ATTACHMENT (3)

UNIT 1

MARKED-UP TECHNICAL SPECIFICATION

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## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1-1.*

ADD asterisk

APPLICABILITY: **MODES** 1 and 2.

ACTION:

a. Whenever the point defined by the combination of the highest operating loop cold leg temperature and **THERMAL POWER** has exceeded the appropriate pressurizer pressure line, be in **HOT STANDBY** within 1 hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour.

c. The Vice President-Nuclear Energy and the offsite review function shall be notified within 24 hours.

d. A Safety Limit Violation Report shall be prepared and submitted to the Commission, the offsite review function and the Vice President - Nuclear Energy within 14 days of the violation.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

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

ACTION:

**MODES** 1 and 2

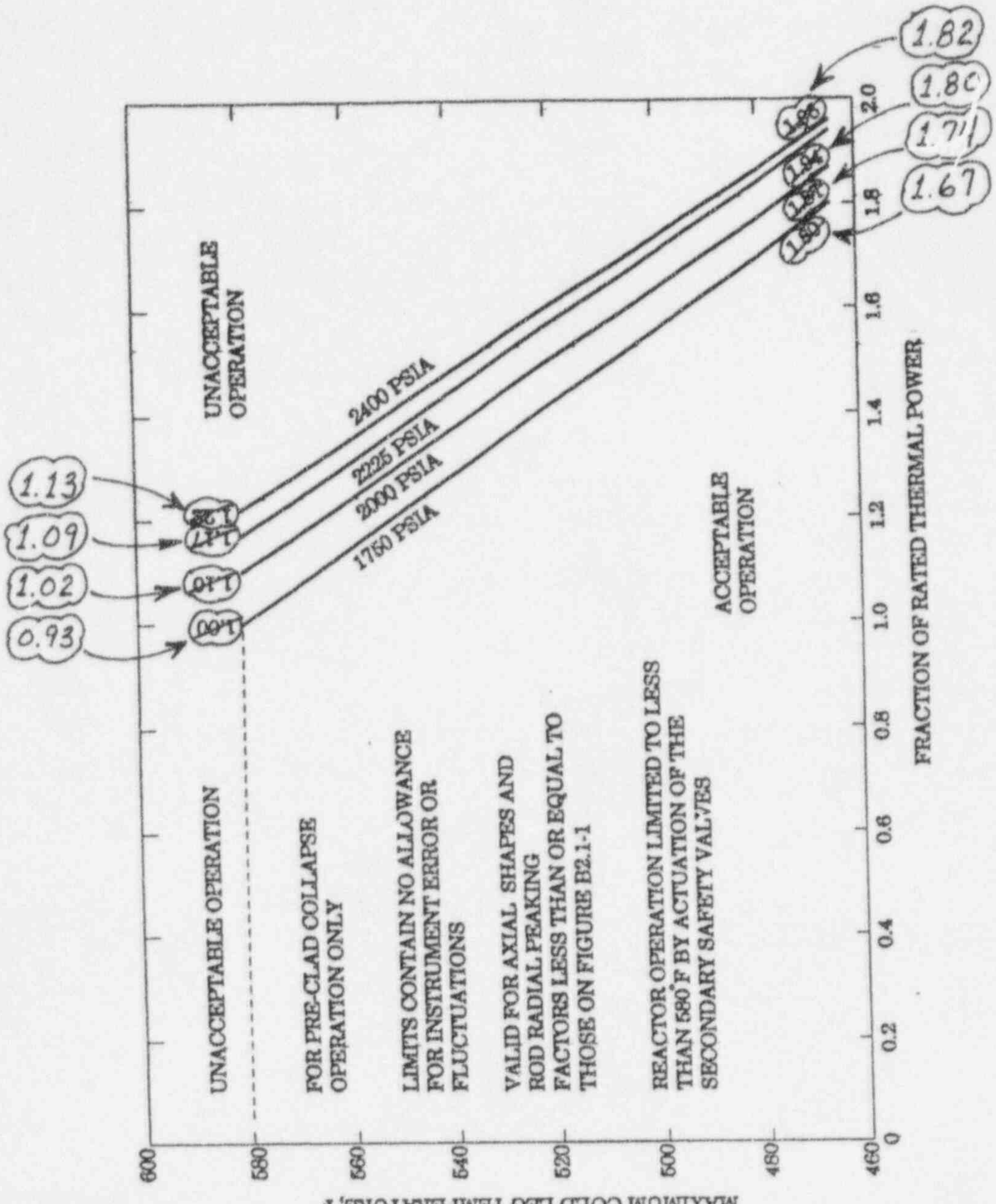
a. Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in **HOT STANDBY** with the Reactor Coolant System pressure within its limit within 1 hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour.

c. The Vice President - Nuclear Energy and the offsite review function shall be notified within 24 hours.

d. A Safety Limit Violation Report shall be prepared and submitted to the Commission, the offsite review function and the Vice President - Nuclear Energy within 14 days of the violation.

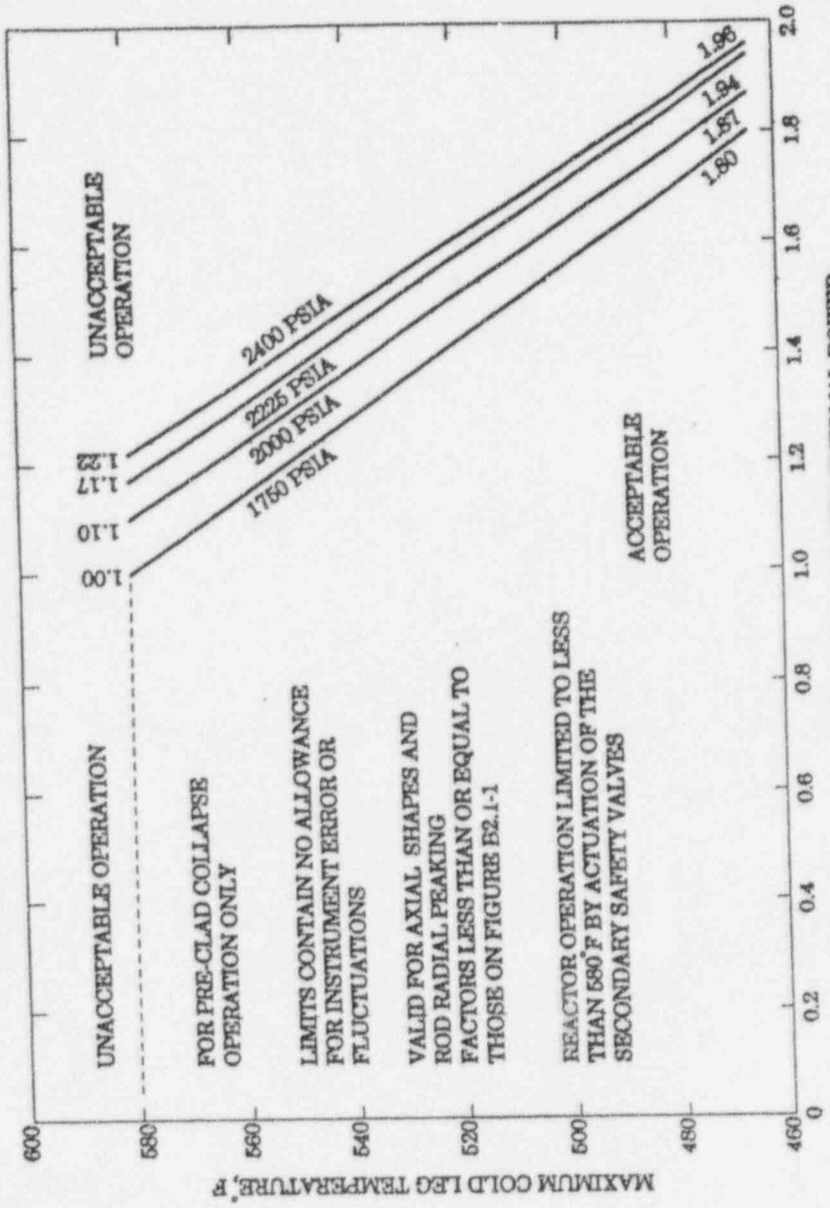
* Figure 2.1-1a shall apply through Unit 1 Cycle 13.




EFFECTIVE AFTER UNIT 1 CYCLE 13.

FIGURE 2.1-1

REACTOR CORE THERMAL MARGIN SAFETY LIMIT




EFFECTIVE THROUGH UNIT 1 CYCLE 13.

ADD

FIGURE 2.1-1a

REACTOR CORE THERMAL MARGIN SAFETY LIMIT

## TABLE 2.2-1

## REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. Manual Reactor Trip | Not Applicable | Not Applicable |
| 2. Power Level - High | ≤ 10% above **THERMAL POWER**, with a minimum setpoint of 30% of **RATED THERMAL POWER**, and a maximum of ≤ 107.0% of **RATED THERMAL POWER**. | ≤ 10% above **THERMAL POWER**, and a minimum setpoint of 30% of **RATED THERMAL POWER** and a maximum of ≤ 107.0% of **RATED THERMAL POWER**. |
| 3. Reactor Coolant Flow - Low (1) | ≥ 95%* of design reactor coolant flow | ≥ 95%* of design reactor coolant flow |
| 4. Pressurizer Pressure - High | ≤ 2400 psia | ≤ 2400 psia |
| 5. Containment Pressure - High | ≤ 4 psig | ≤ 4 psig |
| 6. Steam Generator Pressure - Low (2) | ≥ 685 psia | ≥ 685 psia |
| 7. Steam Generator Water Level - Low | ≥ 10 inches below top of feed ring | ≥ 10 inches below top of feed ring |

92%**

**TABLE 2.2-1** (Continued)

**TABLE NOTATION**

* See Specification 3.2.5, "DNB Parameters," for the design reactor coolant flow.

(1) Trip may be bypassed below $10^{-4}$% OF RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 785 psia; bypass shall be automatically removed at or above 785 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below $10^{-4}$% and above 12% of RATED THERMAL POWER.

** The Reactor Coolant Flow - Low trip setpoint and allowable value shall be ≥ 95% of design reactor coolant flow through Unit 1, Cycle 13.

3/4.2.5 DNB PARAMETERS

## LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown:

a. Cold Leg Temperature ≤ 548°F

b. Pressurizer Pressure ≥ 2200 psia*

c. Reactor Coolant System Total Flow Rate ≥ ~~370,000~~ 340,000 gpm**

d. **AXIAL SHAPE INDEX, THERMAL POWER** as specified in the COLR.

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 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.

** The Reactor Coolant System Total Flow Rate limit shall be 370,000 gpm through Unit 1 Cycle 13.

* 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**.

## TABLE 4.7-1

## STEAM LINE SAFETY VALVES PER LOOP

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

1050 psig**

** The maximum allowable lift setting for the highest set valves shall be 1065 psig through Unit 1, Cycle 13.

1050 psig (between 935 and 1065 psig through Unit 1, Cycle 13).

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

### 3/4.7.1.4 Activity

The limitations on Secondary System specific activity ensure that the resultant off-site radiation dose will be limited to ~~a small fraction of~~ less than the 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident ~~1.0 GPM~~ 100 gallons per day primary to secondary tube leak in the steam generator of the affected steam line and concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

### 3/4.7.1.5 Main Steam Line Isolation Valves

The **OPERABILITY** of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The **OPERABILITY** of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses. The main steam isolation valves are surveilled to close in less than 5.2 seconds to ensure that under reverse steam flow conditions, the valves will close in less than the 6.0 seconds assumed in the accident analysis.

### 3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 80°F and 200 psig are based on steam generator secondary side limitations and are sufficient to prevent brittle fracture.

### 3/4.7.3 COMPONENT COOLING WATER SYSTEM

The **OPERABILITY** of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

### 3/4.7.4 SERVICE WATER SYSTEM

The **OPERABILITY** of the Service Water System ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

## ATTACHMENT (4)

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## UNIT 2

## MARKED-UP TECHNICAL SPECIFICATION

## PAGES

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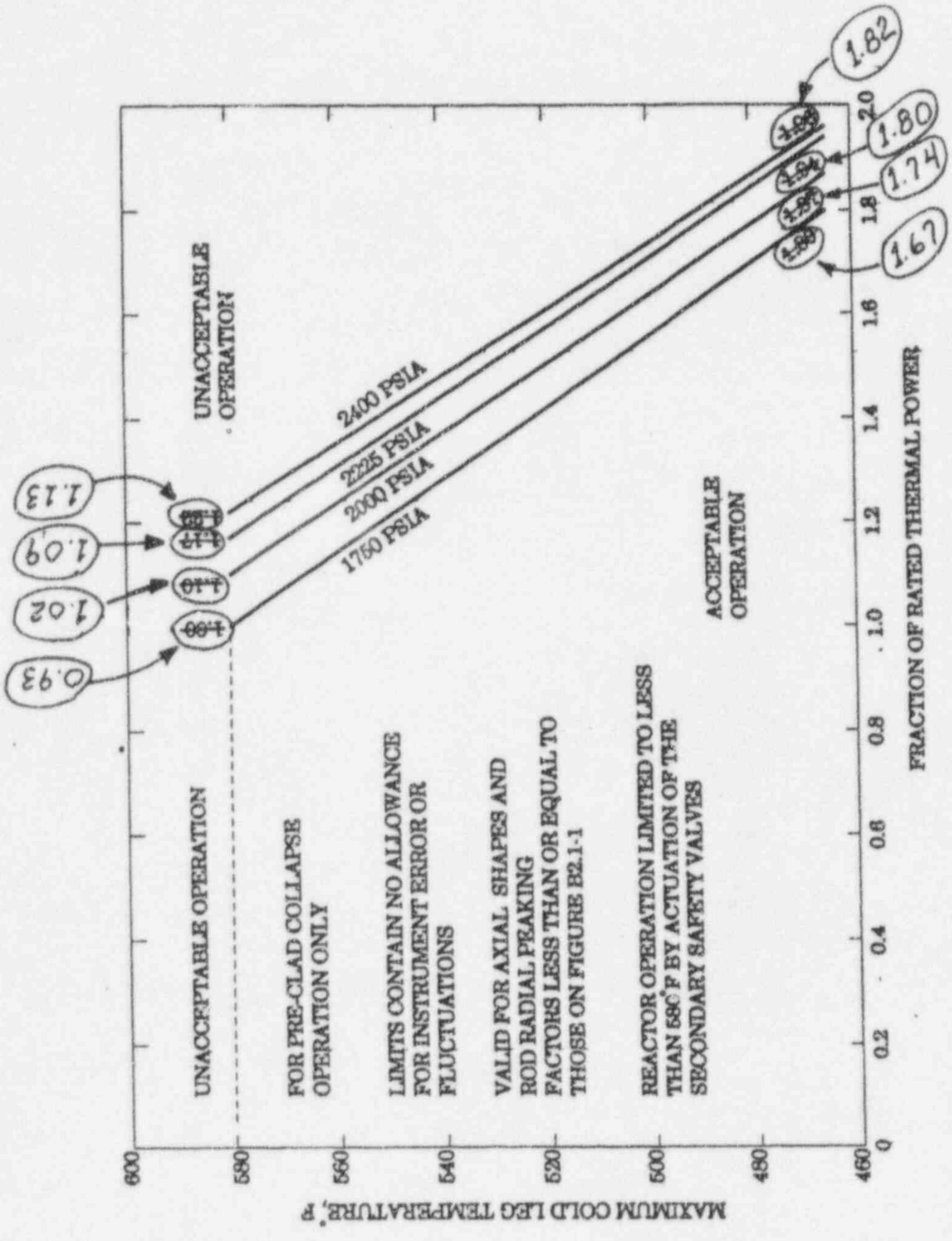


**FIGURE 2.1-1**

**REACTOR CORE THERMAL MARGIN SAFETY LIMIT**

## TABLE 2.2-1

### REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. Manual Reactor Trip | Not Applicable | Not Applicable |
| 2. Power Level - High | ≤ 10% above **THERMAL POWER**, with a minimum setpoint of 30% of **RATED THERMAL POWER**, and a maximum of ≤ 107.0% of **RATED THERMAL POWER**. | ≤ 10% above **THERMAL POWER**, and a minimum setpoint of 30% of **RATED THERMAL POWER** and a maximum of ≤ 107.0% of **RATED THERMAL POWER**. |
| 3. Reactor Coolant Flow - Low (1) | ≥ 95% of design reactor coolant flow | ≥ 95% of design reactor coolant flow |
| 4. Pressurizer Pressure - High | ≤ 2400 psia | ≤ 2400 psia |
| 5. Containment Pressure - High | ≤ 4 ps[illegible] | ≤ 4 psig |
| 6. Steam Generator Pressure - Low (2) | ≥ 685 [illegible]ia | ≥ 685 psia |
| 7. Steam Generator Water Level - Low | ≥ 10 inches below top of feed ring | ≥ 10 inches below top of feed ring |

92%

3/4.2 **POWER DISTRIBUTION LIMITS**

3/4.2.5 DNB PARAMETERS

## LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown:

a. Cold Leg Temperature ≤ 548°F

b. Pressurizer Pressure ≥ 2200 psia*

340,000

c. Reactor Coolant System Total Flow Rate ≥ ~~370,000~~ gpm

d. **AXIAL SHAPE INDEX, THERMAL POWER** as specified in the COLR

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 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.

* Limit not applicable during either **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**.

## TABLE 4.7-1

## STEAM LINE SAFETY VALVES PER LOOP

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

1050

1050

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

### 3/4.7.1.3 Condensate Storage Tank

The **OPERABILITY** of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at **HOT STANDBY** conditions for 6 hours with steam discharge to atmosphere with concurrent and total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

### 3/4.7.1.4 Activity

less than the

The limitations on Secondary System specific activity ensure that the resultant off-site radiation dose will be limited to ~~a small fraction of~~ 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident ~~1.0 GPM~~ primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

100 gallons per day

### 3/4.7.1.5 Main Steam Line Isolation Valves

The **OPERABILITY** of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The **OPERABILITY** of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses. The main steam isolation valves are surveilled to close in less than 5.2 seconds to ensure that under reverse steam flow conditions, the valves will close in less than the 6.0 seconds assumed in the accident analysis.

### 3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 90°F and 200 psig are based on steam generator secondary side limitations and are sufficient to prevent brittle fracture.

# ATTACHMENT (5)

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**MARKED-UP**

**IMPROVED TECHNICAL SPECIFICATION**

**PAGES**

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**2.0-1**

**2.0-2**

**2.0-2a**

**3.3.1-9**

**3.3.1-11**

**3.4.1-1**

**3.4.1-2**

**B 3.7.14-1**

**B 3.7.14-2**

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# 2.0 SAFETY LIMITS (SLs)

## 2.1 SLs

### 2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1.1-1.

2.1.1.2 In MODES 1 and 2, the peak linear heat rate (LHR) shall be ≤ 21.0 kw/ft.

### 2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psia.

## 2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

NOTE

*For Unit 1 only, Figure 2.1.1-1a shall apply through Cycle 13.*

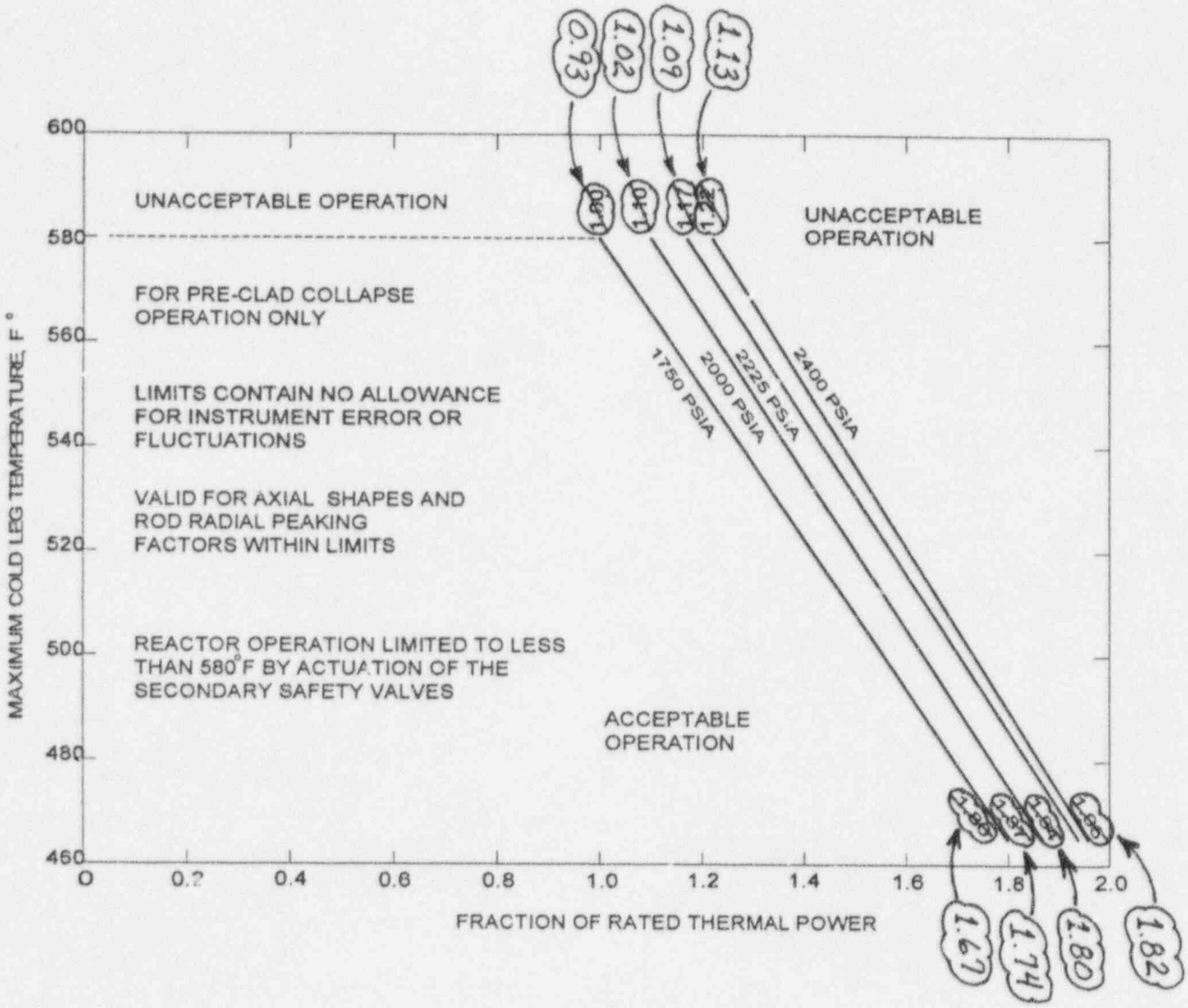
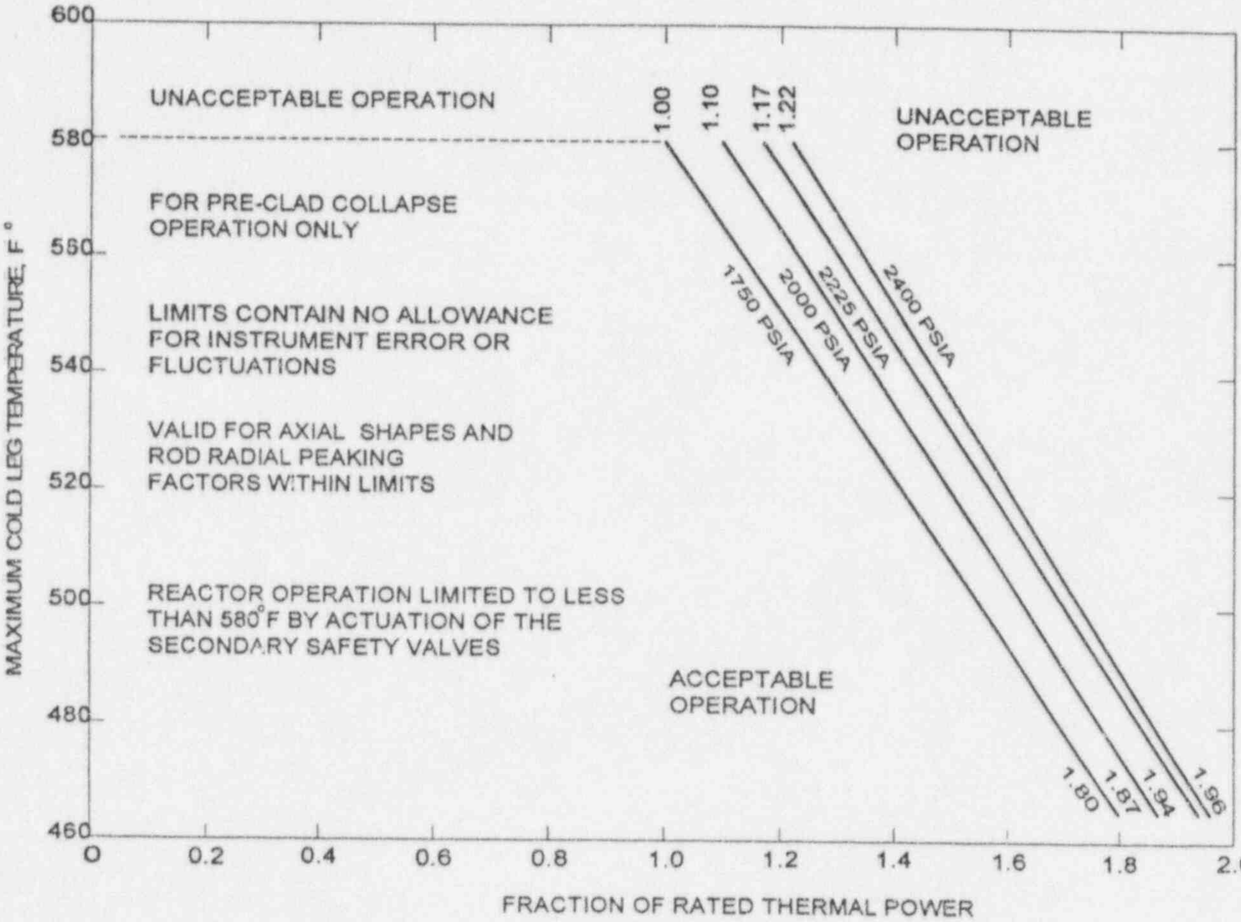



**Figure 2.1.1-1**
**Reactor Core Thermal Margin Safety Limit**




NOTE

This Figure only applies to Unit 1 through Cycle 13.

Figure 2.1.1-1 a
Reactor Core Thermal Margin Safety Limit

**Table 3.3.1-1 (page 1 of 3)**
**Reactor Protective System Instrumentation**

| FUNCTION | MODES | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|
| 1. Power Level-High | 1, 2 | SR 3.3.1.2<br>SR 3.3.1.3<br>SR 3.3.1.4<br>SR 3.3.1.5<br>SR 3.3.1.8<br>SR 3.3.1.9 | ≤ 10% RTP above current THERMAL POWER but not < 30% RTP nor > 107% RTP |
| 2. Rate of Change of Power-High[a] | 1, 2 | SR 3.3.1.1[f]<br>SR 3.3.1.6<br>SR 3.3.1.7<br>SR 3.3.1.8 | ≤ 2.6 dpm |
| 3. Reactor Coolant Flow-Low[b] | 1, 2 | SR 3.3.1.1<br>SR 3.3.1.4<br>SR 3.3.1.7<br>SR 3.3.1.8<br>SR 3.3.1.9 | ≥ ~~95%~~ 92% of Design Flow (g) ADD |
| 4. Pressurizer Pressure-High | 1, 2 | SR 3.3.1.1<br>SR 3.3.1.4<br>SR 3.3.1.8<br>SR 3.3.1.9 | ≤ 2400 psia |
| 5. Containment Pressure-High | 1, 2 | SR 3.3.1.1<br>SR 3.3.1.4<br>SR 3.3.1.8<br>SR 3.3.1.9 | ≤ 4.0 psig |
| 6. Steam Generator Pressure-Low[c] | 1, 2 | SR 3.3.1.1<br>SR 3.3.1.4<br>SR 3.3.1.7<br>SR 3.3.1.8<br>SR 3.3.1.9 | ≥ 685 psia |

**Table 3.3.1-1 (page 3 of 3)**
**Reactor Protective System Instrumentation**

(a) Bistable trip unit may be bypassed when THERMAL POWER is < 1E-4% RTP or > 12% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ 1E-4% RTP and ≤ 12% RTP.

(b) Bistable trip unit may be bypassed when THERMAL POWER is < 1E-4%. Bypass shall be automatically removed when THERMAL POWER is ≥ 1E-4% RTP. During testing pursuant to LCO 3.4.16, trips may be bypassed below 5% RTP.

(c) Bistable trip unit may be bypassed when steam generator pressure is < 785 psig. Bypass shall be automatically removed when steam generator pressure is ≥ 785 psig.

(d) Bistable trip unit may be bypassed when THERMAL POWER is < 15% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ 15% RTP.

(e) Trip is only applicable in MODE 1 ≥ 15% RTP.

(f) CHANNEL CHECK only applies to Wide Range Logarithmic Neutron Flux Monitor.

(g) The Reactor Coolant Flow - Low allowable value shall be ≥ 95 % for Unit 1 only, through Cycle 13.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2200 psia;

b. RCS cold leg temperature ($T_c$) ≤ 548°F; and

c. RCS total flow rate ≥ 370,000 gpm. 340,000

APPLICABILITY: MODE 1.

---------------------------- NOTE ----------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

--------------------------------------------------------------

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Pressurizer pressure or RCS flow rate not within limits. | A.1 | Restore parameter(s) to within limit. | 2 hours |

---------------------------- NOTE ----------------------------
The RCS total flow rate limit shall be ≥ 370,000 for Unit 1 only, through Cycle 13.

--------------------------------------------------------------

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Be in MODE 2. | 6 hours |
| C. RCS cold leg temperature not within limits. | C.1 Restore cold leg temperature to within limits. | 2 hours |
| D. Required Action and associated Completion Time of Condition C not met. | D.1 Reduce THERMAL POWER to ≤ 30% RTP. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.1.1 | Verify pressurizer pressure ≥ 2200 psia. | 12 hours |
| SR 3.4.1.2 | Verify RCS cold leg temperature ≤ 548°F. | 12 hours |
| SR 3.4.1.3 | -------------------NOTE-------------------<br>1. Only required to be met in MODE 1.<br>------------------------------------------<br>Verify RCS total flow rate ≥ ~~370,000~~ 340,000 gpm. | 12 hours |
| SR 3.4.1.4 | Verify measured RCS total flow rate is within limits. | 24 months |

2. For Unit 1 only, the RCS total flow rate shall be ≥ 370,000 through Cycle 13.

# B 3.7 PLANT SYSTEMS

## B 3.7.14 Secondary Specific Activity

BASES

**BACKGROUND**

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 100 gallons per day ~~1 gpm~~ tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.15, "RCS Specific Activity"). The main steam line break (MSLB) is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

**APPLICABLE SAFETY ANALYSES**

The accident analysis of the MSLB, as discussed in the Updated Final Safety Analysis Report (UFSAR), Chapter 14 (Ref. 2), assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that

BASES

the radiological consequences of an MSLB do not exceed ~~a~~ ~~small fraction of~~ the unit exclusion area boundary limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through main steam safety valves (MSSVs) and atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Shutdown Cooling System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through MSSVs and ADVs during the event.

Secondary specific activity limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

---

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of $\leq$ 0.10 $\mu$Ci/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

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