES 5 and 6:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to $\leq$ 500 psia, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to **MODE** 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-1

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

* Limit not applicable with $T_{avg} \leq 250°F$.

TABLE 4.4-3

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

* Not required with $T_{avg} \leq 250°F$.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 µCi/gram **DOSE EQUIVALENT I-131**, and

b. $\leq 100/\bar{E}$ µCi/gram.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant > 1.0 µCi/gram **DOSE EQUIVALENT I-131** but within the allowable limit (below and to the left of the line) shown on Figure 3.4.8-1, operation may continue for up to 100 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 µCi/gram **DOSE EQUIVALENT I-131** for more than 100 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4.8-1, be in at least **HOT STANDBY** with T_{avg} < 500°F within 6 hours.

c. With the specific activity of the primary coolant > $100/\bar{E}$ µCi/gram, be in at least **HOT STANDBY** with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant > 1.0 µCi/gram **DOSE EQUIVALENT I-131** or > $100/\bar{E}$ µCi/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. Whenever the specific activity of the primary coolant exceeds 1.0 µCi/gram **DOSE EQUIVALENT I-131** for in excess of 50 hours for one continuous time interval or 5 percent of the unit's total yearly operating time pursuant to **ACTION** (a) above, a Special Report shall be prepared and submitted to the

* With $T_{avg} \geq$ 500°F.

LIMITING CONDITION FOR OPERATION (Continued)

Commission pursuant to Specification 6.9.2 within the next 30 days. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,
2. Fuel burnup by core region,
3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,
4. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and
5. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/gram **DOSE EQUIVALENT I-131.**

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for **DOSE EQUIVALENT I-131** Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the **DOSE EQUIVALENT I-131** exceeds 1.0 μCi/gram, and	$1^{\#}$, $2^{\#}$, $3^{\#}$, $4^{\#}$, $5^{\#}$
	b) One sample between 2 and 6 hours following a **THERMAL POWER** change exceeding 15 percent of the **RATED THERMAL POWER** within a one hour period.	1, 2, 3

\# Until the specific activity of the Primary Coolant System is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of **POWER OPERATION** have elapsed since reactor was last subcritical for 48 hours or longer.

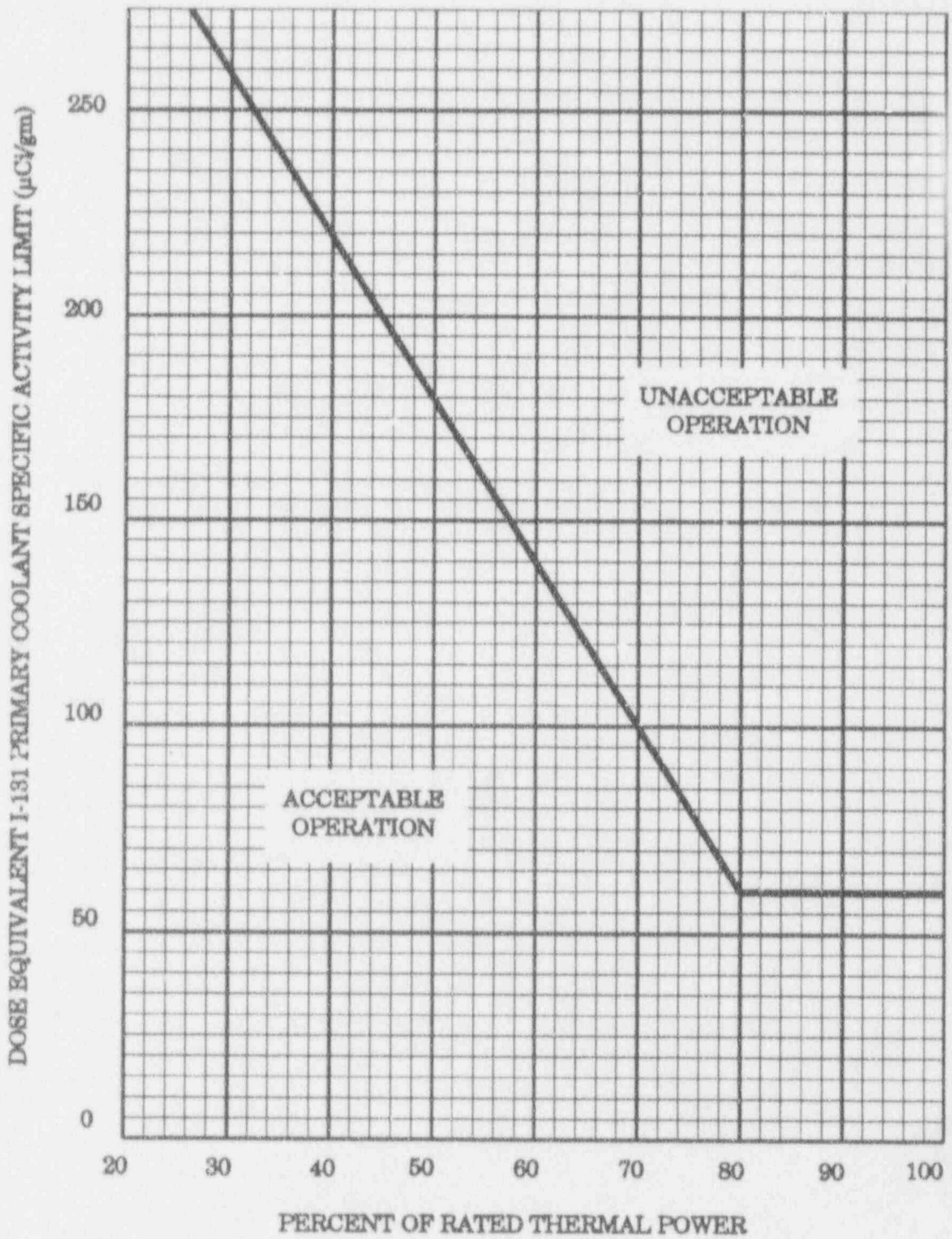

FIGURE 3.4.8-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY >1.0µCi/GRAM DOSE EQUIVALENT I-131

3/4.4 **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 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 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	> 180°F
40°F in any one hour period	180°F to 140°F
15°F in any one hour period	< 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.

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, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4.9-1 and 3.4.9-2.

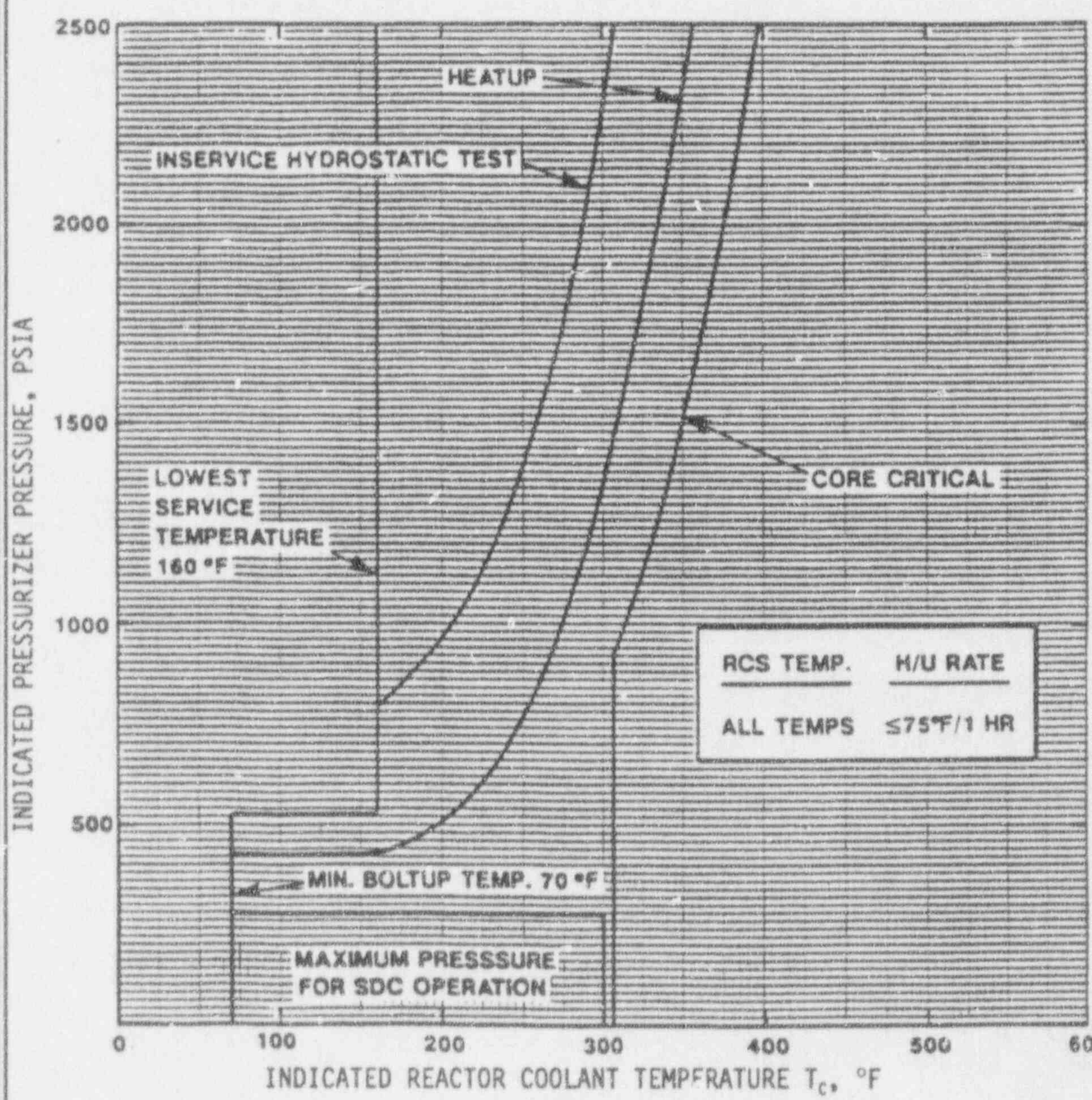

FIGURE 3.4.9-1

CALVERT CLIFFS UNIT 2 HEATUP CURVE, for FLUENCE ≤ 1.92×10^{19} n/cm^2
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS

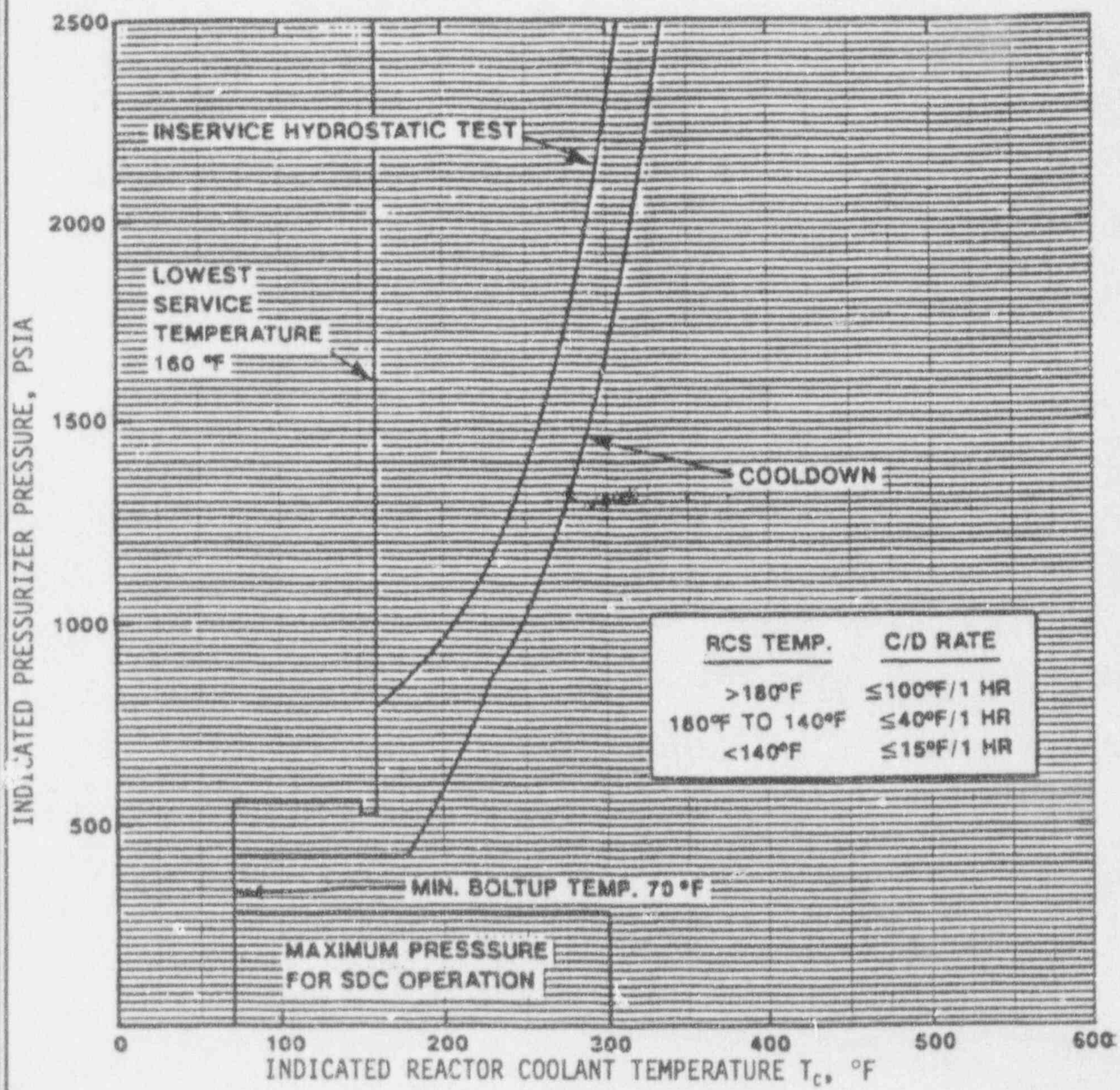

FIGURE 3.4.9-2

CALVERT CLIFFS UNIT 2 COOLDOWN CURVE, for FLUENCE ≤ 1.92×10^{19} n/cm²
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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.

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 lift setting of $\leq$ 430 psia with their associated block valves open, or

2. A single PORV with a lift setting of $\leq$ 430 psia with its associated block valve open and a Reactor Coolant System vent of $\geq$ 1.3 square inches, or

3. A Reactor Coolant System (RCS) vent $\geq$ 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 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 $\leq$ 210 gpm OR

2. A Reactor Coolant System vent of $\geq$ 2.6 square inches shall exist.

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

APPLICABILITY: When the RCS temperature is $\leq$ 305°F and the RCS is vented to < 8 square inches.

Except when required for testing.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With one PORV inoperable in **MODE** 3 with RCS temperature ≤ 305°F or in **MODE 4**, 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 one PORV inoperable in **MODES** 5 or 6, either restore the inoperable PORV to **OPERABLE** status within 24 hours, or depressurize and vent the RCS through a ≥ 1.3 square inch vent(s) within the next 48 hours; and maintain the RCS in this vented condition until both PORVs have been restored to **OPERABLE** status.

c. 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.

d. 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.

e. 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.

f. 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.

g. 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.

Except when required for testing.

LIMITING CONDITION FOR OPERATION (Continued)

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.

3. If a pressure limit was exceeded, take action in accordance with Specification 3.4.9.1.

h. The provisions of Specification 3.0.4 are not applicable.

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 block 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 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. The above **OPERABLE** pump shall be verified to have its handswitch in pull-to-lock at least once per 12 hours.

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* 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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendation 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.

* Reactor coolant pump flywheel inspections for the first inservice inspection interval may be completed during Unit 2 Refueling Outage No. 9 in conjunction with the reactor coolant pump motor overhaul program.

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.

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 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. However, the PORVs and their circuitry do not perform a safety-related function and, therefore, do not need emergency power as part of their operability requirements.

The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the **ACTION** requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to **OPERABLE** status.

Power is maintained to the block valve when it is closed to control excessive PORV seat leakage. This allows the PORV and block valve to remain **OPERABLE** should the PORV be needed to contain reactor pressure and facilitate decay heat removal during certain accident conditions. The removal of power from a closed block valve for a PORV inoperable due to causes other than excessive PORV seat leakage provides additional assurance that the block valve will not be inadvertently opened when the condition of the PORV is uncertain.

RCS temperature, as used in the applicability statement, is determined as follows: (1) with the RCPs running, the RCS cold leg temperature (Tc) 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.

The testing for transferring motive and control power for the PORVs and block valves from the normal to emergency power bus is done under Technical Specification 4.8.1.1.2.d.3.

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 maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An engineering assessment of steam generator tube integrity will confirm that no undue risk is associated with plant operation beyond 24 months of the previous steam generator tube inspection. To provide this confirmation, the assessment would demonstrate that all tubes will retain adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the previous inspections.

2. An assessment of the structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," that can be expected before the end of the fuel cycle or 30 months, whichever comes first.

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the steam generator tube integrity assessment with actual inspection results from previous inspections.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon per minute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specifications 6.9.2 prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 Leakage Detection Systems

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 Reactor Coolant System Leakage

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM **IDENTIFIED LEAKAGE** limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of **UNIDENTIFIED LEAKAGE** by the Leakage Detection Systems.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 100 gallon-per-day leakage limit per steam generator ensures that steam generator tube integrity is maintained in accordance with the recommendations of Generic Letter 91-04.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any **PRESSURE BOUNDARY LEAKAGE** requires the unit to be promptly placed in **COLD SHUTDOWN**.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associatedeffects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor

Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the **SITE BOUNDARY** will not exceed an appropriately small fraction of Part 100 limits following a 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 μCi/gram **DOSE EQUIVALENT I-131**, but within the allowable limit shown on Figure 3.4.8-1, accommodates possible iodine spiking phenomenon which may occur following changes in **THERMAL POWER**. Operation with specific activity levels exceeding 1.0 μCi/gram **DOSE EQUIVALENT I-131** but within the limits shown on Figure 3.4.8-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4.8-1 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 even more restrictive Pressure/Temperature limits must be observed. The current limits, Figures 3.4.9-1 and 3.4.9-2, are for up to and including a fluence of 1.92×10^{19} n/cm^2 at the inner surface of the reactor vessel, which corresponds to approximately 13.8 Effective Full Power Years.

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 1.92×10^{19} n/cm^2, at the 1/4 T position, the adjusted reference temperature (ART) value is less than 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 a fluence of 1.92×10^{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 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.

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°F + 10°F = 40°F. However, for conservatism, a minimum boltup temperature of 70°F is utilized in the analysis to establish the low temperature PORV lift setpoint.

BASES

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

- An 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 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.

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.

The allowed out-of-service times for degraded low temperature overpressure protection system in **MODES** 5 and 6 are based on the guidance provided in Generic Letter 90-06 and the time required to conduct a controlled, deliberate cooldown, and to depressurize and vent the RCS under the **ACTION** statement entry conditions.

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.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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 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.

3/4.4.13 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.