VOLUME 4

CNP UNITS 1 AND 2
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0
SAFETY LIMITS

Revision 0

LIST OF ATTACHMENTS

1. ITS Chapter 2.0

ATTACHMENT 1

ITS Chapter 2.0, Safety Limits

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

ITS

A.1

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ A.1

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in ~~Figure 2.1-1~~ ~~for 4 loop operation~~. A.2

the COLR

Add proposed SL 2.1.1.1 and SL 2.1.1.2 LA.1

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2

2.2.2.1 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

2.2.2.2 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

ITS Chapter 2.0

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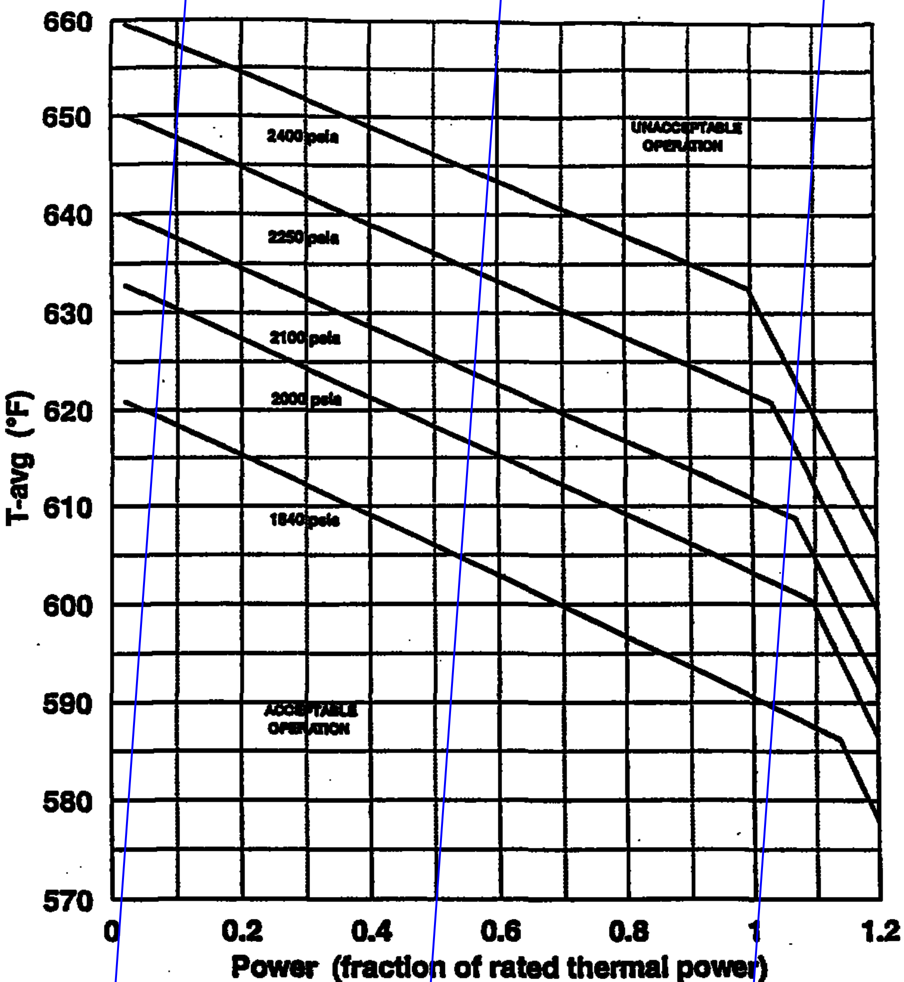
ITS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

A.1



LA.1

PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN °F)		
1840	(0.02, 620.86),	(1.136, 586.17),	(1.2, 577.94)
2000	(0.02, 632.79),	(1.094, 600.31),	(1.2, 586.52)
2100	(0.02, 639.85),	(1.068, 608.72),	(1.2, 591.77)
2250	(0.02, 649.96),	(1.031, 620.83),	(1.2, 599.40)
2400	(0.02, 659.52),	(0.996, 632.42),	(1.2, 606.63)

Figure 2.1-1 Reactor Core Safety Limits

ITS

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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A.1

ITS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system 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.

See ITS 3.3.1

ACTION:

With a reactor trip system 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.

Attachment 1, Volume 4, Rev. 0, Page 8 of 46

A.1

ITS

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

A.1

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - less than or equal to 25% of RATED THERMAL POWER High Setpoint - less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - less than or equal to 26% of RATED THERMAL POWER High Setpoint - less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1875 psig	Greater than or equal to 1865 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level - - High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

See ITS 3.3.1

*Design flow is 1/4 Reactor Coolant System total flow rate from Table 3.2.-1.

ITS

A.1

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

A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13.	Steam Generator Water Level -- Low-Low	Greater than or equal to 17% of narrow range instrument span - each steam generator	Greater than or equal to 16% of narrow range instrument span - each steam generator
14.	Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Less than or equal to 0.71 x 10^6 lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator	Less than or equal to 0.73 x 10^6 lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator
15.	Undervoltage - Reactor Coolant Pumps	Greater than or equal to 2750 volts - each bus	Greater than or equal to 2725 volts - each bus
16.	Underfrequency - Reactor Coolant Pumps	Greater than or equal to 57.5 Hz - each bus	Greater than or equal to 57.4 Hz each bus
17.	Turbine Trip		
	A. Low Fluid Oil Pressure	Greater than or equal to 800 psig	Greater than or equal to 750 psig
	B. Turbine Stop Valve Closure	Greater than or equal to 1% open	Greater than or equal to 1% open
18.	Safety Injection Input from ESF	Not Applicable	Not Applicable
19.	Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

See ITS 3.3.1

ITS

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}\right] (T-T') + K_3 (P-P') - f_1(\Delta I)\right]$

Where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, °F
T'	=	Indicated T_{avg} at RATED THERMAL POWER (≤ 574.0 °F)
P	=	Pressurizer pressure, psig
P'	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig)
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs. τ_2 = 4 secs.
S	=	Laplace transform operator

See ITS 3.3.1

ITS

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

$K_1 = 1.17$
$K_2 = 0.0230$
$K_3 = 0.00110$

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37 percent and +3 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 0.33 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +3 percent, the ΔT trip setpoint shall be automatically reduced by 2.34 percent of its value at RATED THERMAL POWER.

See ITS 3.3.1

Attachment 1, Volume 4, Rev. 0, Page 12 of 46

ITS

A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_0 \; [K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T'') - f_2(\Delta I)]$

Where:	ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T''	=	Indicated T_{avg} at RATED THERMAL POWER (≤ 562.1 °F)
	K_4	=	1.083
	K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.0015 for $T > T''$; $K_6 = 0$ for $T \leq T''$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constants utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
	S	=	Laplace transform operator
	$f_2(\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

See ITS 3.3.1

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73.

b. Each REPORTABLE EVENT shall be reviewed by the PORC, and the results of this review shall be submitted to the NSRB and the Site Vice President.

See CTS Chapter 6.0

6.7 SAFETY LIMIT VIOLATION

2.2 6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Chairman of the NSRB shall be notified within 24 hours.

A.3

LA.2

b. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems or structures; and (3) corrective action taken to prevent recurrence.

A.3

c. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the NSRB and the Senior Vice President – Nuclear Operations within 14 days of the violation.

LA.2

d. Operation of the unit shall not be resumed until authorized by the Commission.

A.3

Attachment 1, Volume 4, Rev. 0, Page 14 of 46

ITS

A.1

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ A.1

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in ~~Figure 2.1-1~~ ~~for 4 loop operation~~. A.2

the COLR

Add proposed SL 2.1.1.1 and SL 2.1.1.2 LA.1

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2

2.2.2.1 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

2.2.2.2 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

ITS

A.1

DESIGN FLOW = 91.600 GPM/LOOP

DESCRIPTION OF SAFETY LIMITS

Pressure (psia)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)
1775	0.00	615.4	0.98	583.8	1.02	580.9	1.2	558.1
2000	0.00	631.8	0.86	605.8	0.96	597.5	1.2	568.5
2100	0.00	639.1	0.82	614.0	0.96	601.6	1.2	573.1
2250	0.00	649.2	0.72	628.6	0.98	605.2	1.2	580.4
2400	0.00	659.0	0.62	642.0	1.1	599.0	1.2	588.1

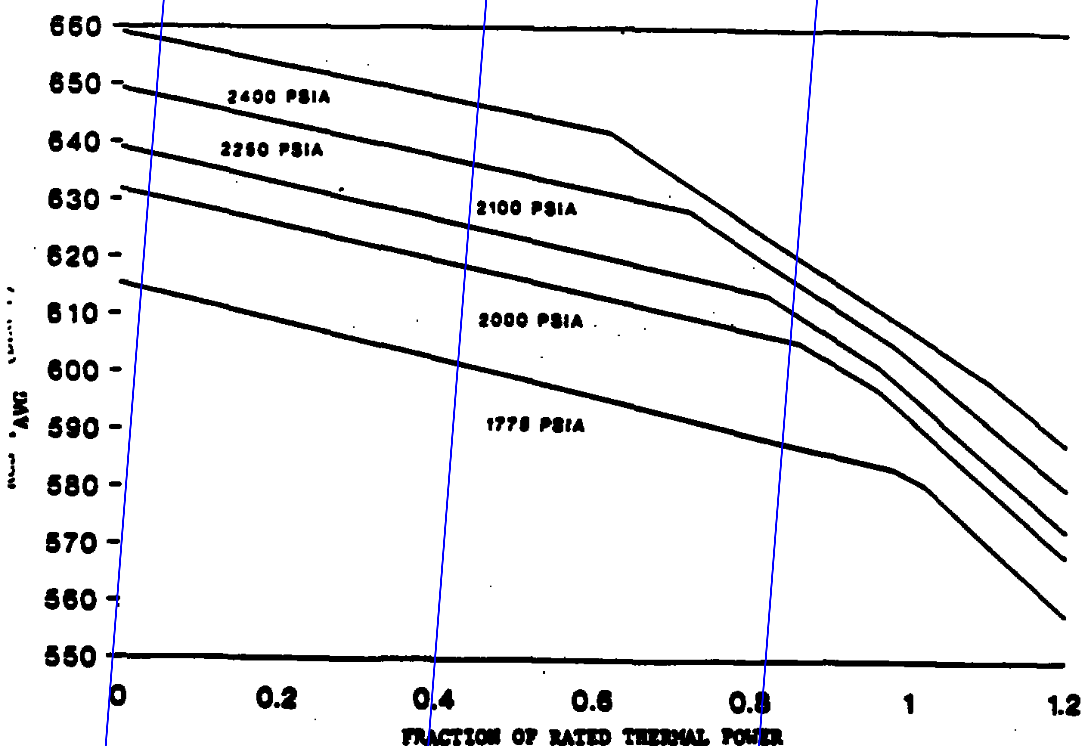

Figure 2.1-1 Reactor Core Safety Limits
Four Loops in Operation

LA.1

COOK NUCLEAR PLANT - UNIT 2 2-2 AMENDMENT NO. 82,107, 134

ITS

A.1

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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Attachment 1, Volume 4, Rev. 0, Page 17 of 46

ITS

A.1

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system 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.

See ITS 3.3.1

ACTION:

With a reactor trip system 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.

A.1

ITS

See ITS 3.3.1

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - Less than or equal to 25% of RATED THERMAL POWER	Low Setpoint - Less than or equal to 26% of RATED THERMAL POWER
	High Setpoint - Less than or equal to 109% of RATED THERMAL POWER	High Setpoint - Less than or equal to 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5. Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3 x 10^5 counts per second
7. Overtemperature Delta T	See Note 1	See Note 3
8. Overpower Delta T	See Note 2	See Note 4
9. Pressurizer Pressure -- Low	Greater than or equal to 1950 psig	Greater than or equal to 1940 psig
10. Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11. Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12. Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

* Design flow is 91,600 gpm per loop.

A.1

ITS

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	Greater than or equal to 21% of narrow range instrument span - each steam generator	Greater than or equal to 19.2% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Less than or equal to 1.47 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator	Less than or equal to 1.56 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	Greater than or equal to 2905 volts - each bus	Greater than or equal to 2870 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	Greater than or equal to 57.5 Hz - each bus	Greater than or equal to 57.4 Hz - each bus
17. Turbine Trip		
A. Low Fluid Oil Pressure	Greater than or equal to 58 psig	Greater than or equal to 57 psig
B. Turbine Stop Valve Closure	Greater than or equal to 1% open	Greater than or equal to 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

See ITS 3.3.1

Attachment 1, Volume 4, Rev. 0, Page 20 of 46

ITS

A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1:

Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2[(1 + \tau_1 S)/(1 + \tau_2 S)](T - T') + K_3(P - P') - f_1(\Delta I)]$

Where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, °F
T'	=	Indicated T_{avg} at RATED THERMAL POWER less than or equal to 576.0 °F
P	=	Pressurizer Pressure, psig
P'	=	2235 psig (indicated RCS nominal operating pressure)
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg}: τ_1 = 28 secs, τ_2 = 4 secs.
S	=	Laplace transform operator

See ITS 3.3.1

COOK NUCLEAR PLANT - UNIT 2 2-7 AMENDMENT NO. 82,134

ITS

A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

4 Loops in Operation

K1 = 1.09

K2 = 0.01331

K3 = 0.00058

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -33 percent and +6 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER);

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -33 percent, the ΔT trip setpoint shall be automatically reduced by 3.5 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by 1.0 percent of its value at RATED THERMAL POWER.

See ITS 3.3.1

ITS

A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o\ [K_4 - K_5[\tau_3 S/(1+\tau_3 S)]T - K_6[T-T''] - f_2(\Delta I)]$

Where:

ΔT_o	= Indicated ΔT at rated power
T	= Average temperature, °F
T''	= Indicated T_{avg} at RATED THERMAL POWER less than or equal to 576.0 °F
K_4	= 1.08
K_5	= 0.02/°F for increasing average temperature and 0 for decreasing average temperature
K_6	= 0.00197 for T greater than T"; K_6 = 0 for T less than or equal to T"
$\tau_3 S/(1+\tau_3 S)$	= The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	= Time constant utilized in the rate lag controller for T_{avg}; τ_3 = 10 secs.
S	= Laplace transform operator
$f_2(\Delta I)$	= 0.0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.3 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.0 percent ΔT span.

See ITS 3.3.1

Attachment 1, Volume 4, Rev. 0, Page 23 of 46

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73.

b. Each REPORTABLE EVENT shall be reviewed by the PORC, and the results of this review shall be submitted to the NSRB and the Site Vice President.

See CTS Chapter 6.0

6.7 SAFETY LIMIT VIOLATION

2.2 6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. ~~The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour.~~ ~~The Chairman of the NSRB shall be notified within 24 hours.~~ A.3 LA.2

b. ~~A Safety Limit Violation Report shall be prepared.~~ ~~This report shall be reviewed by the PORC.~~ ~~The report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems or structures; and (3) corrective action taken to prevent recurrence.~~ LA.2 A.3

c. ~~The Safety Limit Violation Report shall be submitted to the Commission,~~ ~~the Chairman of the NSRB and the Senior Vice President – Nuclear Operations within 14 days of the violation.~~ LA.2

d. ~~Operation of the unit shall not be resumed until authorized by the Commission.~~ A.3

Attachment 1, Volume 4, Rev. 0, Page 24 of 46

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS

A.1 In the conversion of the CNP 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-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

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

A.2 CTS 2.1.1 references a curve providing limits on THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (Tavg) "for 4 loop operation." ITS 2.1.1 does not contain this amplifying information.

This change is acceptable because the requirements have not changed. Both the ITS (ITS 3.4.4) and the CTS (CTS 3/4.4.1.1) require all four loops to be in operation in the applicable MODES (MODES 1 and 2). This change is designated as administrative because it eliminates redundant information in the CTS.

A.3 In the event that a safety limit is violated, CTS 6.7.1.a requires the NRC Operations Center to be notified by telephone within one hour, CTS 6.7.1.b requires a Safety Limit Violation Report to be prepared and specifies the information the report must contain, CTS 6.7.1.c requires the report to be submitted to the NRC, and CTS 6.7.1.d precludes resumption of operation of the unit until authorized by the NRC. The ITS does not specify any of these requirements.

These deletions are acceptable since the actual requirements are not being changed. These CTS requirements are duplicative of those currently located in 10 CFR 50.36(c)(1). Since CNP is required by the Operating License to comply with 10 CFR 50, the deletion of these requirements from the Technical Specifications is acceptable. The changes are designated as administrative since they are duplicative of regulations.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 2.1.1 requires that the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature not exceed the limits in Figure 2.1-1.

Attachment 1, Volume 4, Rev. 0, Page 25 of 46

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS

ITS 2.1.1 states that the combination of THERMAL POWER, RCS highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR and provides specific limits on DNBR and peak fuel centerline temperature. This changes the CTS by relocating limits that must be confirmed on a cycle specific basis to the COLR. The limiting Safety Limit parameters are retained in the SL.

The removal of these cycle specific parameter limits from the Technical Specifications to the COLR and the retention of the limiting Safety Limits in the Technical Specifications is acceptable because the cycle specific limits are developed or utilized under NRC-approved methodologies that ensure the Safety Limits are met. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Safety Limits. NRC-approved Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," determined that the specific values for these parameters may be relocated to the COLR provided the limiting Safety Limits continue to appear in the Technical Specifications. The methodologies used to develop the parameters in the COLR were approved by the NRC in accordance with Generic Letter 88-16. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits of the safety analysis are met (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits). This change is designated as a less restrictive removal of detail change because information relating to cycle specific parameter limits is being removed from the Technical Specifications.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* In the event that a Safety Limit is violated, CTS 6.7.1.a requires the Chairman of the NSRB to be notified within 24 hours, CTS 6.7.1.b requires the Safety Limit Violation Report to be reviewed by the PORC, and CTS 6.7.1.c requires the report to be submitted to the Chairman of the NSRB and the Senior Vice President - Nuclear Operations within 14 days of the violation. The ITS does not include these requirements; they have been relocated to the Quality Assurance Program Description (QAPD).

The removal of these details for making notifications/reports 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. The notification occurs following the Safety Limit violation and the reports are after-the-fact reports, thus they are not necessary to assure safe operation of the unit. The ITS still requires the unit to be shut down, and 10 CFR 50.36(c)(1) provides NRC reporting requirements and requires the NRC's permission to be obtained prior to restarting the unit. Also, this change is acceptable because these types of details will be adequately controlled in the QAPD. The QAPD is controlled under 10 CFR 50.54 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of

Attachment 1, Volume 4, Rev. 0, Page 26 of 46

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS

detail change because reporting requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

None

Attachment 1, Volume 4, Rev. 0, Page 27 of 46

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

CTS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ~~≥ [1.17 for the WRB-1/WRB-2 DNB correlations]~~ INSERT 1 (1)

2.1.1.2 The peak fuel centerline temperature shall be maintained < [5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup].

2.1.2

2.1.2 Reactor Coolant System Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ [2735] psig. (1)

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)

2.1.2 Action 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.

1

INSERT 1

greater than or equal to the 95/95 DNB criterion for the DNB correlations and methodologies specified in Specification 5.6.5

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0, SAFETY LIMITS

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ISTS 2.2.1 states that if SL 2.1.1 is violated to "restore compliance and be in MODE 3 within 1 hour." SL 2.1.1 is only applicable in MODES 1 and 2. Therefore, since ISTS 2.2.1 requires being in MODE 3 within 1 hour, the "restore compliance" action is superfluous and has been deleted. This is also consistent with the CTS, which only requires being in MODE 3 within 1 hour.

Attachment 1, Volume 4, Rev. 0, Page 31 of 46

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

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

INSERT 1 (1)

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during ~~steady state operation, normal operational transients, and anticipated operational occurrences (AOOs)~~. INSERT 2 This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature (Ref. 2) (2).

The restrictions of this SL prevent overheating of the fuel and cladding, as well as 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.

Trip T (2)

The proper functioning of the Reactor ~~Protection~~ System (R~~P~~S) and ~~steam generator~~ safety valves prevents violation of the reactor core SLs.

main steam (2)

1 **INSERT 1**

Plant Specific Design Criterion (PSDC) 6

1 **INSERT 2**

all expected conditions of normal operation, with appropriate margins for uncertainties and specified transient situations that can be anticipated

BASES INSERT 3 2

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 DNB, and 6

b. The hot fuel pellet in the core must not experience centerline fuel melting. RTS 3

Allowable Values 3

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and RCS flow, ΔI, 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. 2 4

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves. MAIN steam 2 7 7 2 2

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. 3 4 2 3

2 Allowable Values

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB, and 6

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

2

INSERT 3

operational transients and transient conditions arising from faults of moderate frequency

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during ~~steady state~~ operation, ~~normal~~ operational transients, and ~~anticipated operational occurrences (AOOs)~~. To ensure that ~~the~~ RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the ~~saturation~~ enthalpy and ~~the~~ the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during ~~steady state~~ operation, ~~normal~~ operational transients, and ~~AOOs~~.

T 2 normal T 2 4 and INSERT 4 2 the 4 saturation 4 that T 2 4 2 INSERT 4 Normal and

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, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

main steam 2

SAFETY LIMIT VIOLATIONS

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

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10.~~ INSERT 5 1
2. ~~U~~FSAR, Section ~~7.2~~. INSERT 6 2

5 3 2

2

INSERT 4

transient conditions arising from faults of moderate frequency

1

INSERT 5

UFSAR, Section 1.4.2.

2

INSERT 6

2. UFSAR, Section 3.5.3 (Unit 1) and Section 3.4.1 (Unit 2).

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

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, the continued integrity of the RCS 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 pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding. INSERT 1 INSERT 2 INSERT 3 (4) (1)

2485 psig (2)

anticipated operational transients

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited 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). (2)

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach 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). (2)

pressurizer

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure 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%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control

1

INSERT 1

Plant Specific Design Criterion (PSDC) 9, "Reactor Coolant Pressure Boundary"

1

INSERT 2

shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. The RCS, in conjunction with its control and protective provisions, was designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation or anticipated system interactions, and to maintain the stresses within allowable code stress limits.

1

INSERT 3

PSDC 33, "Reactor Coolant Pressure Boundary Capability" (Ref. 1), the reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Attachment 1, Volume 4, Rev. 0, Page 40 of 46

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

Allowable Values

anticipated operational transients

determined

The Reactor Trip System ~~setpoints~~ (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and ~~AOOs~~. The reactor high pressure trip ~~setpoint~~ is specifically ~~set~~ to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices. 2

Allowable Value

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

a. Pressurizer power operated relief valves (PORVs); 6

b. ~~Steam line relief valve~~ Steam generator PORVs; 6 2

c. Steam Dump System;

d. Reactor Control System; 6

e. Pressurizer Level Control System; or

f. Pressurizer spray valve. 5 2

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under (USAS, Section B31.1 (Ref. 6)) is 120% of design pressure. 5 The ~~most~~ more limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig. 4

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

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.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

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 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. The 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.~~ INSERT 4 1

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

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

4. 10 CFR 100.

5. (U)FSAR, Section 7.2. 2 5

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

1

INSERT 4

UFSAR, Sections 1.4.2 and 1.4.6.

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0 BASES, SAFETY LIMITS

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

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

3. Editorial correction made to the Bases.

4. Typographical/grammatical error corrected.

5. The brackets have been removed and the proper plant specific information/value has been provided.

6. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Specific No Significant Hazards Considerations (NSHCs)

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

There are no specific NSHC discussions for this Chapter.

Attachment 1, Volume 4, Rev. 0, Page 46 of 46