ENCLOSURE 2

VOLUME 4

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0
SAFETY LIMITS

Revision 0

LIST OF ATTACHMENTS

1. ITS Chapter 2.0, Safety Limits

ATTACHMENT 1

ITS Chapter 2.0, Safety Limits

Current Technical Specifications (CTS) Markup
and Discussion of Changes (DOCs)

~~SECTION 2.0~~

A01

~~SAFETY LIMITS~~

~~AND~~

~~LIMITING SAFETY SYSTEM SETTINGS~~

A01

(SLs)

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~

SLs

2.1 **2.1 ~~SAFETY LIMITS~~**

2.1.1 SLs

2.1.1 **REACTOR CORE**

the highest operating loop

2.1.1.1 2.1.1 .1 The combination of THERMAL POWER, pressurizer pressure, and ~~maxi-mum~~ cold leg coolant temperature shall not exceed the limits shown ~~on~~ in Figure 2.1-1.

.1

2.1.1.1 ~~APPLICABILITY:~~ In MODES 1 and 2~~.~~ ,

2.1.1.2 ~~ACTION:~~ 2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained within fuel melt temperature limits specified in the COLR.

M01

~~When the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line,~~

2.2.1 be in ~~HOT STANDBY~~ MODE 3 within 1 hour.

SL

2.1.2 **REACTOR COOLANT SYSTEM PRESSURE**

be maintained ≤

2.1.2 The ~~Reactor Coolant System~~ RCS pressure shall ~~not exceed~~ be maintained ≤ 2750 psia.

2.1.2 ~~APPLICABILITY:~~ In MODES 1, 2, 3, 4 and 5~~.~~ ,

2.2 2.2 SAFETY LIMIT VIOLATIONS

~~ACTION:~~

2.2.1 If SL 2.1.1 is violated, restore compliance and

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In **MODE~~S~~ 1 ~~and~~ or 2**, restore compliance and

~~Whenever the Reactor Coolant System pressure has exceeded 2750 psia,~~

2.2.2.1 be in ~~HOT STANDBY with the Reactor Coolant System pressure within its limit~~ MODE 3 within 1 hour.

2.2.2.2 In **MODE~~S~~ 3, 4 ~~and~~ or 5**, restore compliance

~~Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit~~

2.2.2.2 within 5 minutes.

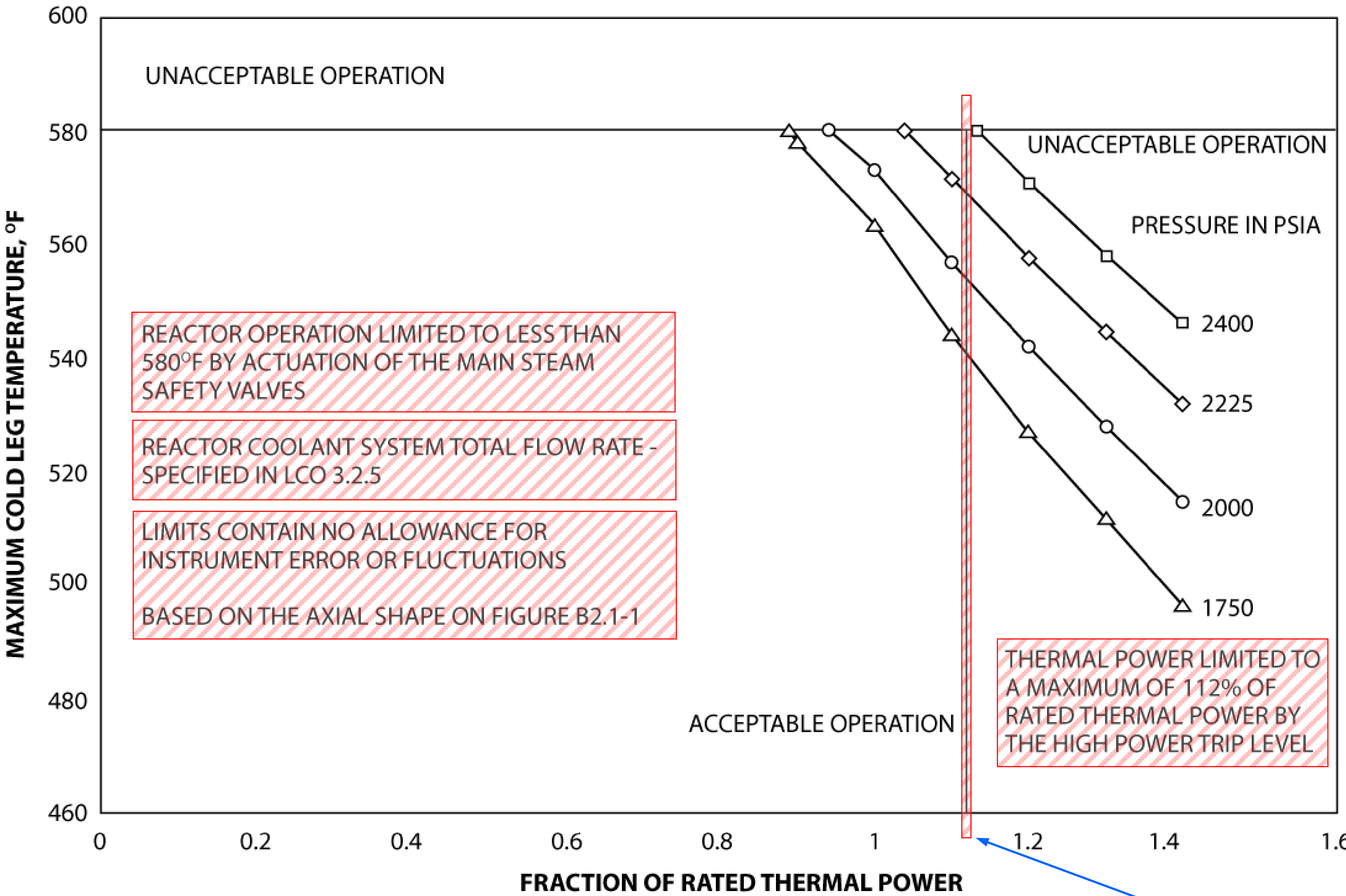


Figure 2.1.1-1

FIGURE 2.1-1~~:~~ .1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT – ~~FOUR REACTOR COOLANT PUMPS OPERATING~~

(page 1 of 1)

A03

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

See
ITS 3.3.1

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

See ITS 3.3.1

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level – High (1) Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of < 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow – Low (1) Four Reactor Coolant Pumps Operating	≥ 95% of minimum reactor coolant flow with 4 pumps operating *	≥ 95% of minimum reactor coolant flow with 4 pumps operating *
4. Pressurizer Pressure – High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure – High	≤ 3.3 psig	≤ 3.3 psig
6. Steam Generator Pressure – Low (2)	≥ 600 psia	≥ 600 psia
7. Steam Generator Water Level – Low	≥ 35.0% Water Level – each steam generator	≥ 35.0% Water Level – each steam generator
8. Local Power Density – High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

* For minimum reactor coolant flow with 4 pumps operating, refer to Technical Specification LCO 3.2.5.

See ITS 3.3.1

See ITS 3.3.1

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4
9a. Steam Generator Pressure Difference High (1) (logic in TM/LP)	≤ 135 psid	≤ 135 psid
10. Loss of Turbine – Hydraulic Fluid Pressure – Low (3)	≥ 800 psig	≥ 800 psig
11. Rate of Change of Power – High (4)	≤ 2.49 decades per minute	≤ 2.49 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide-Range Logarithmic Neutron Flux power is ≥ 1% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power-Range Neutron Flux power is ≥ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4} % and above 15% of RATED THERMAL POWER.

See
ITS 3.3.1

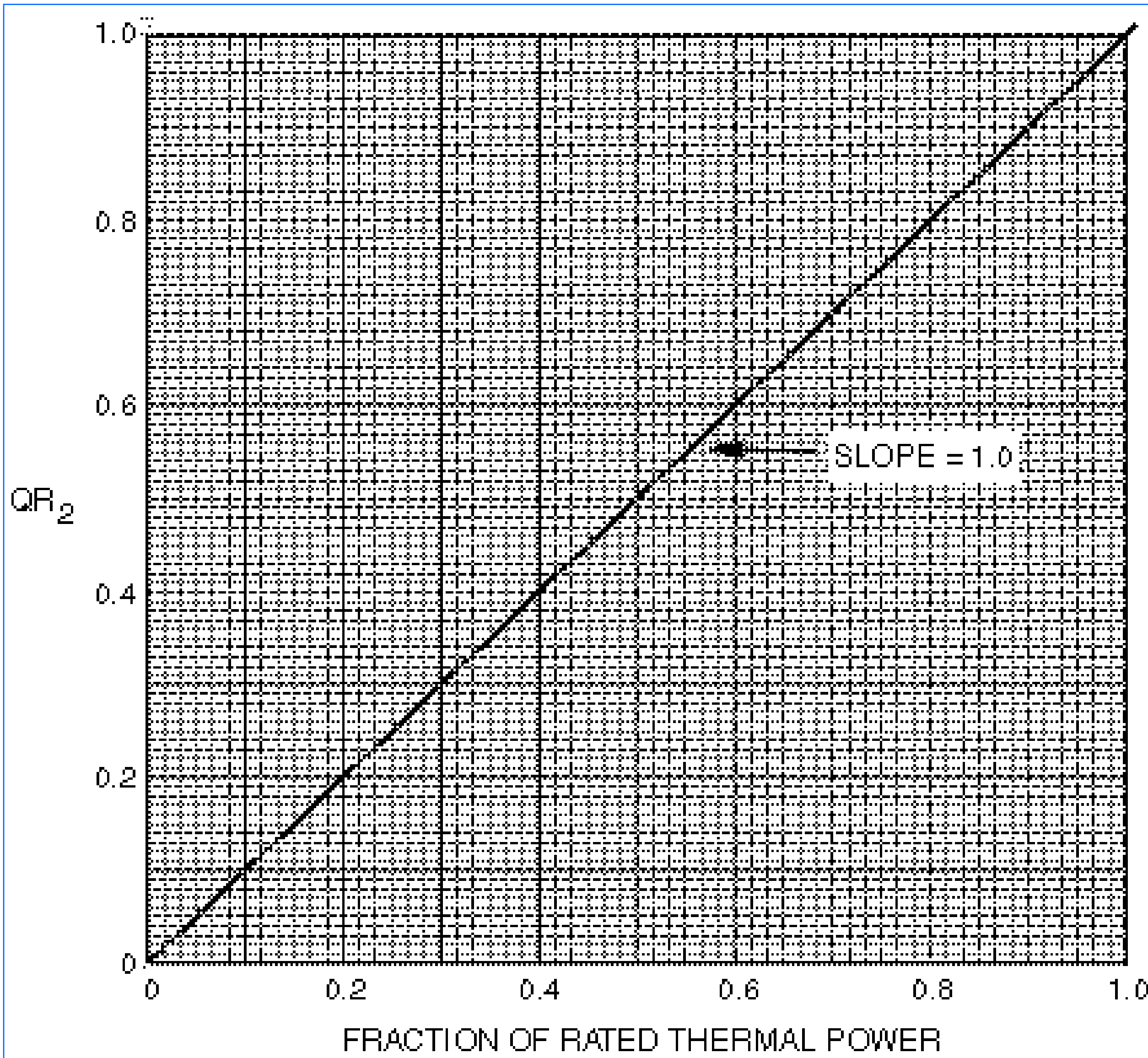
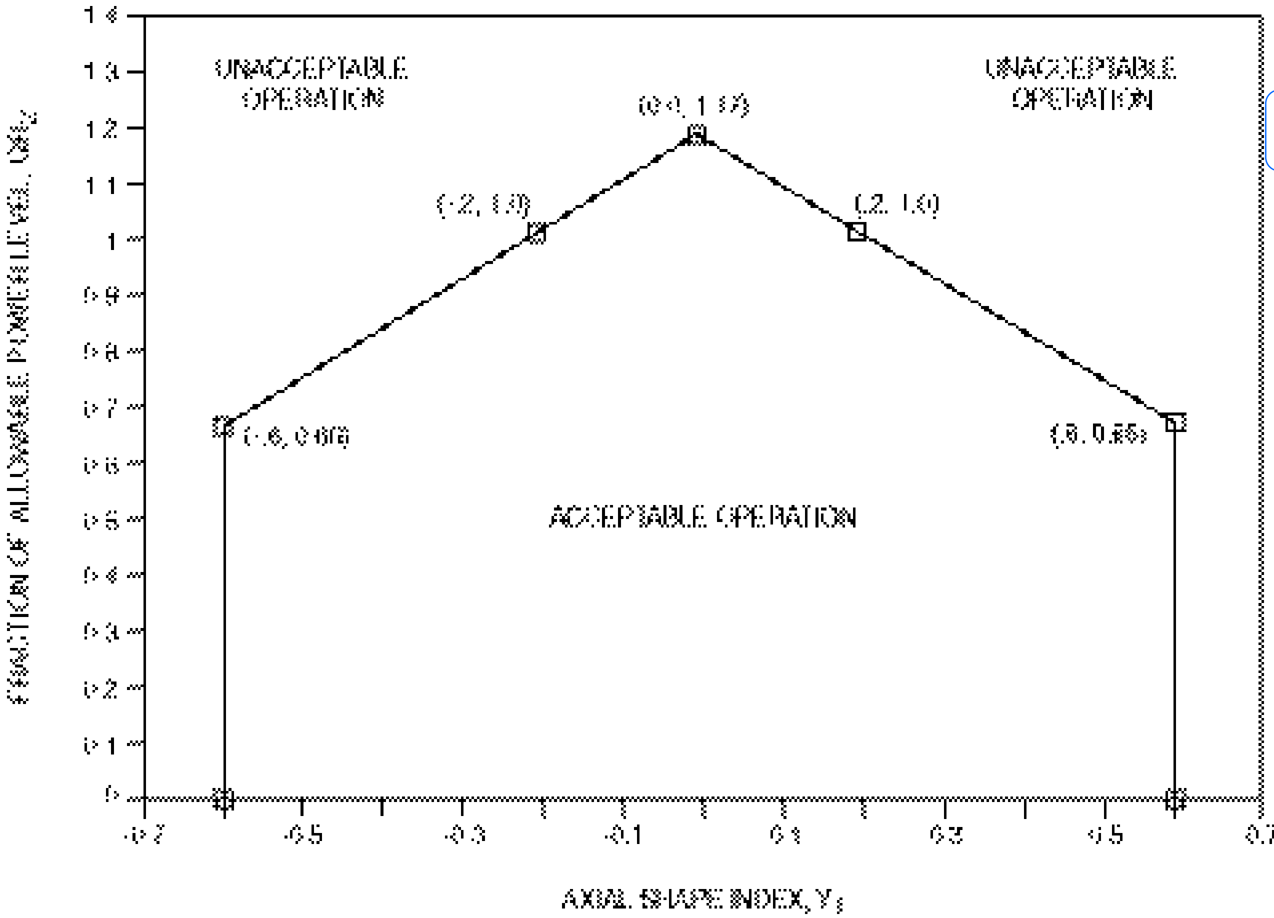


FIGURE 2.2-1

Local Power Density – High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)



See ITS 3.3.1

FIGURE 2.2-2

Local Power Density – High Trip Setpoint Part 2 (QR_2 Versus Y_1)

See
ITS 3.3.1

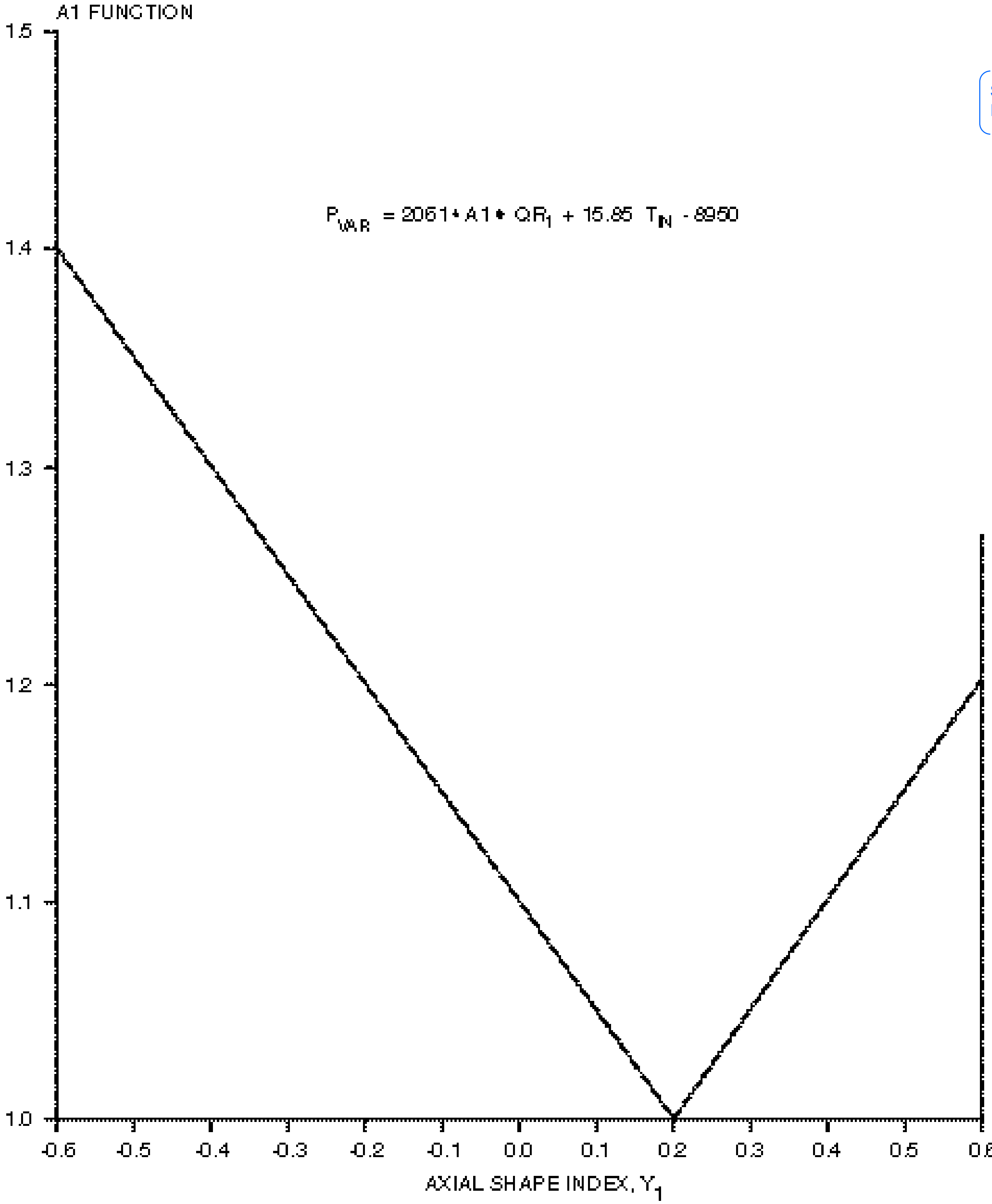


FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint

See
ITS 3.3.1

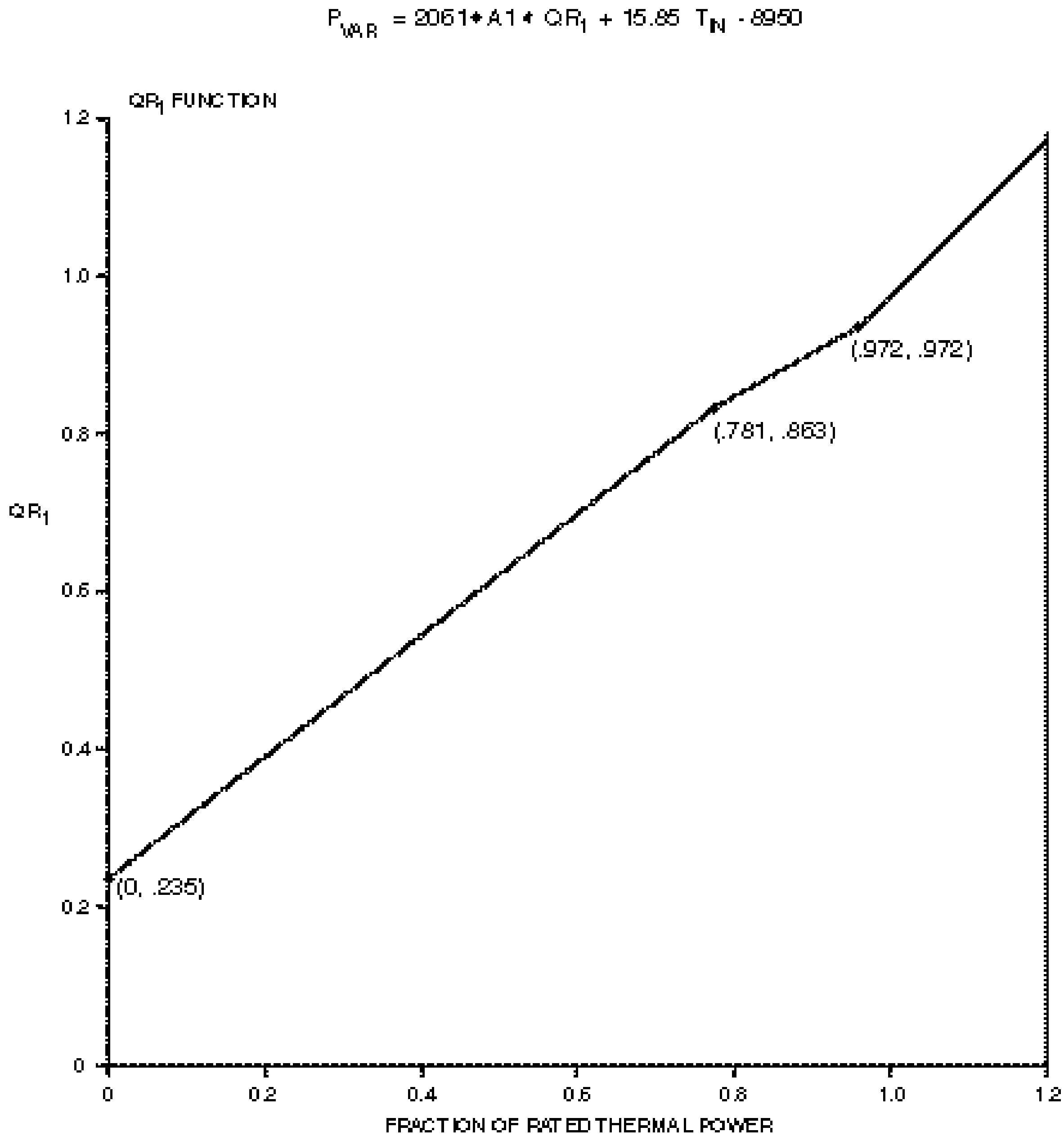


FIGURE 2.2-4

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

A01

(SLs)

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~

SLs

2.1

2.1 ~~SAFETY LIMITS~~

SLs

2.1.1

2.1.1 REACTOR CORE

~~DNBR~~

the highest operating loop

2.1.1.1 2.1.1.1 The combination of THERMAL POWER, pressurizer pressure, and ~~maximum~~ cold leg coolant temperature shall not exceed the limits shown ~~on~~ Figure 2.1-1.

in .1

In ,

2.1.1.1 ~~APPLICABILITY~~: MODES 1 and 2~~.~~

2.1.1.2 ~~ACTION:~~ 2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained below fuel melt temperature limits specified in the COLR.

M01

~~Whenever the combination of THERMAL POWER, pressurizer pressure and maximum cold leg~~
2.2.1 ~~coolant temperature has exceeded the limits shown on Figure 2.1-1,~~ be in ~~HOT STANDBY~~ within 1 hour.

MODE 3

SL

2.1.2

REACTOR COOLANT SYSTEM PRESSURE

be maintained ≤

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

RCS In ,

2.1.2 ~~APPLICABILITY~~: MODES 1, 2, 3, 4 and 5~~.~~

2.2 2.2 SAFETY LIMIT VIOLATIONS

~~ACTION:~~

2.2.1 If SL 2.1.1 is violated, restore compliance and

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE~~S~~ 1 ~~and~~ or 2, restore compliance and

MODE 3

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

2.2.2.2 In MODE~~S~~ 3, 4 ~~and~~ or 5, restore compliance

~~Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the~~
2.2.2.2 ~~Reactor Coolant System pressure to within its limit~~ within 5 minutes.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

See
ITS 3.3.1

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

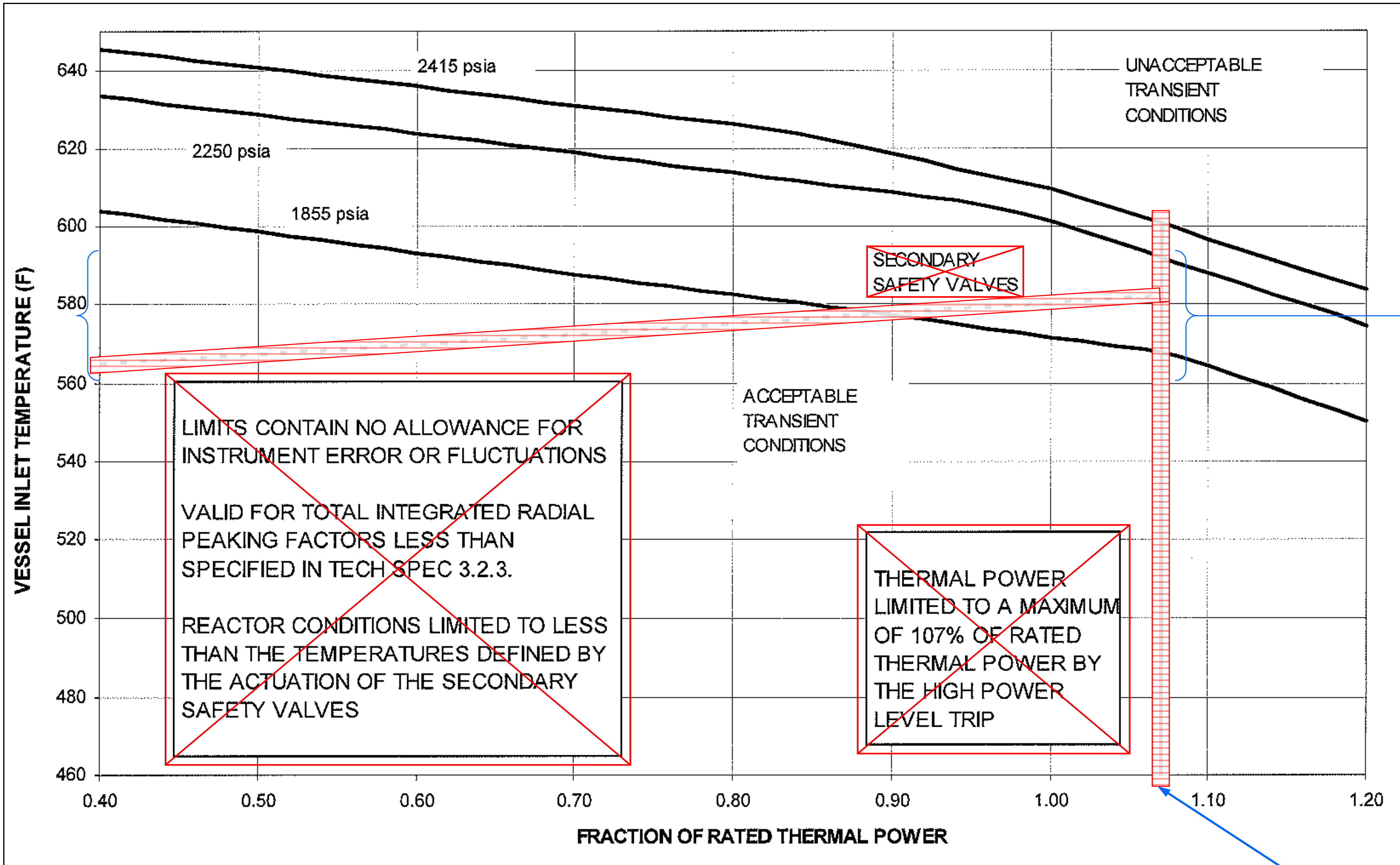


Figure 2.1.1-1

FIGURE 2.1-1~~:~~ REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES
~~FOUR REACTOR COOLANT PUMPS OPERATING~~ A03

.1 (page 1 of 1)

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

See ITS 3.3.1

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Variable Power Level – High[1]		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Pressurizer Pressure – High	≤ 2370 psia	≤ 2374 psia
4. Thermal Margin/Low Pressure[1]		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.
5. Containment Pressure – High	≤ 3.0 psig	≤ 3.1 psig
6. Steam Generator Pressure – Low	≥ 626.0 psia[2] ~~(2)~~	≥ 621.0 psia[2] ~~(2)~~
7. Steam Generator Pressure[1] Difference – High (Logic in TM/LP Trip Unit)	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level – Low~~Low~~	≥ ~~20.5~~35.0%[3] ~~(3)~~	≥ ~~19.5~~35.0%[3] ~~(3)~~

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density – High(5) Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	$\geq$ 636 gpm**	$\geq$ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power – High(4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute
14. Reactor Coolant Flow – Low(1)	$\geq$ 95.4% of minimum Reactor Coolant flow with four pumps operating*	$\geq$ 94.9% of minimum Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low(5)	$\geq$ 800 psig	$\geq$ 800 psig

See ITS 3.3.1

* For minimum reactor coolant flow with four pumps operating, refer to Technical Specification LCO 3.2.5.

** 10-minute time delay after relay actuation.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

See ITS 3.3.1

TABLE NOTATION

(1) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER during testing pursuant to Special Test Exception 3.10.3; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(3) % of the narrow range steam generator level indication.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}$% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(5) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.

See ITS 3.3.1

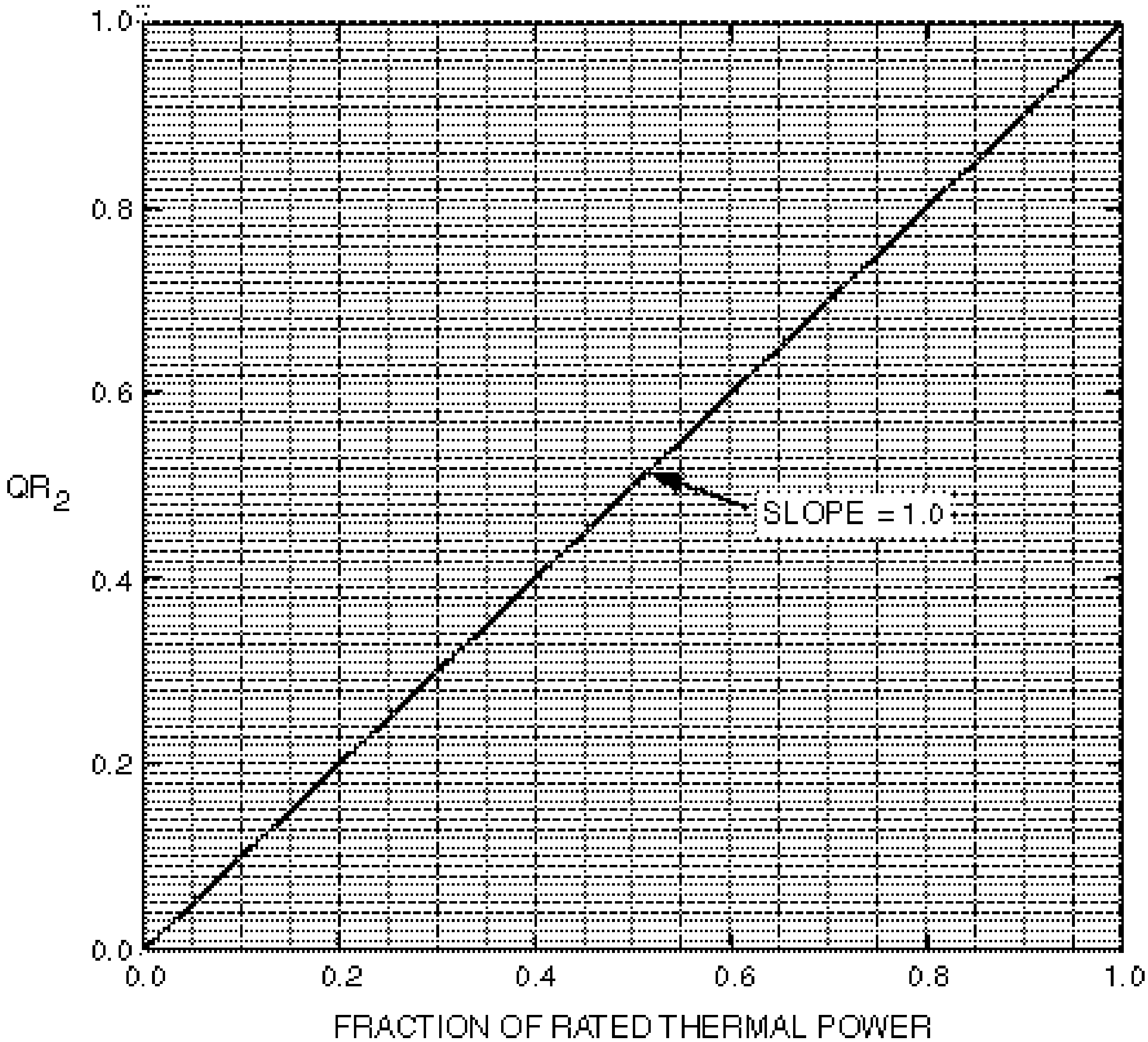


FIGURE 2.2-1

Local power density – High trip setpoint
Part 1 (Fraction of RATED THERMAL POWER versus QR_2)

See
ITS 3.3.1

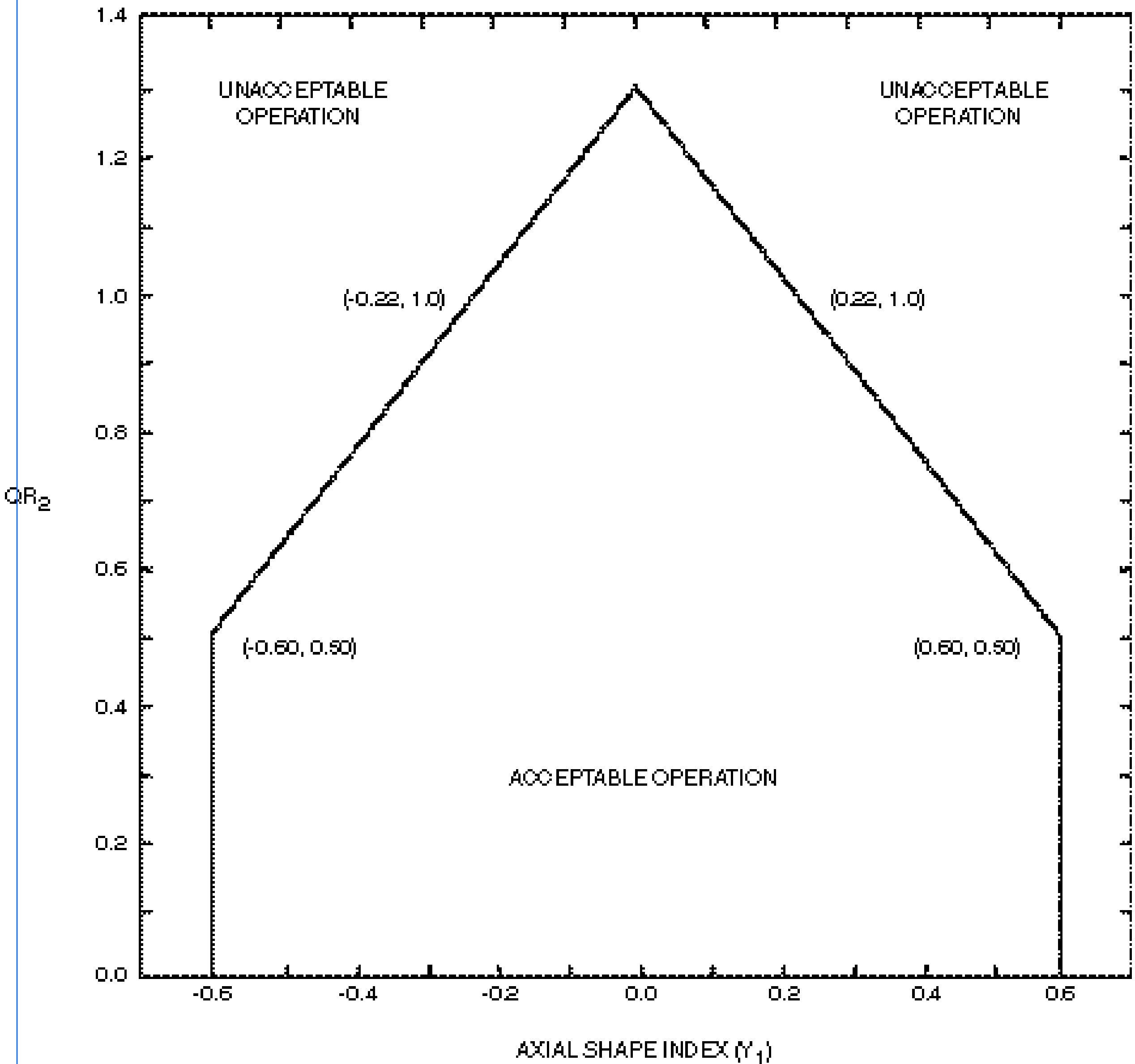


FIGURE 2.2-2

LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 2
(QR_2 versus Y_1)

See ITS 3.3.1

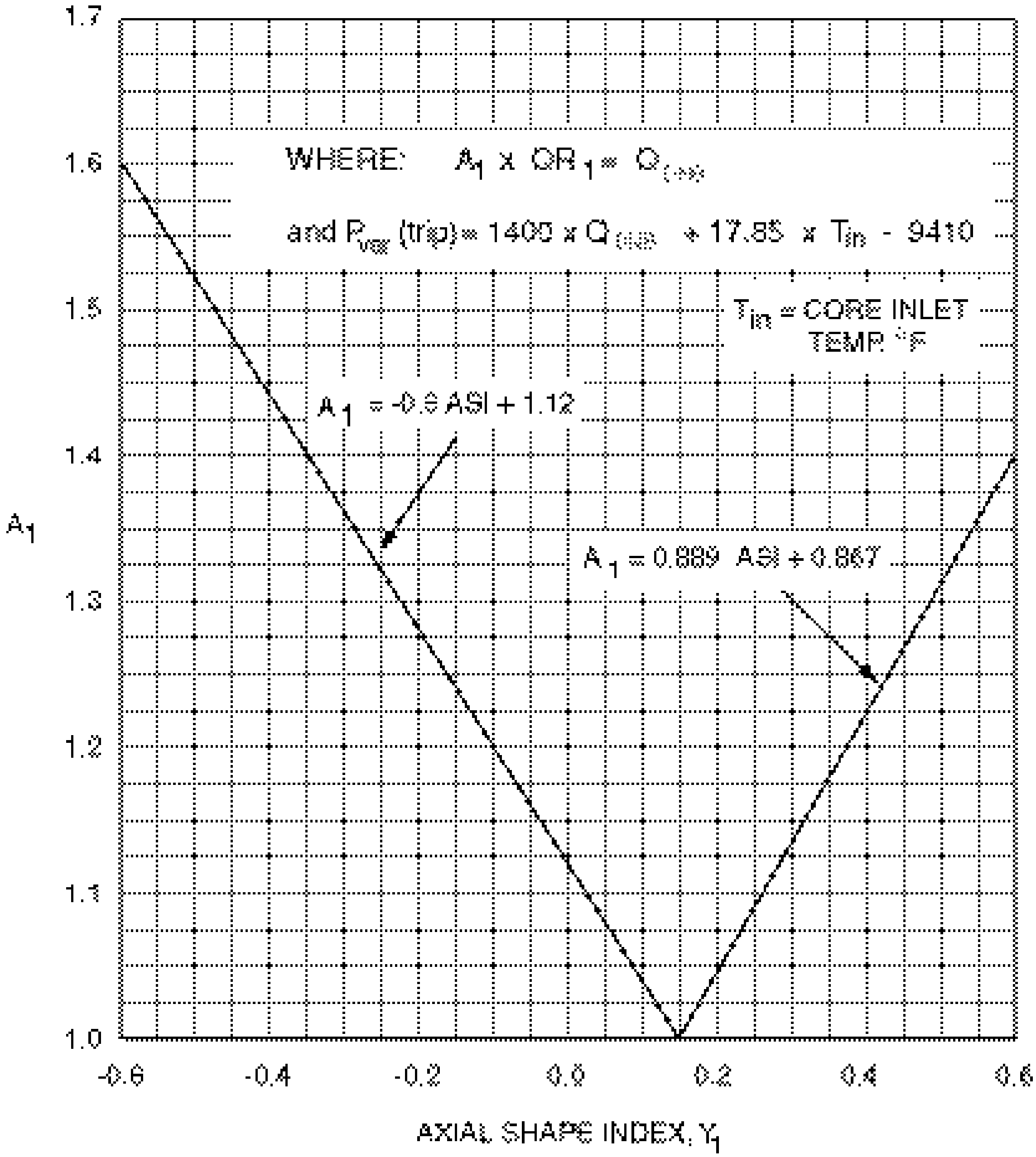


FIGURE 2.2-3

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 1 (Y_1 Versus A_1)

See
ITS 3.3.1

WHERE: $A_1 \times QR_1 = Q_{DNB}$

AND $P_{var}^{trip} = 1400 \times Q_{DNB} + 17.85 \times T_{in} - 9410$

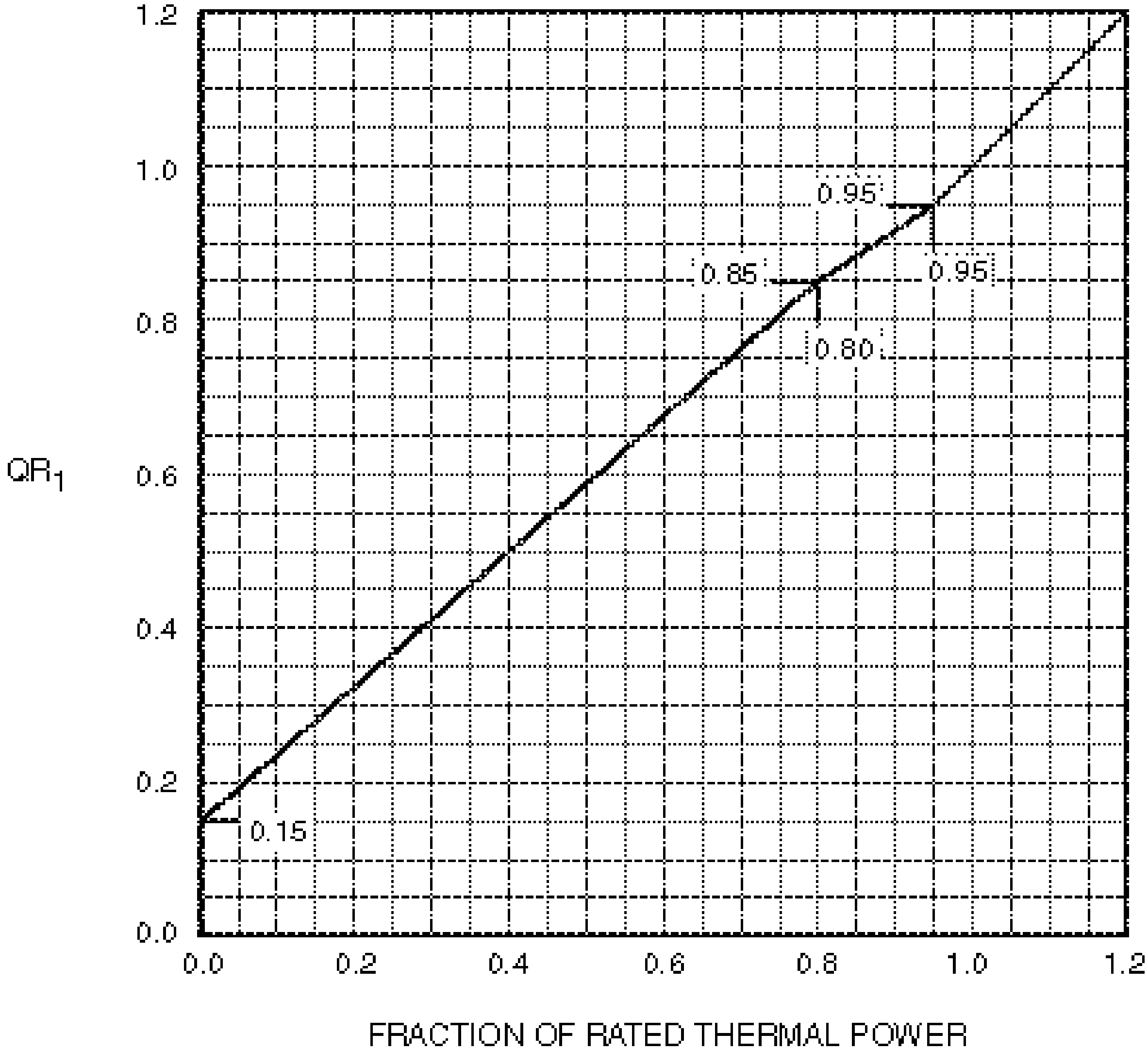


FIGURE 2.2-4

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 1 only:** CTS Figure 2.1-1 states "Reactor Coolant System total flow rate – specified in LCO 3.2.5." ITS Figure 2.1.1-1 does not contain the statement. This changes the CTS by not including a cross reference to the equivalent reference ITS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," which contains the Reactor Coolant System total flowrate. This change is acceptable because the CTS Figure 2.1-1 detail used to alert the user that the Reactor Coolant System total flowrate is covered under a separate LCO requirement is an unnecessary presentation preference. ITS does not include information only type of cross references to other Specifications. ITS Figure 2.1.1-1, Reactor Core Thermal Margin Safety Limit , is retained without the cross reference detail.

This change is designated as administrative as it incorporates an ITS convention with no technical change.

A03 PSL Unit 1 CTS Figure 2.1-1, Reactor Core Thermal Margin Safety Limit – Four Reactor Coolant Pumps Operating contains an operational limit line showing thermal power limited to a maximum of 112% of RATED THERMAL POWER by the High Power Trip. PSL Unit 2 CTS Figure 2.1-1, Reactor Core Thermal Margin Safety Limit – Four Reactor Coolant Pumps Operating contains an operational limit line showing thermal power limited to a maximum of 107% of RATED THERMAL POWER by the High Power Level Trip, and contains an operational limit line showing limits on maximum cold leg temperature due to secondary safety valve actuation. Additionally, the Unit 1 and Unit 2 figure titles include the statement "Four Reactor Coolant Pumps Operating."

The RPS operational limit line is removed from the Unit 1 and Unit 2 figures. This operational limit could be violated during a transient; however, it does not necessarily mean the reactor core SL was violated. Removing the operational limit line shifts the unacceptable area to the actual SL limits based on RCS pressure lines. The RPS operational limit is provided in the ITS Bases. The figure title text "Four Reactor Coolant Pumps Operating" is deleted from the Unit 1 and Unit 2 figures. Unit 1 and Unit 2 only operate with four reactor coolant pumps. Therefore, this text is deleted for consistency with ISTS Figure 2.1.1-1 title. The operational limit line showing limits on maximum cold leg temperature due to secondary safety valve actuation is removed from the Unit 2 figure. This operational limit could be violated during a transient; however, it does not necessarily mean the reactor core SL was violated. Removing this operational limit line shifts the unacceptable area to the actual SL limits based on RCS pressure lines.

This change is designated as administrative as it incorporates an ITS convention to remove superfluous lines not directly related to violation of the safety limit with no technical change.

MORE RESTRICTIVE CHANGES

M01 CTS 2.1.1 does not contain a Safety Limit to ensure fuel centerline temperature remains below the fuel melt temperature during normal operating conditions or design anticipated operational occurrences with adjustments for burnup and burnable poison. ITS requires Reactor Core Safety Limit 2.1.1.2, to ensure fuel centerline temperature remains below the fuel melt temperature during normal operating conditions or design anticipated operational occurrences with adjustments for burnup and burnable poison.

The purpose of ITS 2.1.1.2 is to provide a Safety Limit to ensure fuel centerline temperature remains below the fuel melt temperature during normal operating conditions or design anticipated operational occurrences with adjustments for burnup and burnable poison. This change is acceptable because it adds a Reactor Core Safety Limit that is not in the CTS.

This change is designated as more restrictive because it adds a Reactor Core Safety Limit to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 2.1.1 states “The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.” ITS states “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.” This changes the CTS by moving details shown on the CTS Figure 2.1-1 to the ITS Bases.

PSL Unit 1 CTS Figure 2.1-1 statements “Reactor operation limited to less than 580°F by actuation of Main Steam Safety Valves”, “Limits contain no allowance for instruments error or fluctuations”, “Based on the Axial Shape Index on Figure B 2.1-1,” and “THERMAL POWER limited to a maximum of 112% of RATED THERMAL POWER by the high power trip level” are moved from the Technical Specifications to the ITS Bases. The line showing THERMAL POWER at 112%

is deleted. The Reactor Core Safety Limits are retained in Technical Specification Chapter 2.0.

PSL Unit 2 CTS Figure 2.1-1 statements “Reactor Conditions limited to less than the temperatures defined by the actuation of the secondary safety valves”, “Limits contain no allowance for instruments error or fluctuations”, “Valid for Total Integrated Radial Peaking Factors less than specified in Tech Spec 3.2.3” and “THERMAL POWER limited to a maximum of 107% of RATED THERMAL POWER by the high power level trip” are moved from the Technical Specifications to the ITS Bases. The Reactor Core Safety Limits are retained in Technical Specification Chapter 2.0.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 2.1.1.1 retains the Reactor Core Safety Limit and Figure 2.1.1-1, Reactor Core Thermal Margin Safety Limit. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

2.0 SAFETY LIMITS (SLs) ~~(Analog)~~

2.1 **2.1 SLs**

2.1.1 2.1.1 Reactor Core SLs

2.1.1 and Applicability 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.

DOC M01 2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained ~~at < [5080]°F, decreasing by [58°F per 10,000 MWD/MTU] and adjusted for burnable poison per [CENPD-275-P, Revision 1-P-A or CENPD-382-P-A]~~ below fuel melt temperature limits specified in the COLR. 2

2.1.2 2.1.2 Reactor Coolant System Pressure SL

2.1.2 and Applicability In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2750~~]~~ psia.

2.2 SAFETY LIMIT VIOLATIONS

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

2.1.2 ACTION 2.2.2 If SL 2.1.2 is violated:

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

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

3

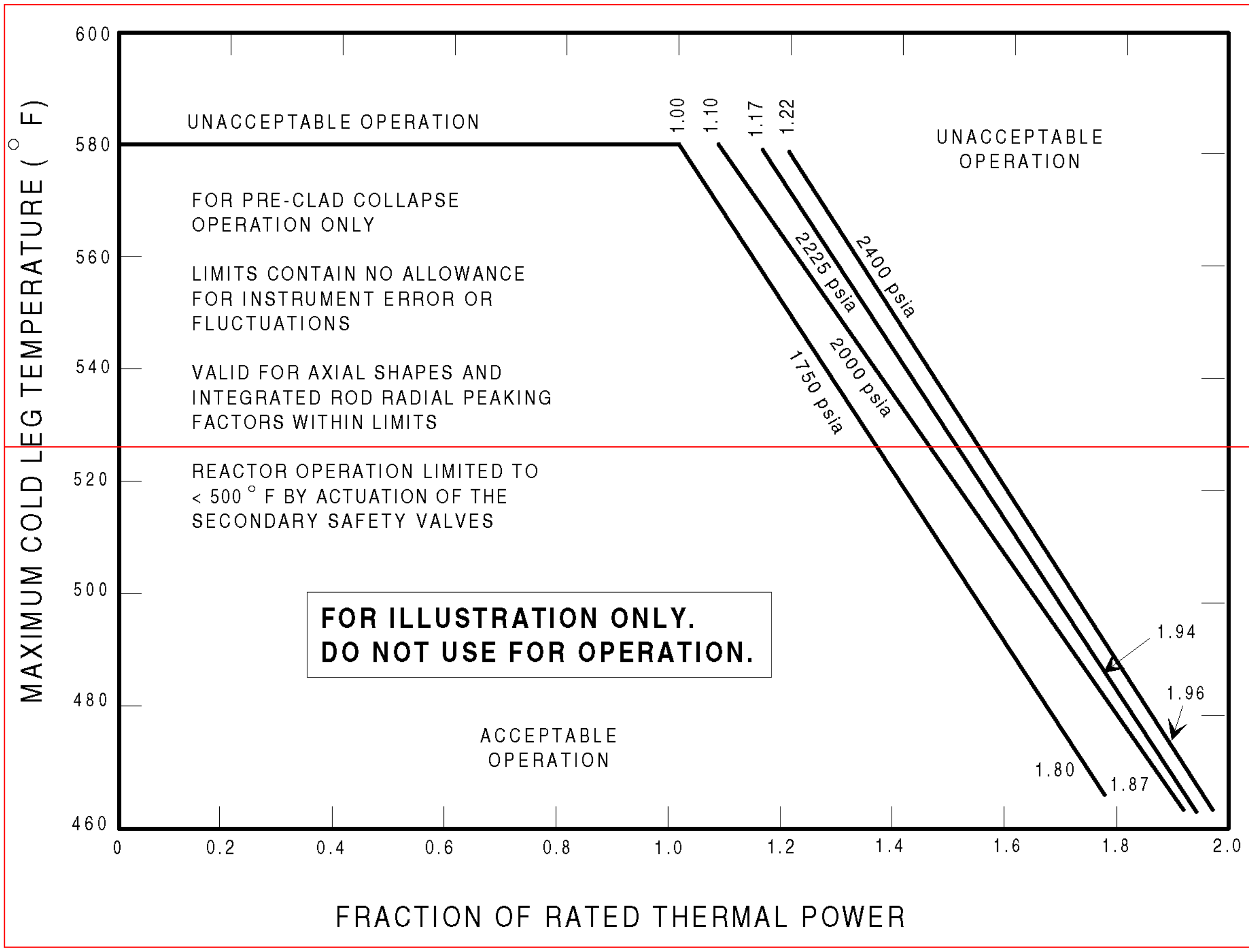


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Figure 2.1.1-1 (page 1 of 1)
Reactor Core Thermal Margin Safety Limit

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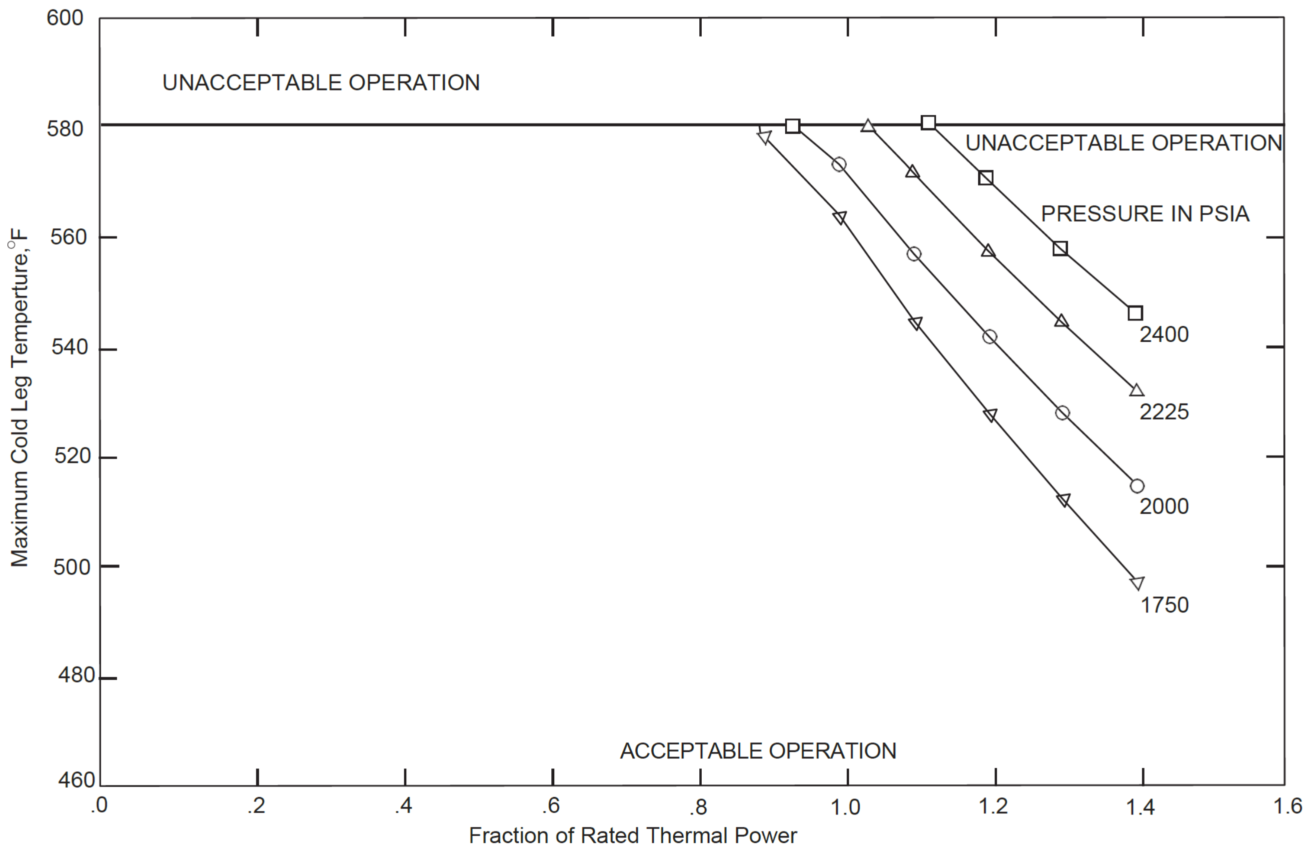
Unit 1 Only



2.0 SAFETY LIMITS (SLs) ~~(Analog)~~

2.1 **2.1 SLs**

2.1.1 2.1.1 Reactor Core SLs

2.1.1 and Applicability 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.

DOC M01 2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained ~~at < [5080]°F, decreasing by [58°F per 10,000 MWD/MTU] and adjusted for burnable poison per [CENPD-275-P, Revision 1-P-A or CENPD-382-P-A]~~. 2

below fuel melt temperature limits specified in the COLR

2.1.2 2.1.2 Reactor Coolant System Pressure SL

2.1.2 and Applicability In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2750~~]~~ psia.

2.2 SAFETY LIMIT VIOLATIONS

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

2.1.2 ACTION 2.2.2 If SL 2.1.2 is violated:

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

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

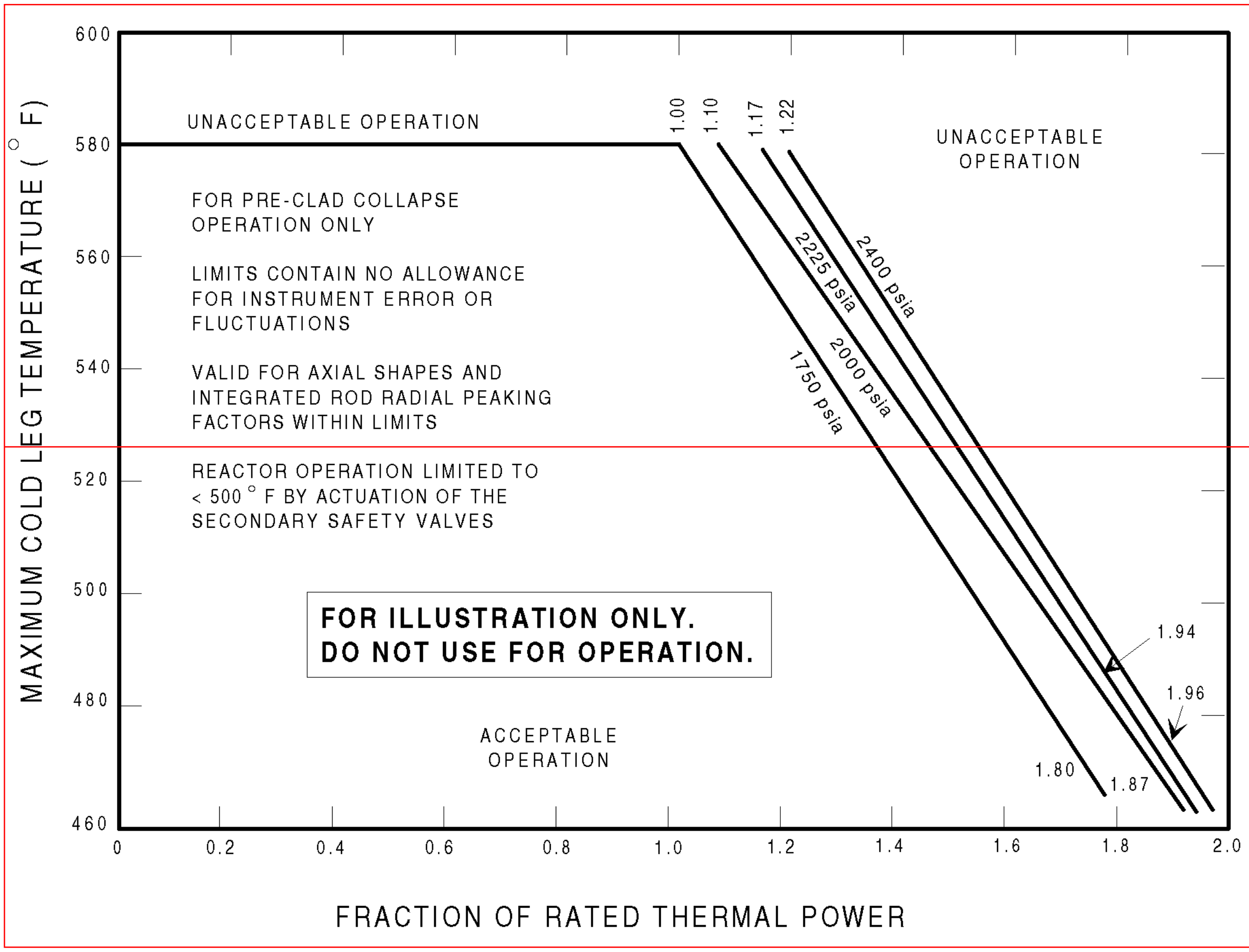


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Figure 2.1.1-1 (page 1 of 1)
Reactor Core Thermal Margin Safety Limit

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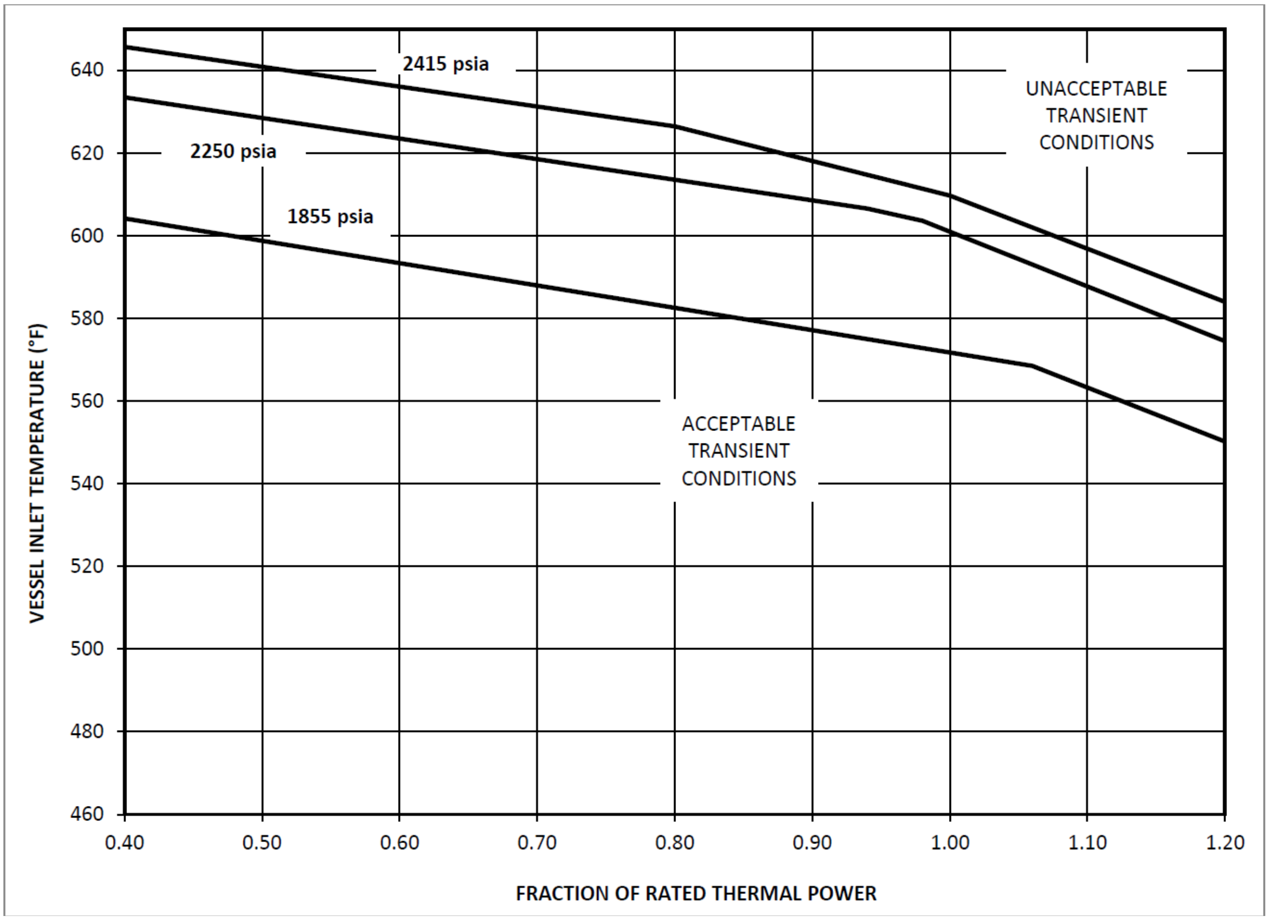
Unit 2 Only



JUSTIFICATION FOR DEVIATIONS
ISTS CHAPTER 2.0, SAFETY LIMITS (SLs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis. PSL does not currently require an SL regarding peak fuel centerline temperature. However, peak fuel centerline temperature is maintained below the fuel melt temperature specified in the COLR. The fuel melt temperature is determined using the NRC approved methodology in Siemens Topical Report XN-NF-79-56(P)(A) Revision 1 and Supplement 1, “Gadolinia Fuel Properties for LWR Fuel Safety Evaluation,” dated October 1981.

3. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 2.0 SAFETY LIMITS (SLs) ~~(Analog)~~

B 2.1.1 Reactor Core SLs ~~(Analog)~~ 5

BASES

BACKGROUND

GDC 10 (Ref. 1) requires and SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operation transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The Reactor Protective System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for RCS temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Pressurizer Pressure High trip,

b. ~~Variable High Power~~ Power Level - High trip, 1

c. Power Rate of Change - High trip,

d. Reactor Coolant Flow - Low trip,

e. Steam Generator Pressure - Low trip,

f. Steam Generator Level - Low trip,

g. ~~Axial Power Distribution~~ Local Power Density - High trip, 1

h. Thermal Margin/Low Pressure trip,

i. Steam Generator Pressure Difference trip, and

j. Steam Generator Safety Valves.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.~~5~~4, "AXIAL SHAPE INDEX (ASI)," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. 1

The SL contains no allowance for instrument error. Actuation of the main steam safety valves limits reactor coolant temperature. 4

BASES

SAFETY LIMITS

The curves provided in Figure ~~B~~ 2.1.1-1 show the loci of points of THERMAL POWER, pressurizer pressure, and highest operating loop cold leg temperature, for which the minimum ~~-~~DNBR is not less than the safety analysis limit. SL 2.1.1.2 ensures that fuel centerline temperature remains below melting. 3

SL 2.1.1.2 ensures that fuel centerline temperature remains below the fuel melt temperature ~~of [5080]°F~~ during normal operating conditions or design AOOs with adjustments for burnup and burnable poison. ~~An adjustment of [58°F per 10,000 MWD/MTU] has been established in [Topical Report CEN-386-P-A] (Ref. 3) and adjustments for burnable poisons are established based on [Topical Reports CENPD-275-P] (Ref. 4) and [CENPD-382-P-A] (Ref.5).~~ 2 2

limits specified in the COLR

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip ~~-~~function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1. 3

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable and reduces the probability of fuel damage.

Fuel melt temperature is determined using the NRC approved methodology in Reference 3. The fuel melt temperature limit is defined as [(2790 - 17.9 x P - 3.2 x B) x 1.8 + 32] °F, where P is the maximum weight percent of Gadolinia (%) and B is the maximum pin burnup (GWD/MTU). 2

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.

2. U FSAR, Section ~~[~~7.2 ~~]~~. 2

~~3. [Topical Report CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992.]~~ 2

~~4. [Topical Report CENPD-275-P, Revision 1-P-A, "CE Methodology for Core Designs Containing Gadolini-Urania Burnable Absorbers," May 1988.]~~ 2

~~5. [Topical Report CENPD-382-P- A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.]~~ 2

3. Siemens Power Corporation Topical Report XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," October 1981. 2

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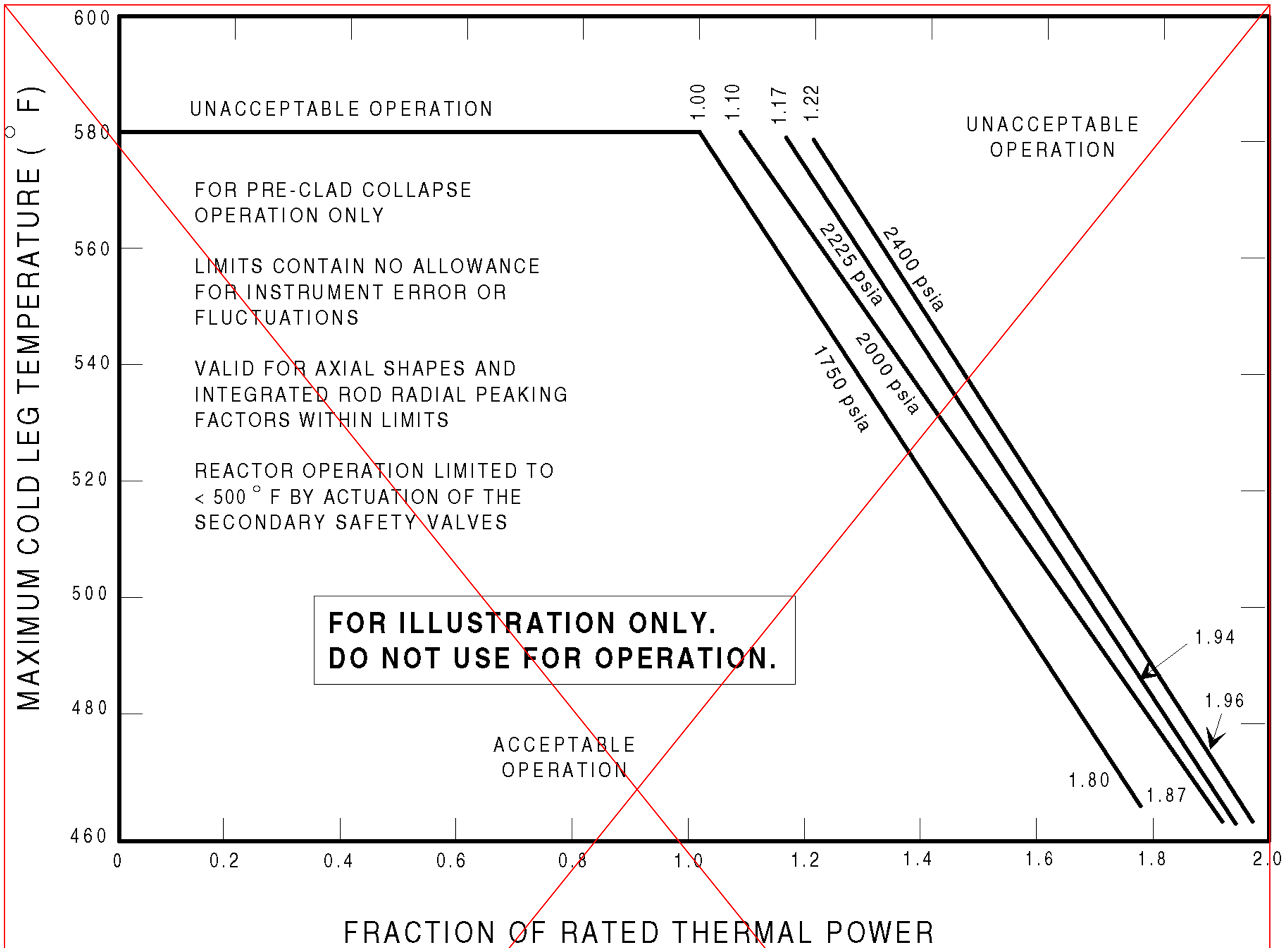

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Figure 2.1.1-1 (page 1 of 1)
Reactor Core Thermal Margin Safety Limit

1

B 2.0 SAFETY LIMITS (SLs) ~~(Analog)~~ 5

B 2.1.2 Reactor Coolant System (RCS) Pressure SL ~~(Analog)~~

BASES

BACKGROUND	The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, continued RCS integrity is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, according to GDC 28 (Ref. 1), "Reactivity Limits," reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding. The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation, when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3). Overpressurization of the RCS could result in a breach of the RCPB. If this occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).
APPLICABLE SAFETY ANALYSES	The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the Reactor Pressure - High trip have settings established to ensure that the RCS pressure SL will not be exceeded. The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and

5

BASES

APPLICABLE SAFETY ANALYSES (continued)

hence the valve size requirements and lift settings, is a ~~[~~complete loss of turbine-generator ~~external~~ load without a direct reactor trip~~]~~. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained. 2

The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the Pressurizer Pressure - High trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Safety analyses for both the Pressure - High trip and the RCS pressurizer safety valves are performed, using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs),

b. Steam Bypass Control System,

c. Pressurizer Level Control System, or

d. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings under ~~[~~USAS, Section B31.1 (Ref. 6)~~]~~, is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 2750 psia. 2

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

1

BASES

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.2.1

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III~~, Article NB-7000~~. 1

3. ASME, Boiler and Pressure Vessel Code, Section XI~~, Article IWX-5000~~.

4. 10 CFR 100.

5. U FSAR, Section ~~[~~15.2.7~~]~~. 2

~~[~~ 6. ASME, USAS B31.1, Standard Code for Pressure Piping, 1967. ~~]~~ 2

5

B 2.0 SAFETY LIMITS (SLs) ~~(Analog)~~

5

B 2.1.1 Reactor Core SLs ~~(Analog)~~

BASES

BACKGROUND

GDC 10 (Ref. 1) requires and SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operation transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The Reactor Protective System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

1

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for RCS temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Pressurizer Pressure High trip,

b. ~~Variable High Power~~ Power Level - High trip, 1

c. Power Rate of Change - High trip,

d. Reactor Coolant Flow - Low trip,

e. Steam Generator Pressure - Low trip,

f. Steam Generator Level - Low trip,

g. ~~Axial Power Distribution~~ Local Power Density - High trip, 1

h. Thermal Margin/Low Pressure trip,

i. Steam Generator Pressure Difference trip, and

j. Steam Generator Safety Valves.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR) ," and LCO 3.2.~~5~~4, "AXIAL SHAPE INDEX (ASI)," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. 1

5

The SL contains no allowance for instrument error. Actuation of the main steam safety valves limits reactor coolant temperature.

4

BASES

SAFETY LIMITS

The curves provided in Figure ~~B~~ 2.1.1-1 show the loci of points of THERMAL POWER, pressurizer pressure, and highest operating loop cold leg temperature, for which the minimum DNBR is not less than the safety analysis limit. SL 2.1.1.2 ensures that fuel centerline temperature remains below melting.

3

SL 2.1.1.2 ensures that fuel centerline temperature remains below the fuel melt temperature ~~of [5080]°F~~ during normal operating conditions or design AOOs with adjustments for burnup and burnable poison. ~~An adjustment of [58°F per 10,000 MWD/MTU] has been established in [Topical Report CEN-386-P-A] (Ref. 3) and adjustments for burnable poisons are established based on [Topical Reports CENPD-275-P] (Ref. 4) and [CENPD-382-P-A] (Ref.5)~~.

limits specified in the COLR

2

2

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

3

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable and reduces the probability of fuel damage.

Fuel melt temperature is determined using the NRC approved methodology in Reference 3. The fuel melt temperature limit is defined as [(2790 - 17.9 x P - 3.2 x B) x 1.8 + 32] °F, where P is the maximum weight percent of Gadolinia (%) and B is the maximum pin burnup (GWD/MTU).

2

1

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.

2. U FSAR, Section ~~[~~7.2 ~~]~~. 2

~~3. [Topical Report CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992.]~~ 2

~~4. [Topical Report CENPD-275-P, Revision 1-P-A, "CE Methodology for Core Designs Containing Gadolini-Urania Burnable Absorbers," May 1988.]~~ 2

~~5. [Topical Report CENPD-382-P- A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.]~~ 2

3. Siemens Power Corporation Topical Report XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," October 1981. 2

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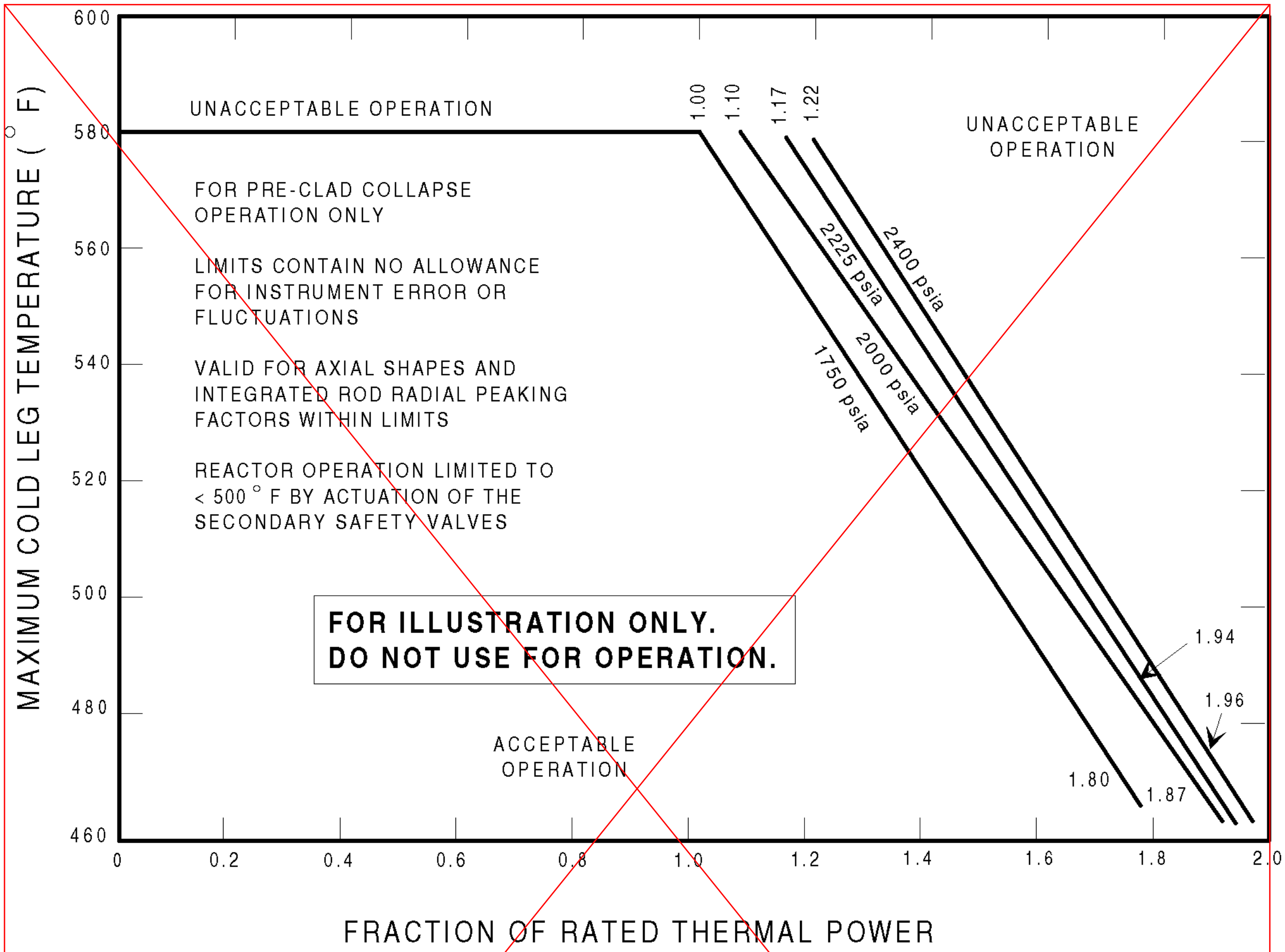

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Figure 2.1.1-1 (page 1 of 1)
Reactor Core Thermal Margin Safety Limit

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 5

B 2.0 SAFETY LIMITS (SLs) ~~(Analog)~~ 5

B 2.1.2 Reactor Coolant System (RCS) Pressure SL ~~(Analog)~~

BASES

BACKGROUND	The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, continued RCS integrity is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, according to GDC 28 (Ref. 1), "Reactivity Limits," reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding. The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation, when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3). Overpressurization of the RCS could result in a breach of the RCPB. If this occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).
APPLICABLE SAFETY ANALYSES	The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the Reactor Pressure - High trip have settings established to ensure that the RCS pressure SL will not be exceeded. The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and

BASES

APPLICABLE SAFETY ANALYSES (continued)

hence the valve size requirements and lift settings, is a ~~[~~complete loss of ~~external~~ turbine-generator load without a direct reactor trip~~]~~. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained. 2

The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the Pressurizer Pressure - High trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Safety analyses for both the Pressure - High trip and the RCS pressurizer safety valves are performed, using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs),

b. Steam Bypass Control System,

c. Pressurizer Level Control System, or

d. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings under ~~[~~USAS, Section B31.1 (Ref. 6)~~]~~, is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 2750 psia. 2

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

BASES

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.2.1

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III~~, Article NB-7000~~. 1

3. ASME, Boiler and Pressure Vessel Code, Section XI~~, Article IWX-5000~~.

4. 10 CFR 100.

5. U FSAR, Section ~~[~~15.2.7~~]~~. 2

~~[~~ 6. ASME, USAS B31.1, Standard Code for Pressure Piping, 1967. ~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ISTS CHAPTER 2.0, BASES, SAFETY LIMITS (SLs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Grammatical error is corrected. Double spacing between sentences and single spacing between words. Single spacing is corrected.

4. CTS Figure 2.1-1 contained details that are moved to the Bases. The details are added to the ITS Figure 2.1.1-1 Bases. See DOC LA01.

5. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

There are no specific No Significant Hazards Considerations for this Specification.