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MS-016

Docket Nos. 50-317
and 50-318

RECEIVED
JUN 23 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. A. E. Lundvall, Jr.
Vice President - Supply
Baltimore Gas & Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Dear Mr. Lundvall:

The Commission has issued the enclosed Amendment Nos. 55 and 38 to Facility Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 8, 1981 and corrective information provided by letters dated February 5, March 24 and April 9, 1981.

The changes to the TS incorporate the decay heat removal capacity requirements and make corrections to one or both unit's TS to make them consistent.

Your January 8, 1981 application was in response to our letter of June 11, 1980, which requested that you amend your TS with respect to decay heat removal capability. Our request was based upon a number of events that have occurred at operating pressurized water reactors where decay heat removal capability was seriously degraded due to inadequate administrative controls utilized when the plants were in shutdown modes of operation. We enclosed with our letter model TS which would provide an acceptable resolution of our concerns.

We have reviewed your submittal and consider that your proposed TS, as modified to meet our requirements and agreed to by your staff, are more conservative than the existing TS with respect to the concerns expressed in our June 11, 1980, letter. In particular, in operating Modes 3-6, your existing TS require only one system operating to remove shutdown decay heat from the core. With the implementation of the proposed TS, an additional operable system must be maintained to assure continuous decay heat removal in the event that the operating system is rendered inoperable by unforeseen circumstances. We therefore find the proposed TS acceptable.

cp

Your letters of February 5, March 24 and April 9, 1981 provided information on corrections needed to either the Unit No. 1 or Unit No. 2 TS. In addition, other corrections were discussed with your Mr. R. C. L. Olson at a meeting in Bethesda, Maryland on April 30, 1981. We find

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that all proposed corrections to the TS for both units are: (a) administrative in nature, (b) provide needed clarification, (c) have previously been approved in content (not necessarily form) for one of the units and (d) bounded by analysis of record or more restrictive than the present TS and are, therefore, acceptable. All changes have been discussed with and agreed to by your staff.

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

A copy of our related Notice of Issuance is also enclosed.

At the April 30, 1981 meeting, our representative staffs discussed outstanding review items for the Calvert Cliffs units. Based on this discussion and subsequent telephone conversations, we dnderstand that the following requests for amendments or review are no longer desirable:

- Relaxation of low end tolerance for steam line safety valves (request Item 4 of July 27, 1977 letter,

- Discussion of proposed RPS trip logic to protect against a total loss of reactor coolant flow.

Therefore, we consider the above items closed and no further review will be performed on these items by our staff.

Sincerely,

Original signed by
Robert A. Clark

~~Robert A. Clark, Chief~~
Operating Reactors Branch #3
Division of Licensing

OFFICE							
SURNAME							
DATE	Enclosures and ccs: See next page						

NRC FORM 318 (10/80) NRCM 0240 OFFICIAL RECORD COPY ☆ USGPO: 1980—329-824

Enclosures:
1. Amendment No. 55 to DPR-53
2. Amendment No. 38 to DPR-69
3. Notice

cc w/enclosures:
See next page

no legal objection to FR Notice & Amdt.

OFFICE	ORB#3:DL	ORB#3:DL	C-ORB#3:DL	AD-OR:DL	OELD		
SURNAME	PKreutzer	MConner/Cab	RAClark	TMNovak			
DATE	5/15/81	5/19/81	5/19/81	5/20/81	5/26/81		

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

June 16, 1981

Docket Nos. 50-317
and 50-318

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Vice President - Supply
Baltimore Gas & Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

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Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures and ccs: See next page

Enclosures:
1. Amendment No. 55 to DPR-53
2. Amendment No. 38 to DPR-69
3. Notice

cc w/enclosures:
See next page

Baltimore Gas and Electric Company

cc:
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General Counsel
Baltimore Gas and Electric Company
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Shaw, Pittman, Potts and Trowbridge
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Plant Superintendent
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Gaithersburg, MD 20760

Combustion Engineering, Inc.
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Windsor, CT 06095

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Calvert County Library
Prince Frederick, MD 20678

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Lusby, MD 20657

Ms. Mary Harrison, President
Calvert County Board of County Commissioners
Prince Frederick, MD 20768

U. S. Environmental Protection Agency
Region III Office
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Mr. Ralph E. Architzel
Resident Reactor Inspector
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Manager - Washington Nuclear Operations
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Calvert Cliffs Nuclear Power Plant
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Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
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Washington, D. C. 20460

Mr. W. J. Lippold, Supervisor
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Mr. R. E. Denton, General Supervisor
Training & Technical Services
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 & 4
Lusby, MD 20657

cc w/enclosure(s) and incoming
dated: 1/8/81, 2/5/81, 3/24/81, 4/9/81

Administrator, Power Plant Siting Program
Energy and Coastal Zone Administration
Department of Natural Resources
Tawes State Office Building
Annapolis, MD 21204

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
License No. DPR- 53

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated January 8, 1981, and corrective information provided by letters dated February 5, March 24 and April 9, 1981, 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. 55 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on July 1, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: June 16, 1981

ATTACHMENT TO LICENSE AMENDMENT NOS. 55 AND 38

FACILITY OPERATING LICENSE NOS. DPR-53 AND DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following pages of the Appendix A Technical Specifications for both units with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

IV
VIII (separate pages)
IX
X
XIII
2-12 (for clarity)
2-13
2-14 (remove)
2-15 "
2-16 "
2-17 "
2-18 "
2-19 "
3/4 1-3
3/4 1-16
3/4 1-23
3/4 2-8 (printing error)
3/4 2-13
3/4 4-1
3/4 4-2

Pages

3/4 4-2a (new page)
3/4 4-2b (new page)
3/4 5-1
3/4 5-2
3/4 5-7
3/4 7-26
3/4 7-45 (Unit 2 only)
3/4 7-46 (Unit 2 only)
3/4 7-51 (Unit 1 only)
3/4 7-52 (Unit 1 only)
3/4 7-53 (Unit 2 only)
3/4 7-62 (Unit 1 only)
3/4 9-8
3/4 9-8a (new page)
3/4 10-2
B 3/4 1-2
B 3/4 2-2
B 3/4 4-1
B 3/4 9-2
5-4

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

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BASES

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DESIGN FEATURES

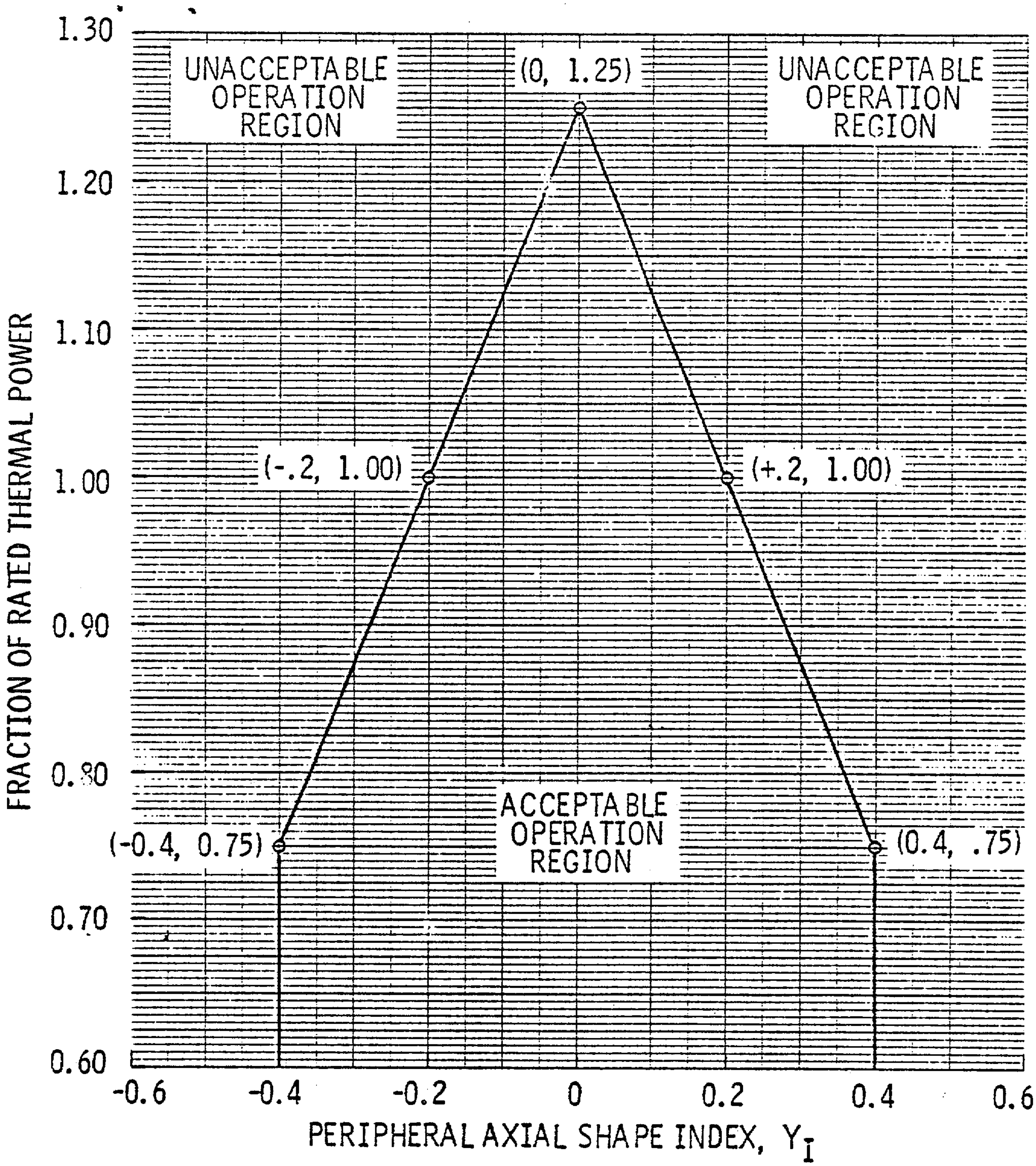

Figure 2.2-1

PERIPHERAL AXIAL SHAPE INDEX, Y_I VS FRACTION OF RATED THERMAL POWER

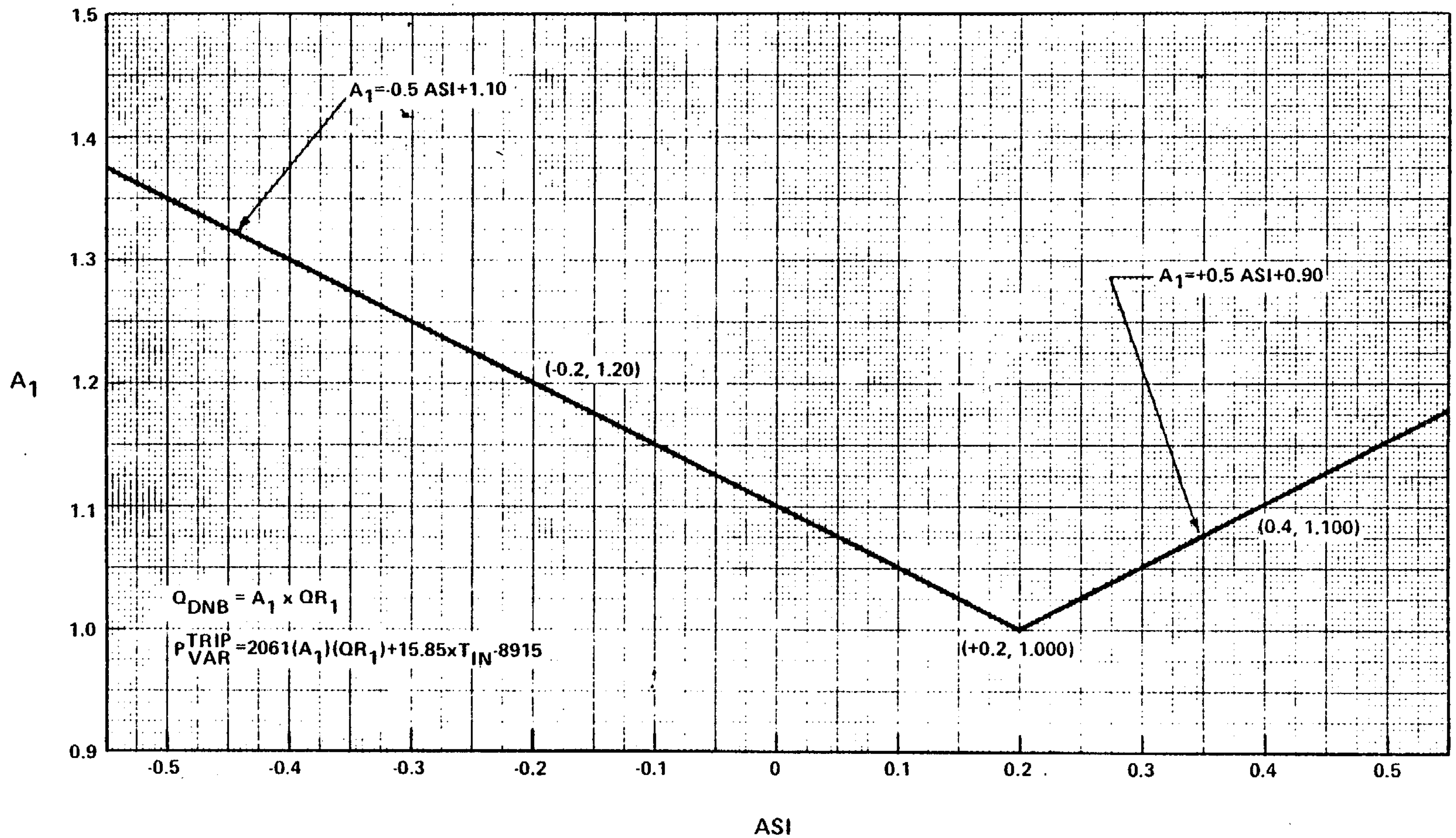

FIGURE 2.2-2

Thermal Margin/Low Pressure Trip Setpoint
Part 1 (ASI Versus A_1)

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 3.0% $\Delta k/k$.

APPLICABILITY: MODE 5

a. Pressurizer level $\geq$ 90 inches from bottom of the pressurizer.

b. Pressurizer level < 90 inches from bottom of the pressurizer and all sources of non-borated water $\leq$ 88 gpm.

ACTION:

a. With the SHUTDOWN MARGIN < 3.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 2300 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

b. With the pressurizer drained to $\leq$ 90 inches and all sources of non-borated water > 88 gpm, immediately suspend all operations involving positive reactivity changes while the SHUTDOWN MARGIN is increased to compensate for the additional sources of non-borated water or reduce the sources of non-borated water to $\leq$ 88 gpm.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.0% $\Delta k/k$:

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

b. At least once per 24 hours by consideration of the following factors:

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

4.1.1.2.2. With the pressurizer drained to $\leq$ 90 inches determine:

a. Within one hour and every 12 hours thereafter that the level in the reactor coolant system is above the bottom of the hot leg nozzles, and

b. Within one hour and every 12 hours thereafter that the sources of non-borated water are $\leq$ 88 gpm or the shutdown margin has compensated for the additional sources.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be ≥ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm through the reactor coolant system.

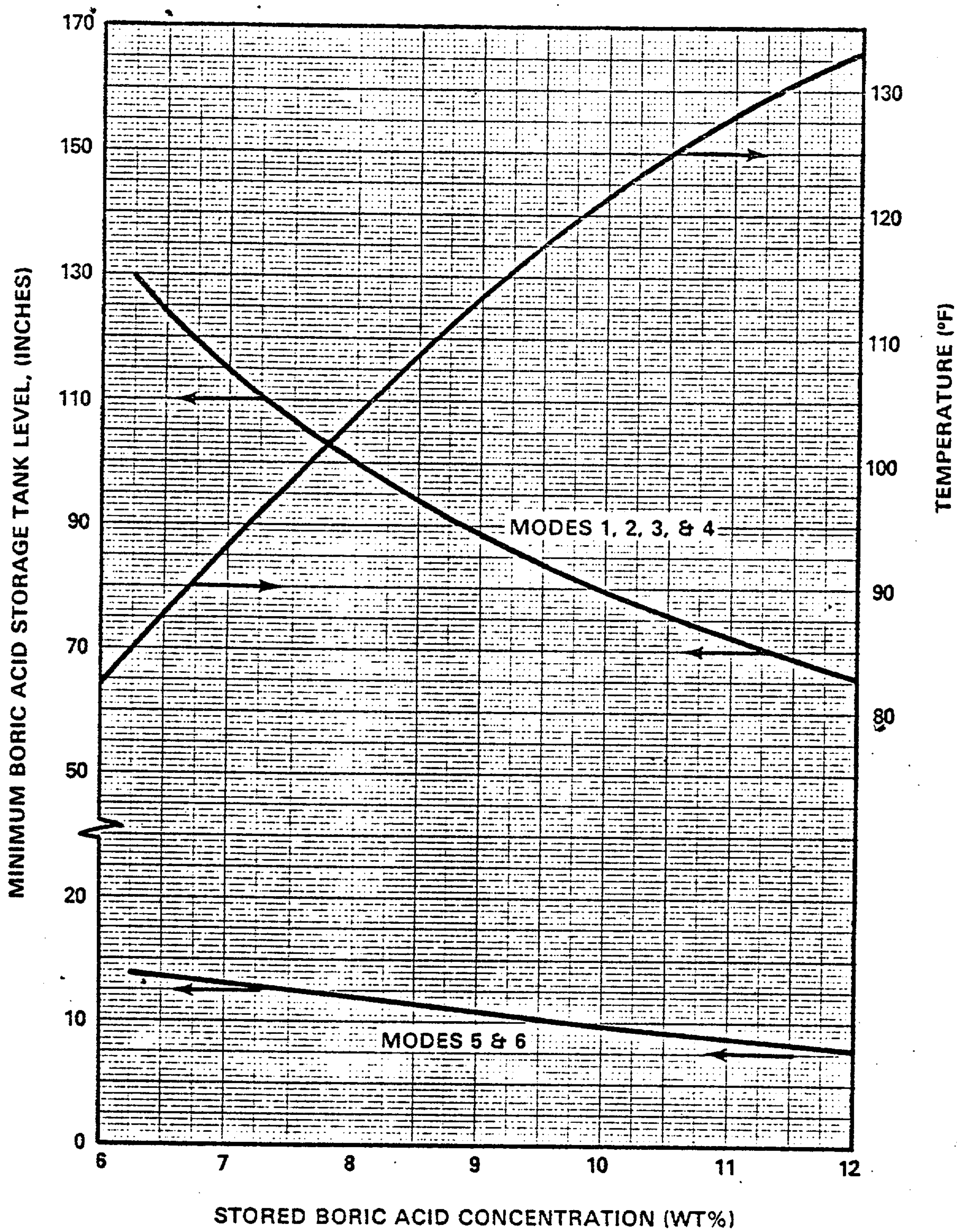

FIGURE 3.1-1

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

b. The refueling water tank with:

1. A minimum contained borated water volume of 400,000 gallons,
2. A boron concentration of between 2300 and 2700 ppm,
3. A minimum solution temperature of 40°F, and
4. A maximum solution temperature of 100°F in MODE 1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,
2. Verifying the contained borated water volume in each water source, and
3. Verifying the boric acid storage tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 40°F.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq$ 515°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, within one hour either:

a. Withdraw the CEA to at least 129.0 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

#With $K_{eff} \geq 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

4.2.2.4 T_q shall be determined each time a calculation of F^T_{xy} is required and the value of T_q used to determine F^T_{xy} shall be the measured value of T_q.

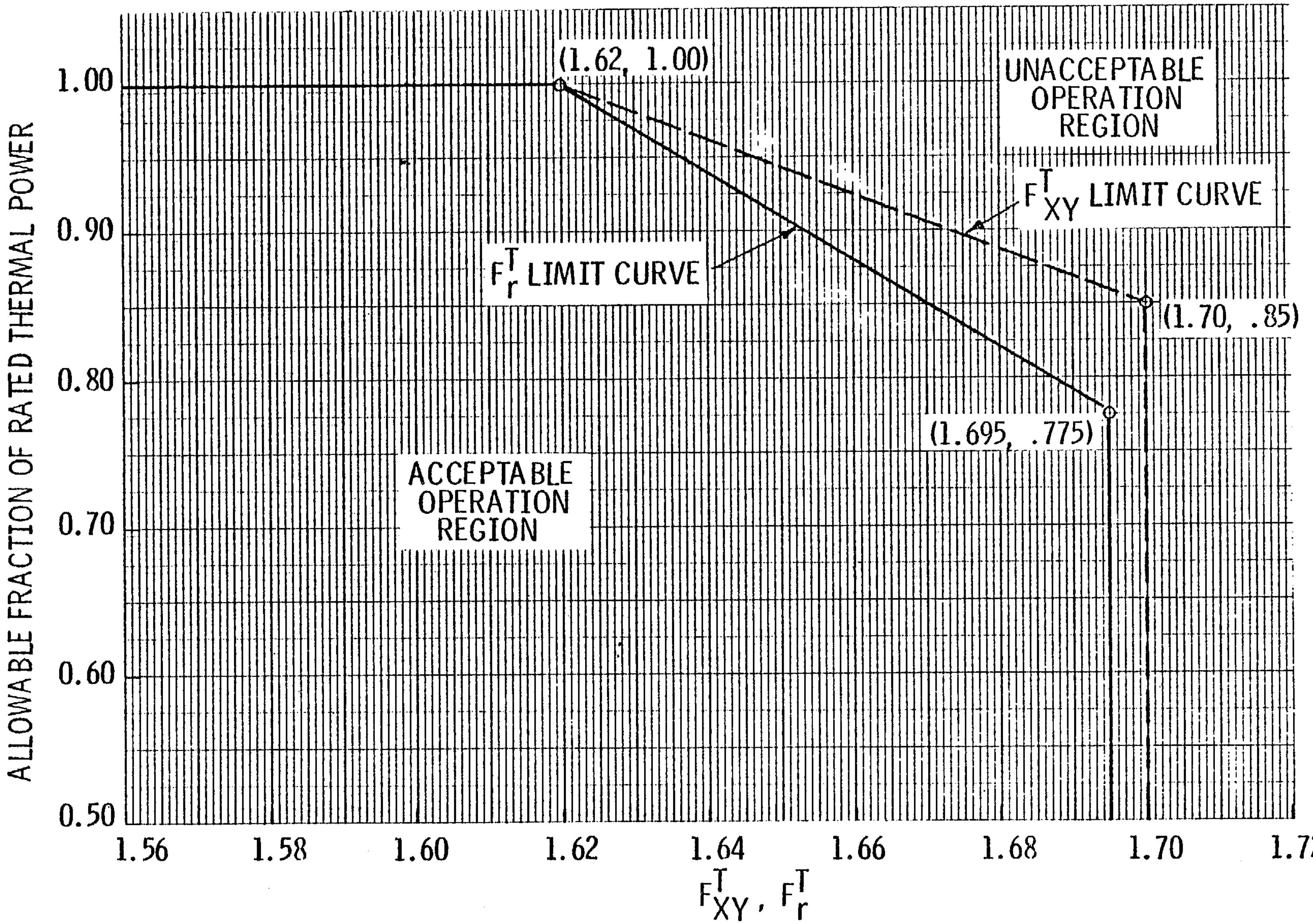

Figure 3.2-3

TOTAL RADIAL PEAKING FACTORS vs ALLOWABLE FRACTION OF RATED THERMAL POWER

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

Parameter	LIMITS			
	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	Two Reactor Coolant Pumps Operating-Same Loop	Two Reactor Coolant Pumps Operating-Opposite Loop
Cold Leg Temperature	$\leq$ 548°F	**	**	**
Pressurizer Pressure	$\geq$ 2225 psia*	**	**	**
Reactor Coolant System Total Flow Rate	$\geq$ 370,000 gpm	**	**	**
AXIAL SHAPE INDEX	Figure 3.2-4	**	**	**

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

**These values left blank pending NRC approval of ECCS analyses for operation with less than four reactor coolant pumps operating.

COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.3.

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

1. Reactor Coolant Loop #11 (#21) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop #12 (#22) and at least one associated reactor coolant pump.

b. At least one of the above Reactor Coolant Loops shall be in operation*.

APPLICABILITY: MODE 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

COOLANT LOOPS AND COOLANT CIRCULATION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

1. Reactor Coolant Loop #11 (#21) and its associated steam generator and at least one associated reactor coolant pump,
2. Reactor Coolant Loop #12 (#22) and its associated steam generator and at least one associated reactor coolant pump,
3. Shutdown Cooling Loop #11 (#21)*,
4. Shutdown Cooling Loop #12 (#22)*.

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

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

***A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to $275^{o}F$ unless (1) the pressurizer water volume is less than 600 cubic feet or (2) the secondary water temperature of each steam generator is less than $46^{o}F$ ($34^{o}F$ when measured by a surface contact instrument) above each of the RCS cold leg temperatures.

#See Special Test Exception 3.10.5.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 1113 and 1179 cubic feet of borated water (equivalent to tank levels of between 187 and 199 inches, respectively),

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

d. A nitrogen cover-pressure of between 200 and 250 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 12 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each safety injection tank isolation valve is open.

*With pressurizer pressure $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by verifying the boron concentration of the safety injection tank solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig, by verifying that power to the isolation valve operator is removed by maintaining the feeder breaker open under administrative control.

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

e. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

   1. When the RCS pressure exceeds 300 psia, and

   2. Upon receipt of a safety injection test signal.

f. Within one hour prior to each increase in solution volume of $\geq$ 1% of normal tank volume by verifying the boron concentration at the operating high pressure safety injection pump discharge is between 2300 and 2700 ppm.

ECCS SUBSYSTEMS - T_{avg} < 300°F

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 3* and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All high-pressure safety injection pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is ≤ 275°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

*With pressurizer pressure < 1750 psia.

#A maximum of one high-pressure safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is ≤ 275°F.

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

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

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

c. A minimum water temperature of 40°F, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is $< 40°F$,

3/4.7.8 HYDRAULIC SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All hydraulic snubbers listed in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more hydraulic snubbers inoperable, replace or restore the inoperable snubber(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Hydraulic snubbers shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Each hydraulic snubber with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment and approved as such by the NRC, shall be determined OPERABLE at least once after not less than 4 months but within 6 months of initial criticality and in accordance with the inspection schedule of Table 4.7-4 thereafter, by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors. Initiation of the Table 4.7-4 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.

b. Each hydraulic snubber with seal material not fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment shall be determined OPERABLE at least once per 31 days by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-4, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 lb. capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

d. Snubbers served by a common hydraulic reservoir are indicated by a bracket in Table 2.7-4. All reservoirs serving more than one snubber shall be inspected to ensure adequate hydraulic level:

1. Within 7 days after reactor startup following a major outage or following any maintenance in the immediate vicinity of these snubbers, reservoirs or associated hydraulic piping; and

2. Every 31 days $\pm$ 25 percent.

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1-61-17	CONTAINMENT SPRAY D/STRM S/D H/X -15'	A	No	No
1-63-9	S.G. #11 BLOWDOWN ORIFICE LINE 70'	I	Yes	No
1-63-10	S.G. #11 BLOWDOWN ORIFICE BYPASS 78'	I	Yes	No
1-63-11	NITROGEN to S.G. #12 74'	I	Yes	No
1-63-12	NITROGEN to S.G. #12 69'	I	Yes	No
1-63-13	STEAM GENERATORS 75'	I	Yes	No
1-63-14	STEAM GENERATORS 75'	I	Yes	No
1-63-15	STEAM GENERATORS 75'	I	Yes	No
1-63-16	STEAM GENERATORS 75'	I	Yes	No
1-63-17	STEAM GENERATORS 75'	I	Yes	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1-63-18	STEAM GENERATORS 75'	I	Yes	No
1-63-19	STEAM GENERATORS 75'	I	Yes	No
1-63-20	STEAM GENERATORS 75'	I	Yes	No
1-63-21	STEAM GENERATORS 75'	I	Yes	No
1-63-22	STEAM GENERATORS 75'	I	Yes	No
1-63-23	STEAM GENERATORS 75'	I	Yes	No
1-63-24	STEAM GENERATORS 75'	I	Yes	No
1-63-25	STEAM GENERATORS 75'	I	Yes	No
1-63-26	STEAM GENERATORS 75'	I	Yes	No
1-63-27	STEAM GENERATORS 75'	I	Yes	No
1-63-28	STEAM GENERATORS 75'	I	Yes	No
1-64-1	LINE TO PRESS. RELIEF MOV-403 81'	I	Yes	No

TABLE 4.7-4

HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 or 4	124 days ± 25%
5, 6, or 7	62 days ± 25%
≥8	31 days ± 25%

* Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

** The required inspection interval shall not be lengthened more than one step at a time.

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1600 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The weight of each load, other than a fuel assembly and CEA, shall be verified to be $\leq$ 1600 pounds prior to moving it over fuel assemblies.

REFUELING OPERATIONS

SHUTDOWN COOLING AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 6 at all reactor water levels.

ACTION:

a. With less than one shutdown cooling loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. The shutdown cooling pumps may be de-energized during the time intervals required for local leak rate testing of containment penetration number 41 pursuant to the requirements of Specification 4.6.1.2.d and/or to permit maintenance on valves located in the common shutdown cooling suction line, provided (1) no operations are permitted which could cause dilution of the reactor coolant system boron concentration, (2) all CORE ALTERATIONS are suspended, (3) all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere are maintained closed, and (4) the water level above the top of the irradiated fuel is greater than 23 feet.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 3000 gpm** at least once per 4 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

**$\geq$ 1500 gpm when the Reactor Coolant System is drained to a level below the midplane of the hot leg.

SHUTDOWN COOLING AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two (2) independent shutdown cooling loops shall be OPERABLE*#.

APPLICABILITY: Mode 6 when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, initiate corrective action to return loops to OPERABLE status within one hour.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

*Normal or emergency power source may be inoperable for each shutdown cooling loop.

#One shutdown cooling loop may be replaced by one spent fuel pool cooling loop when it is lined up to provide cooling flow to the irradiated fuel in the reactor core and the heat generation rate of the core is below the heat removal capacity of the spent fuel pool cooling loop.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

MODERATOR TEMPERATURE COEFFICIENT, CEA INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

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

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The minimum available SHUTDOWN MARGIN for no load operating conditions at beginning of life is 4.1% $\Delta k/k$ and at end of life is 4.3% $\Delta k/k$. The SHUTDOWN MARGIN is based on the safety analyses performed for a steam line rupture event initiated at no load conditions. The most restrictive steam line rupture event occurs at EOC conditions. For the steam line rupture event at beginning of cycle conditions, a minimum SHUTDOWN MARGIN of less than 4.1% $\Delta k/k$ is required to control the reactivity transient, and end of cycle conditions require 4.3% $\Delta k/k$. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} \leq 200^{o}F$, the reactivity transients resulting from any postulated accident are minimal and a 3% $\Delta k/k$ shutdown margin provides adequate protection. With the pressurizer level less than 90 inches, the sources of non-borated water are restricted to increase the time to criticality during a boron dilution event.

3/4.1.1.3 BORON DILUTION

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

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

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

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

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

3/4.1.2 BORATION SYSTEMS

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

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

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 3.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6500 gallons of 7.25% boric acid solution from the boric acid tanks or 55,627 gallons of 2300 ppm borated water from the refueling water tank. However, to be consistent with the ECCS requirements, the RWT is required to have a minimum contained volume of 400,000 gallons during MODES 1, 2, 3 and 4. The maximum boron concentration of the refueling water tank shall be limited to 2700 ppm and the maximum boron concentration of the boric acid storage tanks shall be limited to 8% to preclude the possibility of boron precipitation in the core during long term ECCS cooling.

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

3/4.2.1 LINEAR HEAT RATE

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

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.070, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02.

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

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

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

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

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.195 throughout each analyzed transient.

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

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

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

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

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

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

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq 275^{\circ}F$ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than $46^{\circ}F$ ($34^{\circ}F$ when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 7.6×10^{5} lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

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

3/4.4.3 RELIEF VALVES

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

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer with the level as programmed ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The programmed level 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

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2300 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE OPERABILITY

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of CEAs and fuel assemblies, (2) the refueling machine has sufficient load capacity to lift a CEA or fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly and CEA over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

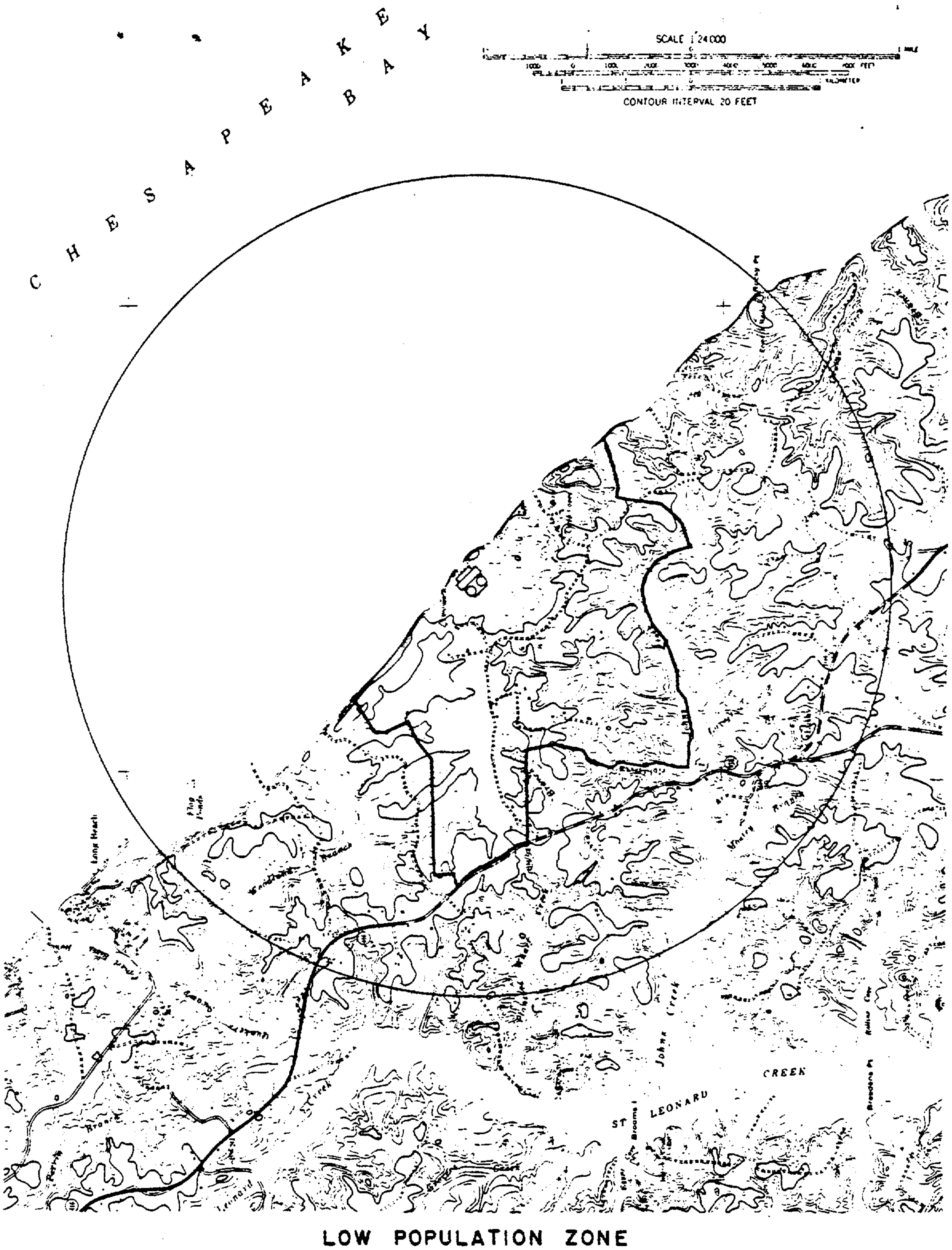
3/4.9.8 COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the core ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.



LOW POPULATION ZONE

FIG. 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 276°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 3000 grams uranium. The initial core loading shall have a maximum enrichment of 2.99 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent U-235.

5.3.2 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

CONTROL ELEMENT ASSEMBLIES

5.3.3 The reactor core shall contain 77 full length and no part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-69

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated January 8, 1981, and corrective information provided by letters dated February 5, March 24 and April 9, 1981, 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. 38 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on July 1, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: June 16, 1981

ATTACHMENT TO LICENSE AMENDMENT NOS. 55 AND 38

FACILITY OPERATING LICENSE NOS. DPR-53 AND DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following pages of the Appendix A Technical Specifications for both units with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages	Pages
IV	3/4 4-2a (new page)
VIII (separate pages)	3/4 4-2b (new page)
IX	3/4 5-1
X	3/4 5-2
XIII	3/4 5-7
2-12 (for clarity)	3/4 7-26
2-13	3/4 7-45 (Unit 2 only)
2-14 (remove)	3/4 7-46 (Unit 2 only)
2-15 "	3/4 7-51 (Unit 1 only)
2-16 "	3/4 7-52 (Unit 1 only)
2-17 "	3/4 7-53 (Unit 2 only)
2-18 "	3/4 7-62 (Unit 1 only)
2-19 "	3/4 9-8
3/4 1-3	3/4 9-8a (new page)
3/4 1-16	3/4 10-2
3/4 1-23	B 3/4 1-2
3/4 2-8 (printing error)	B 3/4 2-2
3/4 2-13	B 3/4 4-1
3/4 4-1	B 3/4 9-2
3/4 4-2	5-4

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

BASES

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DESIGN FEATURES

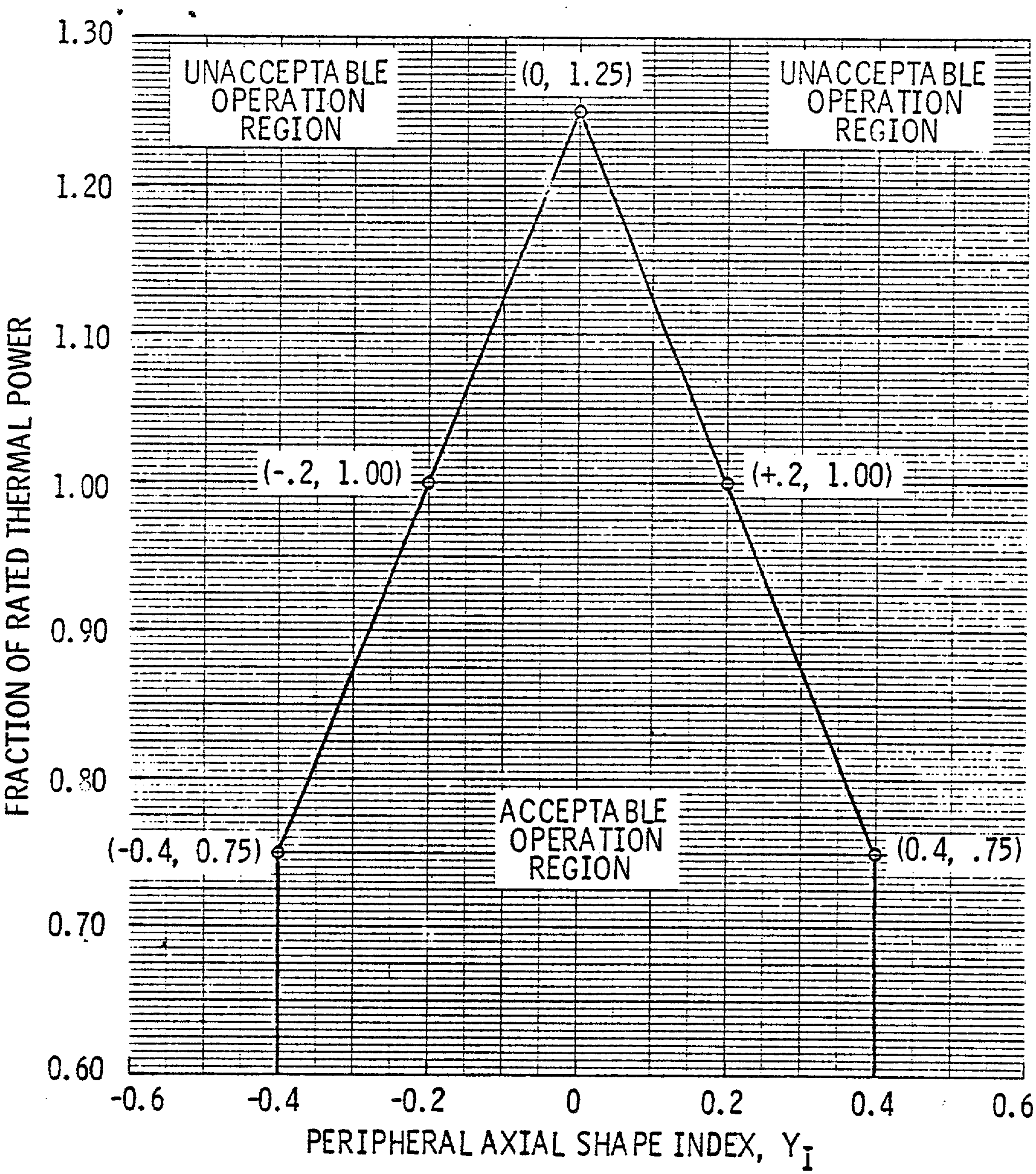

Figure 2.2-1

PERIPHERAL AXIAL SHAPE INDEX, Y_I VS FRACTION OF RATED THERMAL POWER

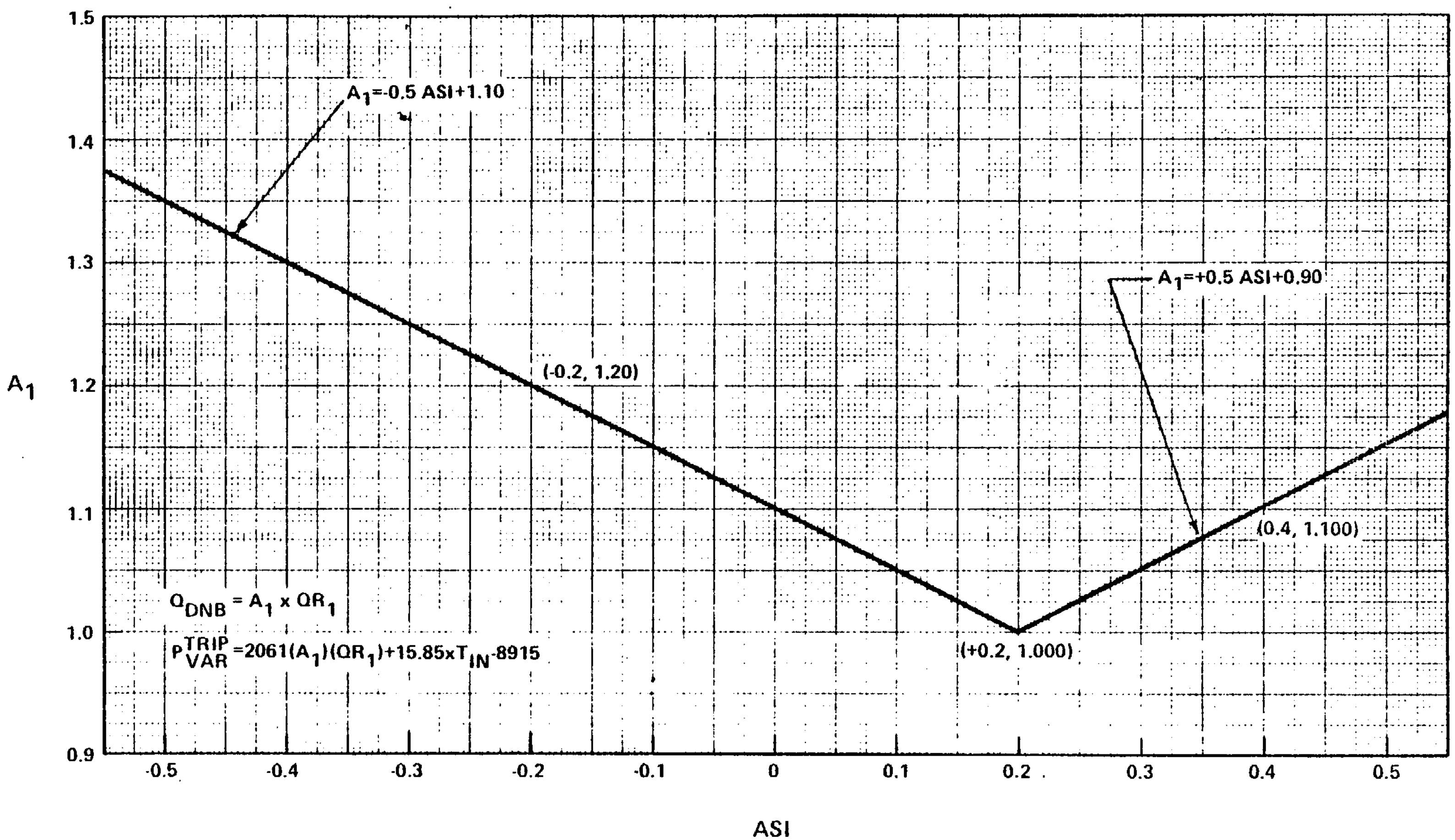

FIGURE 2.2-2

Thermal Margin/Low Pressure Trip Setpoint
Part 1 (ASI Versus A_1)

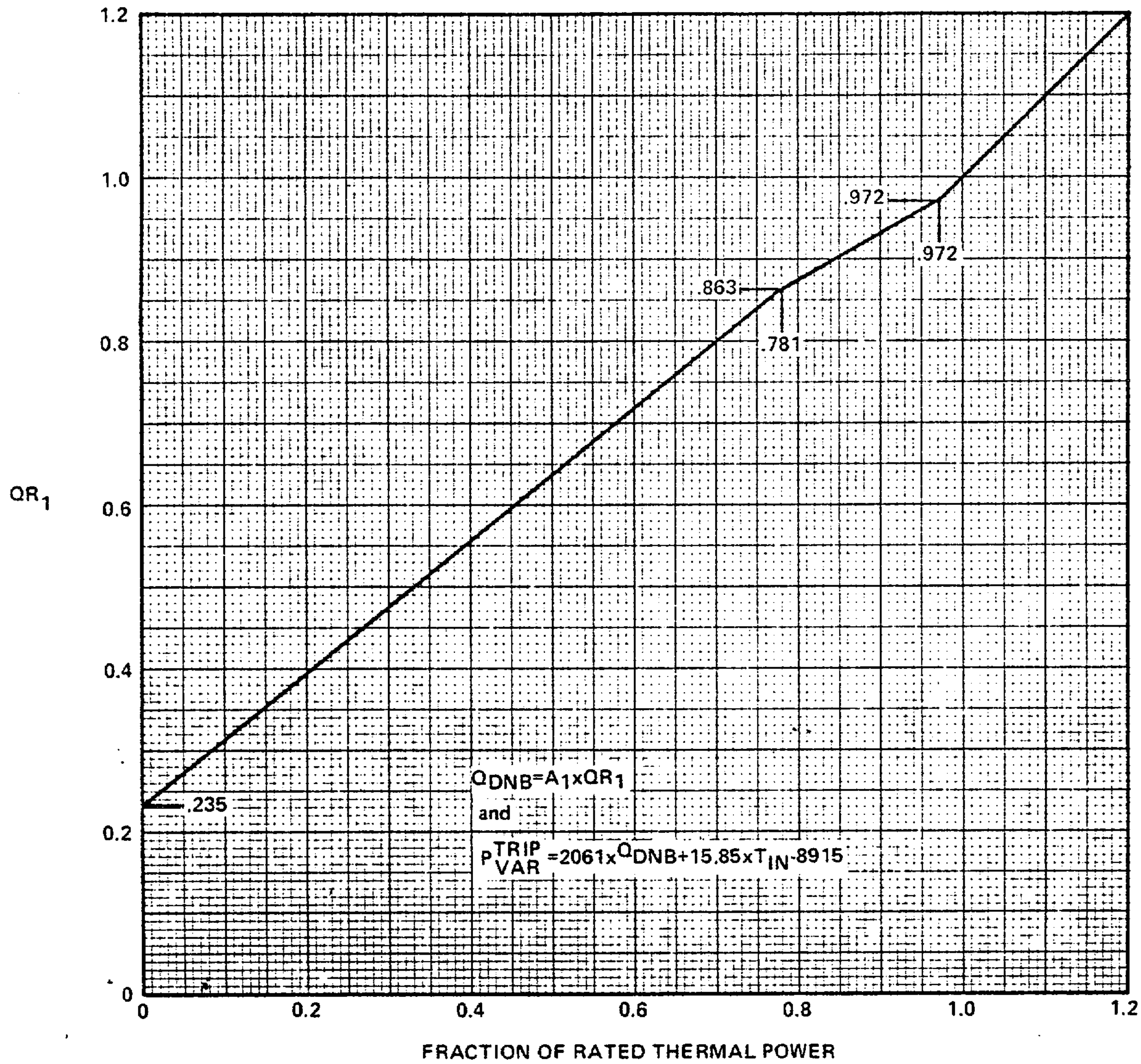

FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint
Part 2 (Fraction of RATED THERMAL POWER versus QR_1)

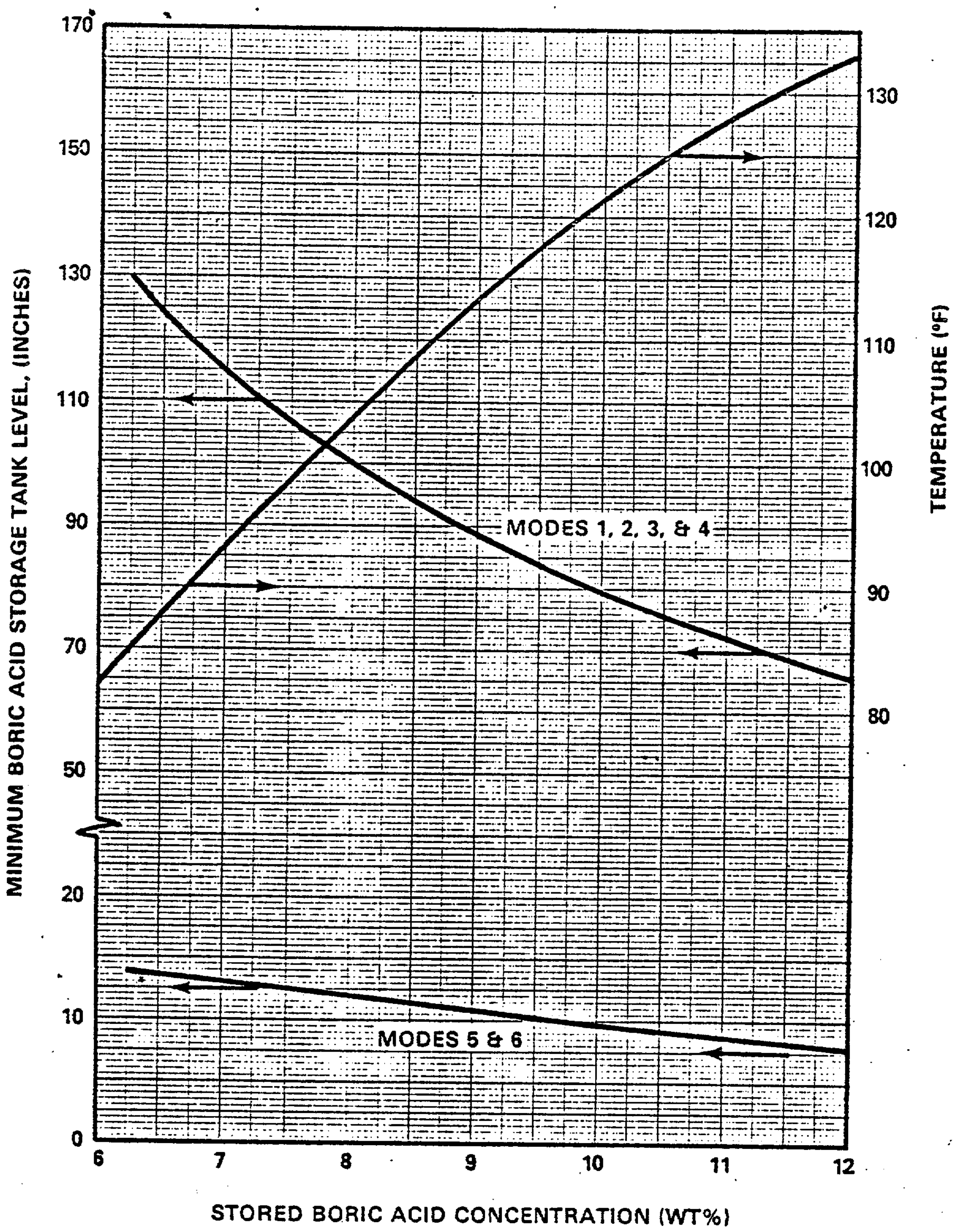

FIGURE 3.1-1

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

b. The refueling water tank with:

1. A minimum contained borated water volume of 400,000 gallons,
2. A boron concentration of between 2300 and 2700 ppm,
3. A minimum solution temperature of 40°F, and
4. A maximum solution temperature of 100°F in MODE 1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,
2. Verifying the contained borated water volume in each water source, and
3. Verifying the boric acid storage tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 40°F.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, within one hour either:

a. Withdraw the CEA to at least 129.0 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

#With $K_{eff} \geq 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

4.2.2.4 T_q shall be determined each time a calculation of F^T_{xy} is required and the value of T_q used to determine F^T_{xy} shall be the measured value of T_q.

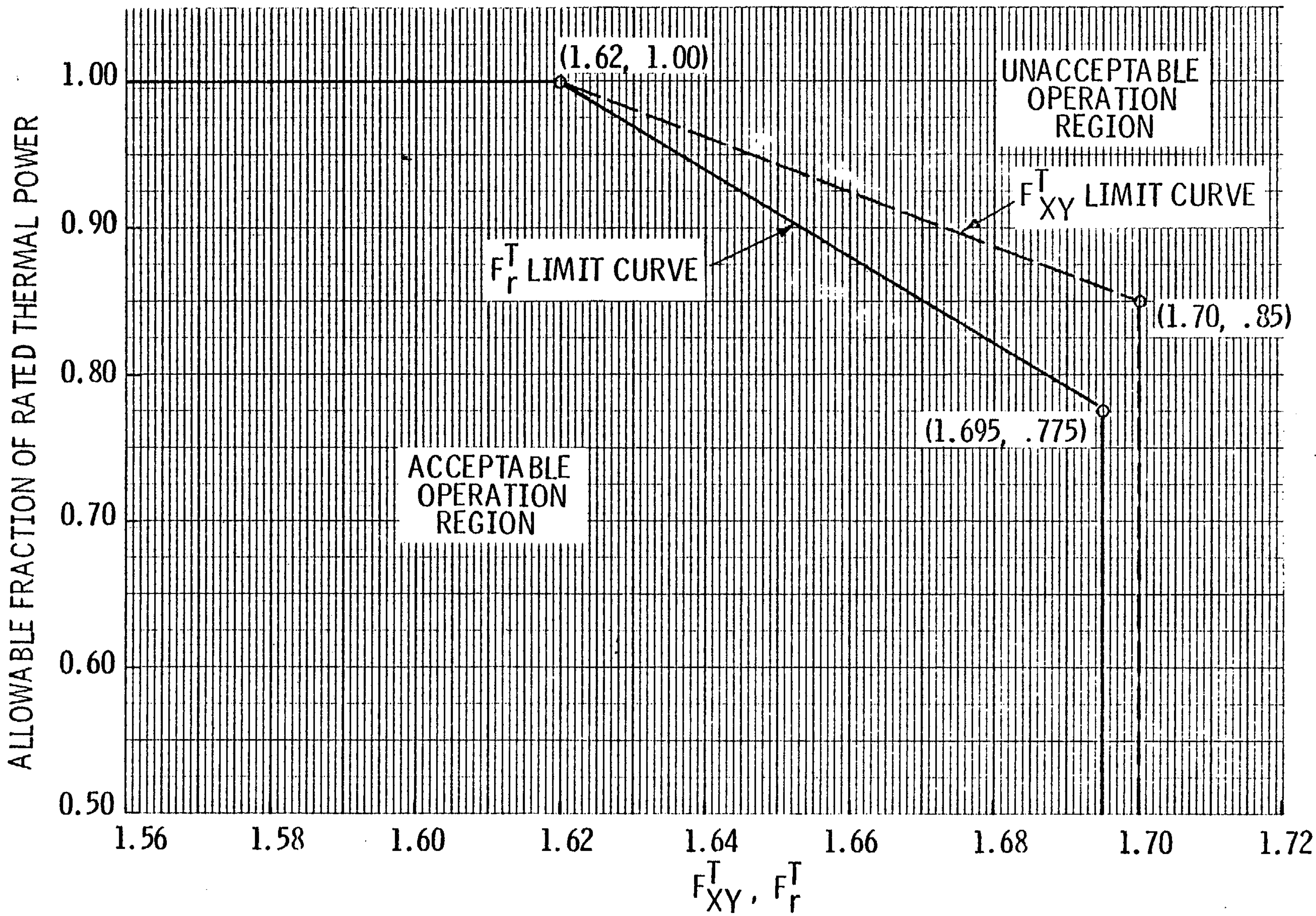

Figure 3.2-3

TOTAL RADIAL PEAKING FACTORS vs ALLOWABLE FRACTION OF RATED THERMAL POWER

COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.3.

COOLANT LOOPS AND COOLANT CIRCULATION

HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

1. Reactor Coolant Loop #11 (#21) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop #12 (#22) and at least one associated reactor coolant pump.

b. At least one of the above Reactor Coolant Loops shall be in operation*.

APPLICABILITY: MODE 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

---

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

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

1. Reactor Coolant Loop #11 (#21) and its associated steam generator and at least one associated reactor coolant pump,

2. Reactor Coolant Loop #12 (#22) and its associated steam generator and at least one associated reactor coolant pump,

3. Shutdown Cooling Loop #11 (#21)*,

4. Shutdown Cooling Loop #12 (#22)*.

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

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

---

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

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

***A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 275°F unless (1) the pressurizer water volume is less than 600 cubic feet or (2) the secondary water temperature of each steam generator is less than 46°F (34°F when measured by a surface contact instrument) above each of the RCS cold leg temperatures.

#See Special Test Exception 3.10.5.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 1113 and 1179 cubic feet of borated water (equivalent to tank levels of between 187 and 199 inches, respectively),

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

d. A nitrogen cover-pressure of between 200 and 250 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 12 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each safety injection tank isolation valve is open.

*With pressurizer pressure $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by verifying the boron concentration of the safety injection tank solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig, by verifying that power to the isolation valve operator is removed by maintaining the feeder breaker open under administrative control.

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

e. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

   1. When the RCS pressure exceeds 300 psia, and

   2. Upon receipt of a safety injection test signal.

f. Within one hour prior to each increase in solution volume of $\geq$ 1% of normal tank volume by verifying the boron concentration at the operating high pressure safety injection pump discharge is between 2300 and 2700 ppm.

ECCS SUBSYSTEMS - T_{avg} < 300°F

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 3* and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All high-pressure safety injection pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is ≤ 275°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

*With pressurizer pressure < 1750 psia.

#A maximum of one high-pressure safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is ≤ 275°F.

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

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

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

c. A minimum water temperature of 40°F, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 40°F.

3/4.7.8 HYDRAULIC SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All hydraulic snubbers listed in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more hydraulic snubbers inoperable, replace or restore the inoperable snubber(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Hydraulic snubbers shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Each hydraulic snubber with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment and approved as such by the NRC, shall be determined OPERABLE at least once after not less than 4 months but within 6 months of initial criticality and in accordance with the inspection schedule of Table 4.7-4 thereafter, by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors. Initiation of the Table 4.7-4 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.

b. Each hydraulic snubber with seal material not fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment shall be determined OPERABLE at least once per 31 days by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-4, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 lb. capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

d. Snubbers served by a common hydraulic reservoir are indicated by a bracket in Table 2.7-4. All reservoirs serving more than one snubber shall be inspected to ensure adequate hydraulic level:

   1. Within 7 days after reactor startup following a major outage or following any maintenance in the immediate vicinity of these snubbers, reservoirs or associated hydraulic piping; and

   2. Every 31 days $\pm$ 25 percent.

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-61-19	CONT. SPRAY HDR FOR SPRAY RING #22 39'	I	Yes	Yes
2-63-1	S/G #22 BLOWDOWN LINE 34' 11'	A	No	No
2-63-2	S/G #22 BLOWDOWN LINE 27' 10'	A	No	No
2-63-3	NITROGEN LINE TO S/G #22 77'6"	I	Yes	No
2-63-4	NITROGEN LINE TO S/G #22 77'6"	I	Yes	No
2-63-5	S/G #21 SURFACE BLOWDOWN LINE 76'9"	I	Yes	No
2-63-6	S/G #21 SURFACE BLOWDOWN LINE 76'9"	I	Yes	No
2-63-11	STEAM GENERATOR #21 75'	I	Yes	Yes
2-63-12	STEAM GENERATOR #21 75'	I	Yes	Yes
2-63-13	STEAM GENERATOR #21 75'	I	Yes	Yes
2-63-14	STEAM GENERATOR #21 75'	I	Yes	Yes
2-63-15	STEAM GENERATOR #21 75'	I	Yes	Yes
2-63-16	STEAM GENERATOR #21 75'	I	Yes	Yes
2-63-17	STEAM GENERATOR #21 75'	I	Yes	Yes

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-63-18	STEAM GENERATOR #21 75'	I	Yes	Yes
2-63-19	STEAM GENERATOR #22 75'	I	Yes	Yes
2-63-20	STEAM GENERATOR #22 75'	I	Yes	Yes
2-63-21	STEAM GDNERATOR #22 75'	I	Yes	Yes
2-63-22	STEAM GENERATOR #22 75'	I	Yes	Yes
2-63-23	STEAM GENERATOR #22 75'	I	Yes	Yes
2-63-24	STEAM GENERATOR #22 75'	I	Yes	Yes
2-63-25	STEAM GENERATOR #22 75'	I	Yes	Yes
2-63-26	STEAM GENERATOR #22 75'	I	Yes	Yes
2-64-1	PRESSURIZER REL PIPING UPSTREAM MOV 403 81'6"	I	Yes	No
2-64-2	PRESSURIZER REL PIPING TO RV 200 79'11"	I	Yes	No
2-64-3	PRESSURIZER REL PIPING DOWNSTREAM MOV 405 84'3"	I	Yes	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
2-83B-2	MSIV #21 HYDRAULIC SUPPLY 27'	A	No	No
2-83B-3	MSIV #21 HYDRAULIC RETURN 27'	A	No	No

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-4 provided that a revision to Table 3.7-4 is included with the next License Amendment request.

**Modifications to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License Amendment request.

TABLE 4.7-4

HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 or 4	124 days ± 25%
5, 6, or 7	62 days ± 25%
≥8	31 days ± 25%

* Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

** The required inspection interval shall not be lengthened more than one step at a time.

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1600 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The weight of each load, other than a fuel assembly and CEA, shall be verified to be $\leq$ 1600 pounds prior to moving it over fuel assemblies.

SHUTDOWN COOLING AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 6 at all reactor water levels.

ACTION:

a. With less than one shutdown cooling loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. The shutdown cooling pumps may be de-energized during the time intervals required for local leak rate testing of containment penetration number 41 pursuant to the requirements of Specification 4.6.1.2.d and/or to permit maintenance on valves located in the common shutdown cooling suction line, provided (1) no operations are permitted which could cause dilution of the reactor coolant system boron concentration, (2) all CORE ALTERATIONS are suspended, (3) all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere are maintained closed, and (4) the water level above the top of the irradiated fuel is greater than 23 feet.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 3000 gpm** at least once per 4 hours.

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*The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

**$\geq$ 1500 gpm when the Reactor Coolant System is drained to a level below the midplane of the hot leg.

SHUTDOWN COOLING AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two (2) independent shutdown cooling loops shall be OPERABLE*#.

APPLICABILITY: Mode 6 when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, initiate corrective action to return loops to OPERABLE status within one hour.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

*Normal or emergency power source may be inoperable for each shutdown cooling loop.

#One shutdown cooling loop may be replaced by one spent fuel pool cooling loop when it is lined up to provide cooling flow to the irradiated fuel in the reactor core and the heat generation rate of the core is below the heat removal capacity of the spent fuel pool cooling loop.

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

MODERATOR TEMPERATURE COEFFICIENT, CEA INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

BASES

3/4.2.1 LINEAR HEAT RATE

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

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.070, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F_{xy}^T AND F_r^T AND AZIMUTHAL POWER TILT - T_q

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

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

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

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.195 throughout each analyzed transient.

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

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

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

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

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

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

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq 275^{\circ}F$ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than $46^{\circ}F$ ($34^{\circ}F$ when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 7.6×10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

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

3/4.4.3 RELIEF VALVES

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

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer with the level as programmed ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The programmed level 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 off-site 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

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2300 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE OPERABILITY

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of CEAs and fuel assemblies, (2) the refueling machine has sufficient load capacity to lift a CEA or fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly and CEA over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the core ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-317 AND 318

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 55 and 38 to Facility Operating Licenses Nos. DPR-53 and DPR-69, issued to Baltimore Gas and Electric Company, which revised Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2. The amendments are effective on July 1, 1981.

The amendments incorporate the decay heat removal capacity requirements and make corrections to one or both unit's Technical Specifications to make them consistent.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendments.

For further details with respect to this action, see (1) the application for amendment dated January 8, 1981, as corrected February 5, March 24 and April 9, 1981, (2) Amendment Nos. 55 and 38 to License Nos. DPR-53 and DPR-69, and (3) the Commission's related letter dated June 16, 1981. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D.C. and at the Calvert County Library, Prince Frederick, Maryland. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 16th day of June, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing