ENCLOSURE 2

VOLUME 4

TURKEY POINT NUCLEAR GENERATING STATION UNIT 3 AND UNIT 4

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0 SAFETY LIMITS (SLs)

Revision 0

LIST OF ATTACHMENTS

1. **ITS Chapter 2.0 – Safety Limits (SLs)**

ATTACHMENT 1

ITS Chapter 2.0 – SAFETY LIMITS (SLs)

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

~~SECTION 2.0~~

~~SAFETY LIMITS~~

~~AND~~

~~LIMITING SAFETY SYSTEM SETTINGS~~

A01

A01

ITS

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

2.1 ## 2.1 ~~SAFETY LIMITS~~ SLs

2.1.1 REACTOR CORE

2.1.1 The combination of THERMAL POWER, ~~pressurizer pressure, and the highest operating loop coolant temperature (T_{avg})~~ shall not exceed the limits specified in the COLR~~, for 3 loop operation~~; and the following S~~afety~~ L~~imits~~ shall not be exceeded:

In MODES 1 and 2,

Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure

2.1.1.1 ~~a.~~ The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-1 DNB correlation.

2.1.1.2 ~~b.~~ The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 9°F per 10,000 MWD/MTU of burnup.

2.2 SAFETY LIMIT VIOLATIONS

~~APPLICABILITY~~: MODES 1 and 2,

~~ACTION:~~

If SL 2.1.1 is violated, restore compliance and be in MODE 3

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.

A01

SL

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

be maintained ≤

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

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

~~ACTION:~~ In

If SL 2.1.2 is violated

2.2.2 ~~MODES 1 and 2~~:

In MODE 1 or 2, restore compliance and be in MODE 3

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.

In MODE 3, 4, or 5, restore compliance

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock 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:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the setpoint consistent with the Trip setpoint value within permissible calibration tolerance.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that the affected channel is OPERABLE; or

   2. 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 setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

See ITS 3.3.1

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤ 108.6% of RTP**	108.0% of RTP**
b. Low Setpoint	≤ 28.0% of RTP**	≤ 25% of RTP**
3. Intermediate Range, Neutron Flux	≤ 31.0% of RTP**	≤ 25% of RTP**
4. Source Range, Neutron Flux	≤ 1.4 X 10^5 cps	≤ 10^5 cps
5. Overtemperature ΔT	See Note 2	See Note 1
6. Overpower ΔT	See Note 4	See Note 3
7. Pressurizer Pressure-Low	≥ 1817 psig	≥ 1835 psig
8. Pressurizer Pressure-High	≤ 2403 psig	≤ 2385 psig
9. Pressurizer Water Level-High	≤ 92.2% of instrument span	≤ 92% of instrument span
10. Reactor Coolant Flow-Low	≥ 89.6% of loop design flow*	90% of loop design flow*
11. Steam Generator Water Level Low-Low	≥ 15.5% of narrow range instrument span	16% of narrow range instrument span

* Loop design flow = 86,900 gpm

** RTP = Rated Thermal Power

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
12. Steam/Feedwater Flow Mismatch Coincident with	Feed Flow ≤ 20.7% below rated Steam Flow	Feed Flow 20% below rated Steam Flow
Steam Generator Water Level-Low	≥ 15.5% of narrow range instrument span	16% of narrow range instrument span
13. Undervoltage – 4.16 kV Busses A and B	≥ 69% bus voltage	≥ 70% bus voltage
14. Underfrequency – Trip of Reactor Coolant Pump Breaker(s) Open	≥ 55.9 Hz	≥ 56.1 Hz
15. Turbine Trip		
a. Emergency Trip Header Pressure	≥ 901 psig	1000 psig
b. Turbine Stop Valve Closure	Fully Closed***	Fully Closed***
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	≥ 6.0×10^{-11} amps	Nominal 1×10^{-10} amps

See ITS 3.3.1

*** Limit switch is set when Turbine Stop Valves are fully closed.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

See ITS 3.3.1

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	≤ 13.0% RTP**	Nominal 10% of RTP**
2) Turbine Inlet Pressure	≤ 13.0% Turbine Power	Nominal 10% Turbine Power
c. Power Range Neutron Flux, P-8	≤ 48.0% RTP**	Nominal 45% of RTP**
d. Power Range Neutron Flux, P-10	≥ 7.0% RTP**	Nominal 10% of RTP**
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.

** RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT (Those values denoted with [*] are specified in the COLR.)

$$\Delta T \quad \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \frac{1}{(1+\tau_6 S)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

See ITS 3.3.1

Where:	ΔT	=	Measured ΔT by RTD Instrumentation
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead/Lag compensator on measured ΔT; τ_1 = [*]s, τ_2 = [*]s
	$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT; τ_3 = [*]s
	ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
	K_1	=	[*];
	K_2	=	[*]/°F;
	$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
	τ_4 , τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = [*]s, τ_5 = [*]s;
	T	=	Average temperature, °F;
	$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg}; τ_6 = [*]s
	T'	$\leq$	[*]°F (Indicated Loop T_{avg} at RATED THERMAL POWER);
	K_3	=	[*]/psi;
	P	=	Pressurizer pressure, psig;

See ITS 3.3.1

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P' ≥ [*] psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

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

(1) For $q_t - q_b$ between – [*]% and + [*]%, $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds – [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER.

NOTE 2: The Overtemperature ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.2% ΔT span for the Pressurizer Pressure channel, and 0.4% ΔT span for the f(ΔI) channel. No separate Allowable Value is provided for T_{avg} because this function is part of the ΔT value.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT (Those values denoted with [*] are specified in the COLR.)

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 S}{1+\tau_7 S} \left(\frac{1}{1+\tau_6 S}\right) T - K_6 \left[T \frac{1}{1+\tau_6 S} - T''\right] - f_2(\Delta I)\right\}$$

See ITS 3.3.1

Where: ΔT = As defined in Note 1,

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = As defined in Note 1,

$\frac{1}{1+\tau_3 S}$ = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 = [*],

K_5 ≥ [*]/°F for increasing average temperature and [*]/°F for decreasing average temperature,

$\frac{\tau_7 s}{1+\tau_7 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_7 \geq$ [*] s,

$\frac{1}{1+\tau_6 S}$ = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

See ITS 3.3.1

NOTE 3: (Continued)

K_6 = [*]/°F for T > T"

= [*] for T ≤ T",

T = As defined in Note 1,

T" ≤ [*]°F (Indicated Loop T_{avg} at RATED THERMAL POWER)

S = As defined in Note 1, and

f_2 (ΔI) = [*]

NOTE 4: The Overpower ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel. No separate Allowable Value is provided for T_{avg} because this function is part of the ΔT value.

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Turkey Point Nuclear Generating Station (PTN) Unit 3 and Unit 4, 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. 5.0, "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.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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

2.0 SAFETY LIMITS (SLs)

2.1 **2.1 SLs**

2.1.1,
2.1.1 Applicability

2.1.1 <u>Reactor Core SLs</u>

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

2.1.1.b — 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

9

2.1.2 — 2.1.2 <u>Reactor Coolant System Pressure SL</u>

2.1.2 Applicability

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

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:

2.2.2.1 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.

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

1. Changes are made (additions, deletions, and/or changes) to the Improved Standard Technical Specifications (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 Westinghouse 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.

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

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core

The 1967 proposed draft GDC 6

the minimum

BASES

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). 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. 1

(Ref. 1)

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. 1

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. 1

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 proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents 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 DNB and

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

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, 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.

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

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.

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

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.

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 that 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.

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.

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.~~ UFSAR, Sections 3.1.2 and 3.1.3. 1

2. U FSAR, Section ~~[~~7.2~~]~~. 1 2

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

1967 Proposed GDC 32, “Maximum Reactivity Worth of Control Rods” and 1967 Proposed GDC 33, ”Reactor Coolant Pressure Boundary Capability”

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~~ 1967 Proposed GDC 9, "Reactor Coolant Pressure Boundary," and ~~GDC 15~~ 1967 Proposed GDC 4, "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. 1

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).

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).

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

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.

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

a. Pressurizer power operated relief valves (PORVs),

b. Steam line relief valve,

c. Steam Dump System,

d. Reactor Control System,

e. Pressurizer Level Control System, or

f. Pressurizer spray valve.

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. 2
The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

BASES

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.

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.~~ UFSAR 3.1.2, 4.1.2, and 4.1.3 1

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~~. 1

4. 10 CFR 100.

5. U FSAR, Section 3.2 and ~~[~~7.2~~]~~. 1 2

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

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

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

2. The ISTS Bases contains bracketed information and/or values that are generic to all Westinghouse 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.

Specific No Significant Hazards Considerations (NSHCs)

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

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