ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATIONS

Marked-up Technical Specification/Technical Specification Bases Pages:

TS CHANGE, PART 1

3.5-2
3.5-10
B 3.5-26

TS CHANGE, PART 2

2.0-2
INSERT 1
3.3-18
3.3-21
3.3-22
3.3-37
3.4-1
3.4-2
B 2.0-4
INSERT A
B 3.2-13
B 3.3-17
INSERT B
B 3.3-19
INSERT C
B 3.3.24
B 3.3-25
B 3.3-28
INSERT D
B 3.3-91
B 3.4-2

MARKED-UP

TECHNICAL SPECIFICATION/TECHNICAL SPECIFICATION BASES

PAGES

PART 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7717 gallons and ≤ 8004 gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 610 psig and ≤ 660 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ ~~1900~~ 2400 ppm and ≤ ~~2100~~ 2700 ppm.	31 days AND -----NOTE------ Only required to be performed for affected accumulators -------------- Once within 6 hours after each solution volume increase of ≥ 75 gallons, that is not the result of addition from the refueling water storage tank

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	------------------NOTE------------------- Only required to be performed when ambient air temperature is < 60°F or > 105°F. --- Verify RWST borated water temperature is ≥ 60°F and ≤ 105°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 370,000 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2000 ppm and ≤ 2100 ppm. 2500 2700	7 days

BASES

APPLICABLE SAFETY ANALYSES (continued)

required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 37 seconds without offsite power.

2500

For a large break LOCA analysis, the minimum water volume limit of 370,000 gallons and the lower boron concentration limit of ~~2000~~ ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

2700

The upper limit on boron concentration of ~~2100~~ ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

(continued)

MARKED-UP

TECHNICAL SPECIFICATION/TECHNICAL SPECIFICATION BASES

PAGES

PART 2

Replace with Insert 1

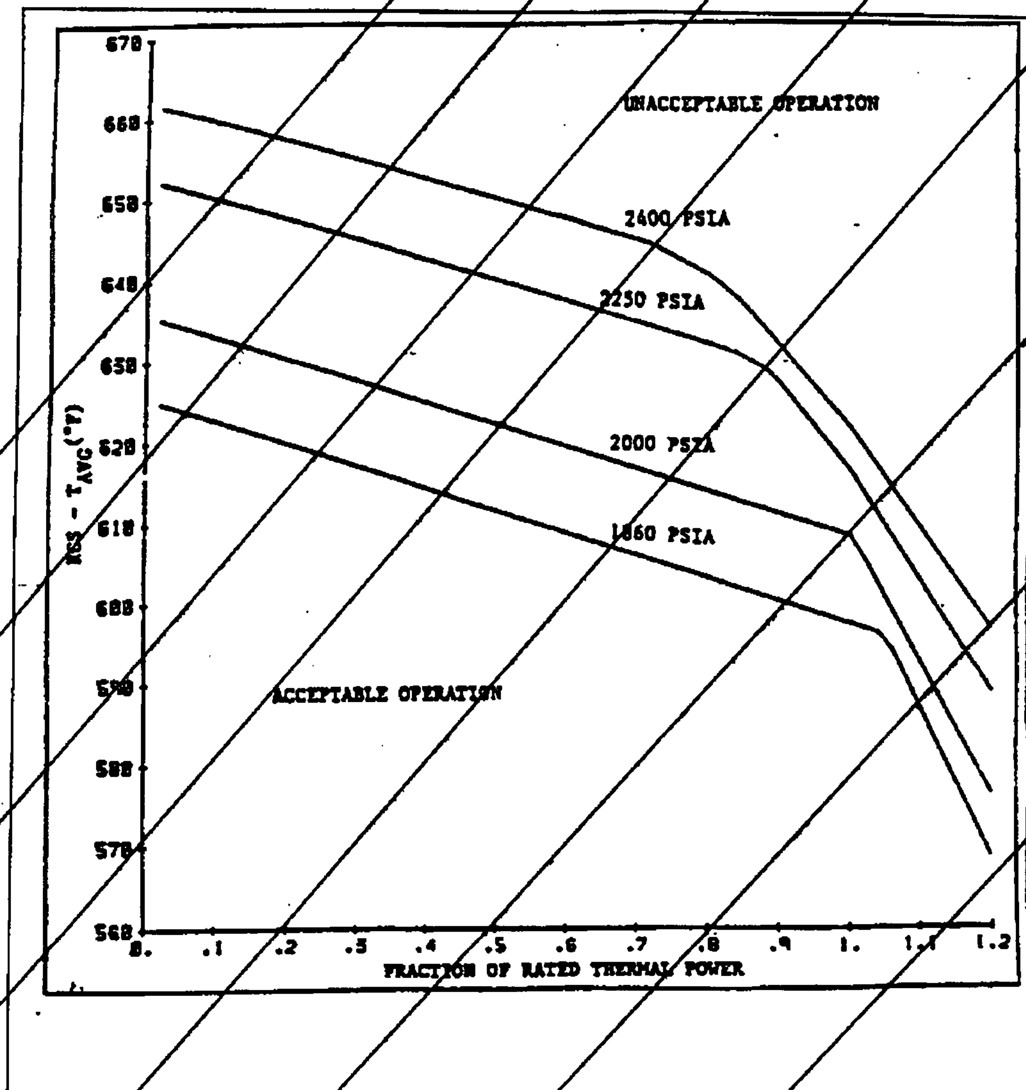

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

Insert 1 for page 2.0-2

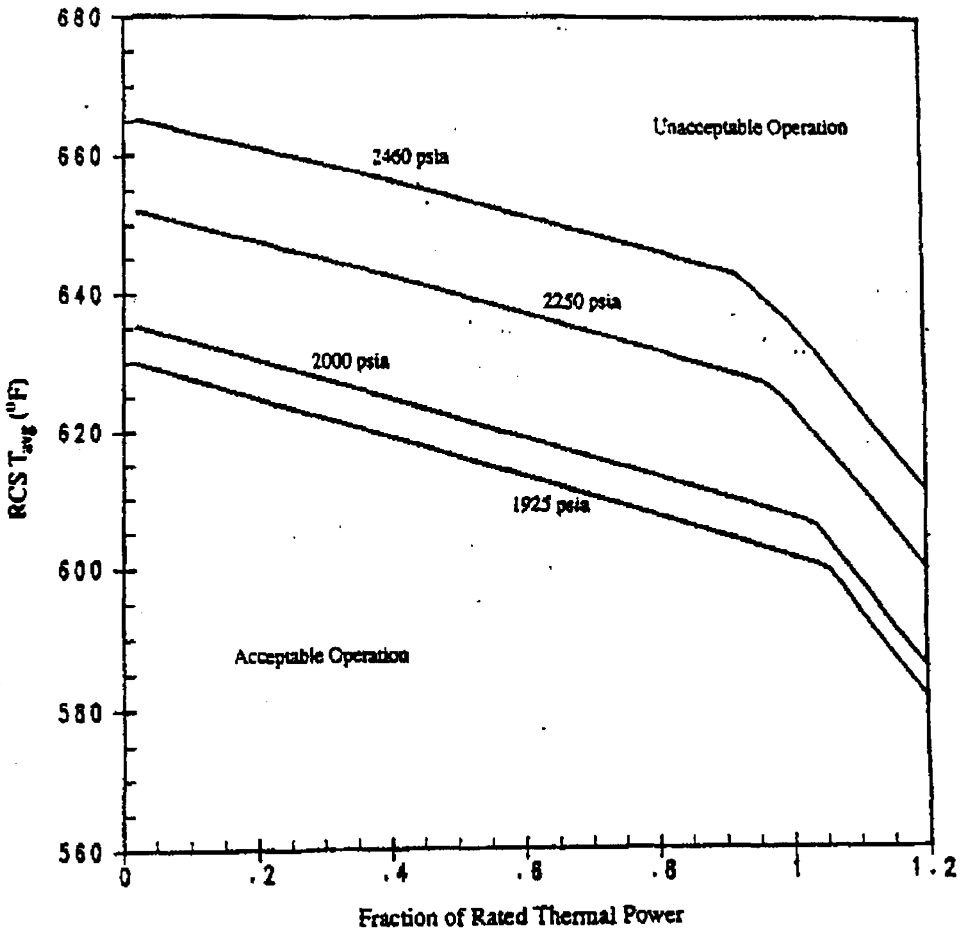

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
13.	SG Water Level--Low-Low Coincident with:	1,2	3/SG	U	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.4% of narrow range span	17% of narrow range span
	a) Vessel ΔT Equivalent to power ≤ 50% RTP	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ ~~52.7%~~ RTP 52.6%	Vessel ΔT variable input 50% RTP
	With a time delay (Ts) if one steam generator is affected					≤ 1.01 Ts (Refer to Note 3, Page 3.3-23)	Ts (Refer to Note 3, Page 3.3-23)
	or						
	A time delay (Tm) if two or more Steam Generators are affected					≤ 1.01 Tm (Refer to Note 3, Page 3.3-23)	Tm (Refer to Note 3, Page 3.3-23)
	OR						
	b) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ ~~52.7%~~ RTP 52.6 %	Vessel ΔT variable input 50% RTP
14.	Turbine Trip						
	a. Low Fluid Oil pressure	1(i)	3	O	SR 3.3.1.10 SR 3.3.1.14	≥ 43 psig	45 psig
	b. Turbine Stop Valve Closure	1(i)	4	Y	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open	1% open

(continued)

(i) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 7 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than ~~2.0% of ΔT/span.~~

1.2% ΔT SPAN

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

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the indicated T_{avg} at RTP, $\le$ 588.2°F.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, $\ge$ 2235 psig

$K_1 \le$ ~~[illegible]~~ 1.16 $\quad K_2 \ge$ ~~[illegible]~~ 0.0183 /°F $\quad K_3 =$ ~~[illegible]~~ 0.000900 /psig
$\tau_1 \ge$ 33 sec $\quad \tau_2 \le$ 4 sec
$\tau_4 \ge$ ~~12~~ 3 sec $\quad \tau_5 \le$ 3 sec

$f_1(\Delta I)$ = ~~[illegible]~~ -2.62 {~~32~~ 22 + $(q_t - q_b)$} when $q_t - q_b <$ ~~[illegible]~~ -22% RTP
0 when ~~-32%~~ RTP $\le q_t - q_b \le$ 10% RTP
~~[illegible]~~ 1.96 $((q_t - q_b) - 10)$ when $q_t - q_b >$ 10% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 8 of 9)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than ~~1.8% of ΔT span.~~

1.0% ΔT SPAN

$$\Delta T\left(\frac{1+\tau_4 s}{1+\tau_5 s}\right) \le \Delta T_0\left\{K_4 - K_5\left(\frac{\tau_3 s}{1+\tau_3 s}\right)T - K_6\left[T - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T'' is the indicated T_{avg} at RTP, ≤ 588.2°F.

1.10

$K_4 \le \cancel{1.091}$

$K_5 \ge 0.02$/°F for increasing T_{avg}
0/°F for decreasing T_{avg}

0.00162

$K_6 \ge \cancel{0.00128}$/°F when $T > T''$
0/°F when $T \le T''$

$\tau_3 \ge 5$ sec

$\tau_4 \ge \cancel{10}$ sec

3

$\tau_5 \le 3$ sec

$f_2(\Delta I) = 0$ for all ΔI.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low	1,2,3	3 per SG	N	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 16.4%	17.0%
Coincident with:						
1) Vessel ΔT equivalent to power ≤ 50% RTP	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ ~~[illegible]%~~ 52.6% RTP	Vessel ΔT variable input 50% RTP
With a time delay (Ts) if one S/G is affected					≤ 1.01 Ts (Note 1, Page 3.3-40)	Ts (Note 1, Page 3.3-40)
or						
A time delay (Tm) if two or more S/G's are affected					≤ 1.01 Tm (Note 1, Page 3.3-40)	Tm (Note 1, (Page 3.3-40)
OR						
2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ ~~[illegible]%~~ 52.6% RTP	Vessel ΔT variable input 50% RTP

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

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

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

a. Pressurizer pressure ≥ 2214 psig;

593.2°F

b. RCS average temperature ≤ ~~[illegible]~~ and

380,000

c. RCS total flow rate ≥ 397,000 gpm (process computer) or ~~[illegible]~~ control board indication).

APPLICABILITY: MODE 1.

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

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

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ 2214 psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ≤ ~~588.2°F~~. 593.2°F	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 380,000 ~~397,000~~ gpm (process computer~~)~~ or ~~398,000 gpm (~~control board indication).	12 hours
SR 3.4.1.4	-------------------NOTE-------------------- Required to be performed within 24 hours after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ ~~397,000~~ gpm. 380,000	18 months

Replace with Insert A

BASES

SAFETY LIMITS (continued)

~~SL 2.1.1 reflects the use of the WRB-1 CHF correlation DNBR limit of 1.17 and a safety analysis DNBR limit of 1.31. Comparison of these DNBR limits results in a 10.7% DNBR margin which is more than sufficient to offset the DNBR penalty due to rod bow.~~

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.

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.

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

(continued)

INSERT A FOR PAGE B 2.0-4

To meet the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. The effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion. SL 2.1.1 reflects the use of the WRB-1 CHF correlation with design limit DNBR values of 1.25 and 1.24 for the typical and thimble cell, respectively.

An additional 10% DNBR margin is maintained by performing the safety analyses to a higher DNBR limit of 1.39 and 1.38 for the typical and thimble cell, respectively. This margin between the design and safety analysis limit is more than sufficient to offset known DNBR penalties (e.g., rod bow) and to provide DNBR margin for operating and design flexibility.

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Refer to the Bases for the Reactor Core Safety Limits, B2.1.1, for a discussion of the applicable DNBR limits.

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

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

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. ~~In the heated rod spans above the first mixing vane grid, the WRB-1 correlation with a DNBR limit of 1.17 is applied~~. The W-3 Correlation with a DNBR limit of 1.3 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

- axial power distribution - the f(ΔI) Overtemperature ΔT Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for delays associated with fluid transport from the core to the loop temperature detectors (RTDs), and thermowell and RTD response time delays.

Insert - B

~~Delta-T_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured by the plant for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT value. Accurate determination of the loop specific ΔT value should be made when performing incore/excore quarterly recalibration and under steady state conditions (i.e., power distribution not affected by xenon transient conditions).~~

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. The pressure and temperature signals are used for other control functions. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels

(continued)

OVERTEMPERATURE

INSERT B FOR PAGE B 3.3-17

ΔT_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured for each loop. T′ represents the 100% RTP Tavg value as measured by the plant for each loop. ΔT_0 and T′ normalize each loop's ΔT setpoint to the actual operating conditions existing at the time of measurement, thus forcing the setpoint to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT and Tavg can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT and Tavg values. Loop specific values of ΔT_0 and T′ must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and adjusted, if required. Tolerances for ΔT_0 and T′ have been included in the determination of the Overtemperature ΔT setpoint.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature – the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature – including dynamic compensation for delays associated with fluid transport from the core to the loop temperature detectors (RTDs), and thermowell and RTD response time delays.

Insert C

~~Delta-T_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured by the plant for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT value. Accurate determination of the loop specific ΔT value should be made when performing incore/excore quarterly recalibration and under steady state conditions (i.e., power distribution not affected by xenon transient conditions).~~

(continued)

OVERPOWER

INSERT C FOR PAGE B 3.3-19

ΔT_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured for each loop. T″ represents the 100% RTP Tavg value as measured for each loop. ΔT_0 and T″ normalize each loop's ΔT setpoint to the actual operating conditions existing at the time of measurement, thus forcing the setpoint to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT and Tavg can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT and Tavg values. Loop specific values of ΔT_0 and T″ must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and adjusted, if required. Tolerances for ΔT_0 and T″ have been included in the determination of the Overpower ΔT setpoint.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Coolant Flow-Low (Single Loop) (continued)

the core. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip (Function 10.b) because of the lower power level and the greater margin to the design limit DNBR.

thermal design flow adjusted for uncertainties (95,000 gpm)

The Reactor Coolant Flow-Low ~~Trip~~ Setpoint and Allowable Value are specified in ~~% nominal~~ flow, however, the Eagle-21™ values entered through the MMI are specified in an equivalent % differential pressure.

b. Reactor Coolant Flow-Low (Two Loops)

The Reactor Coolant Flow-Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow-Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Reactor Coolant Flow-Low (Two Loops) (continued)

The Reactor Coolant Flow-Low Trip Setpoint and Allowable Value are specified in % ~~nominal~~ flow, however, the Eagle-21™ values entered through the MMI are specified in an equivalent % differential pressure.

thermal design flow adjusted for uncertainties (95,000 gpm)

11. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached. The loss of voltage in two loops must be sustained for a length of time equal to or greater than that set in the time delay. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires one Undervoltage RCP channel per bus to be OPERABLE.

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

12. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Steam Generator Water Level-Low Low (continued)

(T_M) for the affected protection set, through the Man-Machine Interface. Failure of the vessel ΔT channel input (failure of more than one T_H RTD or failure of both T_C RTDs) affects the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

The LCO requires three channels of SG Water Level – Low Low per SG to be OPERABLE. This function initiates a reactor trip and the ESFAS function auxiliary feedwater pump start. The reactor trip feature is required to be OPERABLE in MODES 1 and 2 and the auxiliary feedwater pump start feature is required to be OPERABLE in MODES 1, 2, and 3.

In MODE 3, OPERABILITY of loop ΔT input to TTD is not required because MODE 3 $\Delta T = 0$ (by definition). The Eagle-21™ code does not allow anything less than 0. The value of ΔT is low-limited to 0.0 prior to use in the calculation of the single and multiple trip time delays.

Insert D

for performing a channel check on

For MODES 1, 2, and 3, channel check surveillance testing on RCS loop ΔT ~~input to TTD~~ is not required. There are no provisions made to verify the RCS loop ΔT for the SG Level TTD Function. ~~The power level can only be verified by connecting the Eagle-21™ Man-Machine Interface terminal and viewing the Dynamic Information for this channel.~~ The Eagle-21™ system uses a redundant sensor algorithm for the hot leg and cold leg inputs, and will alert the operator if a failure occurs with the sensor or input signal conditioning.

The coefficients (A, B, C, D, E, F, G, and H) shown in the equation of Note 3 represent conservative values for the calculation of the time delay (i.e., the values given are 99% of the values used for the safety analyses). For the Eagle-21™ System, these coefficients are displayed (via the Man-Machine Interface) as A, B, C and D for the single request time delay, and E, F, G and H for the multiple request time delay.

(continued)

STEAM GENERATOR WATER LEVEL - LOW, LOW
INSERT D FOR PAGE B 3.3-28

For MODES 1 and 2, ΔT_0, as used in the Vessel ΔT Equivalent to Power represents the 100% RTP value as measured for each loop. ΔT_0 normalizes each loop's vessel ΔT to the actual operating conditions existing at the time of measurement, thus forcing the TTD to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Loop specific values of ΔT_0 must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and adjusted, if required. Tolerances for ΔT_0 have been included in the determination of the Vessel ΔT Equivalent to Power.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Auxiliary Feedwater-Steam Generator Water Level-Low Low (continued)

The Steam Generator Water Level Channel Trip Time Delay (TTD) creates additional operational margin when the unit needs it most, during power escalation from low power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of vessel ΔT. Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Level channel Trip Setpoint per Note 1 of Table 3.3.2-1. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low Level channel trips.

The algorithm for the TTD, T_s and T_m, determines the trip delay as a function of power level (P) and four constants (A...D for T_s, E...H for T_m). An allowance for the accuracy of the Eagle-21™ time base is included in the determination of the magnitude of the constants. The magnitude of the accuracy allowance is 1%, i.e., the constant values were multiplied by 0.99 to account for this potential error.

Refer to the Bases for the Steam Generator Water Level Low-Low Reactor Trip, B 3.3.1, for a discussion of the required MODES and Normalization of the Vessel ΔT input to the TTD.

In the event of a failure of a Steam Generator Water Level channel, the channel is placed in the trip condition as input to the Solid State Protection System and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation (T_S) to match the multiple steam generator time delay calculation (T_m) for the affected protection set, through the Man Machine Interface. Failure of the vessel ΔT channel input (failure of more than one T_H RTD or failure of both T_C RTDs) affects the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2214 psig and the RCS average temperature limit of ~~[illegible]~~ 593.2°F correspond to analytical limits of ~~2189~~ 2185 psig and ~~[illegible]~~ 594.2°F used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error of 1.6% (process computer) or ~~1.9%~~ 1.8% (control board indication) based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to 1.7% (process computer) or ~~2.0%~~ 1.9% (control board indication).

Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

(continued)

REVISED

TECHNICAL SPECIFICATION/TECHNICAL SPECIFICATION BASES

PAGES

PART 1

Revised Technical Specification/Technical Specifications Bases pages:

TS CHANGE, PART 1

3.5-2
3.5-10
B 3.5-26

TS CHANGE, PART 2

2.0-2
3.3-18
3.3-21
3.3-22
3.3-37
3.4-1
3.4-2
B 2.0-4
B 3.2-13
B 3.3-17
B 3.3-19
B 3.3.24
B 3.3-25
B 3.3-28
B 3.3-91
B 3.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7717 gallons and ≤ 8004 gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 610 psig and ≤ 660 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2400 ppm and ≤ 2700 ppm.	31 days <u>AND</u> -----NOTE------ Only required to be performed for affected accumulators --------------- Once within 6 hours after each solution volume increase of ≥ 75 gallons, that is not the result of addition from the refueling water storage tank

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	------------------NOTE------------------- Only required to be performed when ambient air temperature is < 60°F or > 105°F. -- Verify RWST borated water temperature is ≥ 60°F and ≤ 105°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 370,000 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2500 ppm and ≤ 2700 ppm.	7 days

APPLICABLE SAFETY ANALYSES (continued)

required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 37 seconds without offsite power.

For a large break LOCA analysis, the minimum water volume limit of 370,000 gallons and the lower boron concentration limit of 2500 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2700 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

(continued)

REVISED

TECHNICAL SPECIFICATION/TECHNICAL SPECIFICATION BASES

PAGES

PART 2

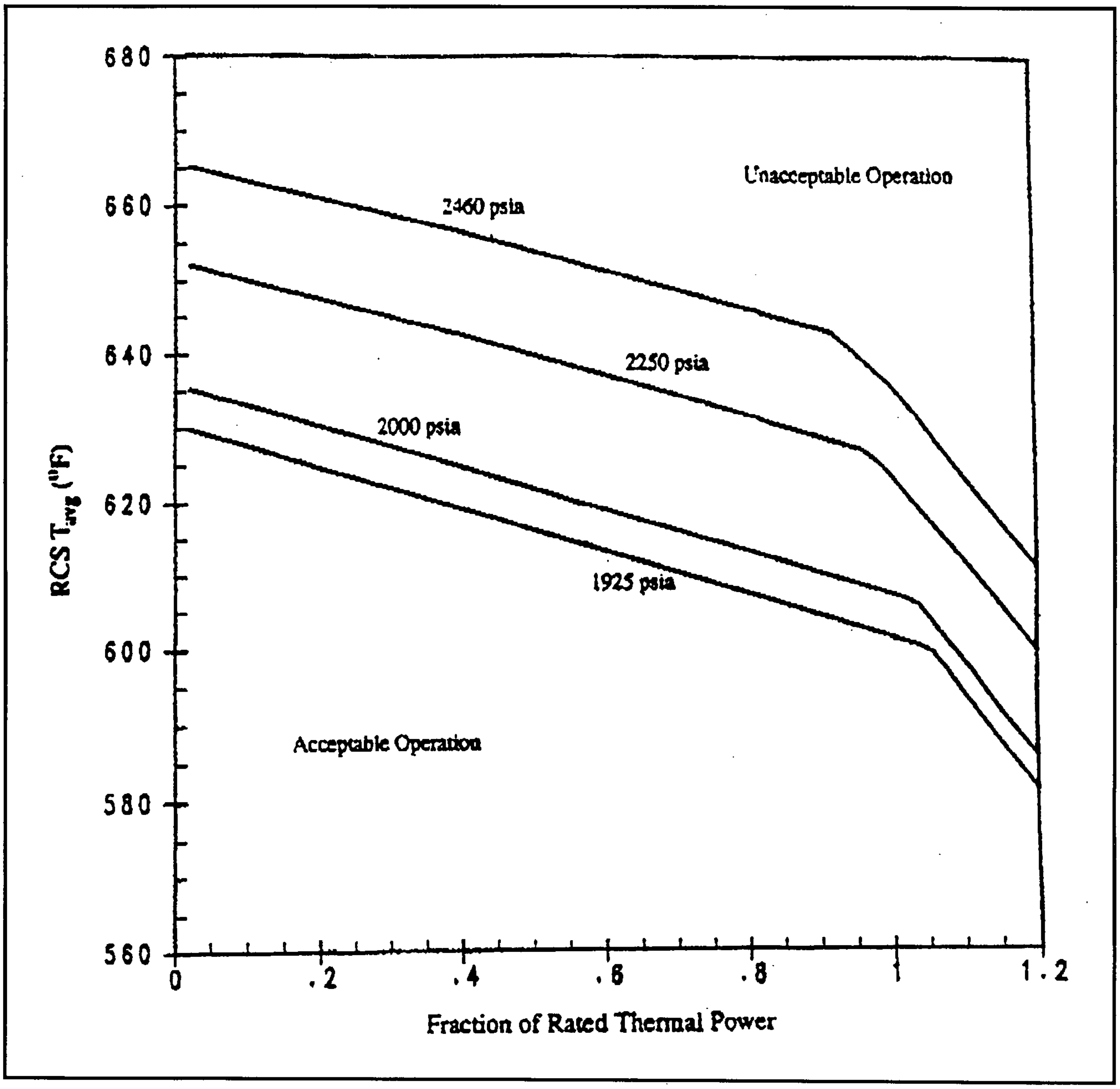

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

BASES

SAFETY LIMITS (continued)

To meet the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. The effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion. SL 2.1.1 reflects the use of the WRB-1 CHF correlation with design limit DNBR values of 1.25 and 1.24 for the typical and thimble cell, respectively.

Additional 10% DNBR margin is maintained by performing the safety analyses to a higher DNBR limit of 1.39 and 1.38 for the typical and thimble cell, respectively. This margin between the design and safety analysis limit is more than sufficient to offset know DNBR penalties (e.g., rod bow) and to provide the DNBR margin for operating and design flexibility.

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.

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.

BASES

VIOLATIONS (continued)

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.

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

2.2.4

If SL 2.1.1 is violated, the Plant Manager, Site Vice President, and Nuclear Safety Review Board (NSRB) shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.5

If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Reference 6). A copy of the report shall also be provided to the Plant Manager, Site Vice President, and NSRB.

SAFETY LIMIT VIOLATIONS (continued)

2.2.6

If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. Watts Bar FSAR, Section 7.2, "Reactor Trip System."

3. WCAP-8746-A, "Design Bases for the Overtemperature DT and the Overpower DT Trips," March 1977.

BASES

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

5. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

6. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

7. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984.

8. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.

9. Tong, L. S., "Critical Heat Fluxes on Rod Bundles," in "Two-Phase Flow and Heat Transfer in Rod Bundles," pages 31 through 41, American Society of Mechanical Engineers, New York, 1969.

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
13. SG Water Level--Low-low Coincident with:	1,2	3/SG	U	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.4% of narrow range span	17% of narrow range span
a) Vessel ΔT Equivalent to power ≤ 50% RTP	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input 50% RTP
With a time delay (Ts) if one steam generator is affected					≤ 1.01 Ts (Refer to Note 3, Page 3.3-23)	Ts (Refer to Note 3, Page 3.3-23)
or A time delay (Tm) if two or more Steam Generators are affected OR					≤ 1.01 Tm (Refer to Note 3, Page 3.3-23)	Tm (Refer to Note 3, Page 3.3-23)
b) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input 50% RTP
14. Turbine Trip						
a. Low Fluid Oil pressure	1(i)	3	O	SR 3.3.1.10 SR 3.3.1.14	³ 43 psig	45 psig
b. Turbine Stop Valve Closure	1(i)	4	Y	SR 3.3.1.10 SR 3.3.1.14	³ 1% open	1% open

(continued)

(i) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 7 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.2% of ΔT span.

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

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the indicated T_{avg} at RTP, ≤ 588.2°F.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, ≥ 2235 psig

$K_1 \leq 1.16$	$K_2 \geq 0.0183/°F$	$K_3 = 0.000900/psig$
$\tau_1 \geq 33$ sec	$\tau_2 \leq 4$ sec	
$\tau_4 \geq 3$ sec	$\tau_5 \leq 3$ sec	

$f_1(\Delta I) =$	$-2.62\{22 + (q_t - q_b)\}$	when $q_t - q_b < -22\%$ RTP
	0	when -22% RTP $\leq q_t - q_b \leq 10\%$ RTP
	$1.96\{(q_t - q_b) - 10\}$	when $q_t - q_b > 10\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 8 of 9)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.0% of ΔT span.

$$\Delta T \left(\frac{1+\tau_4 s}{1+\tau_5 s} \right) \leq \Delta T_0 \left\{ K_4 - K_5 \left(\frac{\tau_3 s}{1+\tau_3 s} \right) T - K_6 \left[T - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T'' is the indicated T_{avg} at RTP, ≤ 588.2°F.

$K_4 \leq 1.10$	$K_5 \geq 0.02/°F$ for increasing T_{avg} 0/°F for decreasing T_{avg}	$K_6 \geq 0.00162/°F$ when $T > T''$ 0/°F when $T \leq T''$
$\tau_3 \geq 5$ sec	$\tau_4 \geq 3$ sec	$\tau_5 \leq 3$ sec

$f_2(\Delta I) = 0$ for all ΔI.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low Coincident with:	1,2,3	3 per SG	M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	³ 16.4%	17.0%
1) Vessel ΔT equivalent to power ≤ 50% RTP	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input £ 52.6% RTP	Vessel ΔT variable input 50% RTP
With a time delay (Ts) if one S/G is affected or					≤ 1.01 Ts (Note 1, Page 3.3-40)	Ts (Note 1, Page 3.3-40)
A time delay (Tm) if two or more S/G's are affected OR					≤ 1.01 Tm (Note 1, Page 3.3-40)	Tm (Note 1, (Page 3.3-40)
2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.6% RTP	Vessel ΔT variable input 50% RTP

(continued)

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

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

a. Pressurizer pressure ≥ 2214 psig;

b. RCS average temperature ≤ 593.2°F; and

c. RCS total flow rate ≥ 380,000 gpm (process computer or control board indication).

APPLICABILITY: MODE 1.

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

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

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ 2214 psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ≤ 593.2°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 380,000 gpm (process computer or control board indication).	12 hours
SR 3.4.1.4	-------------------NOTE-------------------- Required to be performed within 24 hours after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 380,000 gpm.	18 months

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

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

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1, for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

- axial power distribution - the f(ΔI) Overtemperature ΔT Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for delays associated with fluid transport from the core to the loop temperature detectors (RTDs), and thermowell and RTD response time delays.

ΔT_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured by the plant for each loop. T' represents the 100% RTP Tavg value as measured by the plant for each loop. ΔT_0 and T' normalize each loop's ΔT setpoint to the actual operating conditions existing at the time of measurement, thus forcing the setpoint to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT and Tavg can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow , and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT and Tavg values. Loop specific values of ΔT_0 and T' must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and updated, if required. Tolerances for ΔT_0 and T' have been included in the determination of the Overtemperature ΔT setpoint.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. The pressure and temperature signals are used for other control functions. The actuation logic must be able to withstand

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux–High Setpoint trip.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. <u>Overpower ΔT</u> (continued)

The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and

- rate of change of reactor coolant average temperature —including dynamic compensation for delays associated with fluid transport from the core to the loop temperature detectors (RTDs), and thermowell and RTD response time delays.

ΔT_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured by the plant for each loop. T" represents the 100% RTP Tavg value as measured by the plant for each loop. ΔT_0 and T" normalize each loop's ΔT setpoint to the actual operating conditions existing at the time of measurement, thus forcing the setpoint to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT and Tavg can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow , and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT and Tavg values. Loop specific values of ΔT_0 and T" must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and updated, if required. Tolerances for ΔT_0 and T" have been included in the determination of the Overtemperature ΔT setpoint.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Coolant Flow-Low (Single Loop) (continued)

the core. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip (Function 10.b) because of the lower power level and the greater margin to the design limit DNBR.

The Reactor Coolant Flow-Low Trip Setpoint and Allowable Value are specified in % thermal design flow adjusted for uncertainties (95,000 gpm), however, the Eagle-21™ values entered through the MMI are specified in an equivalent % differential pressure.

b. Reactor Coolant Flow-Low (Two Loops)

The Reactor Coolant Flow–Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow–Low channels per loop to be OPERABLE.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow–Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Reactor Coolant Flow-Low (Two Loops) (continued)

The Reactor Coolant Flow-Low Trip Setpoint and Allowable Value are specified in % thermal design flow adjusted for uncertainties (95,000 gpm), however, the Eagle-21™ values entered through the MMI are specified in an equivalent % differential pressure.

11. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. The loss of voltage in two loops must be sustained for a length of time equal to or greater than that set in the time delay. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires one Undervoltage RCP channel per bus to be OPERABLE.

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

12. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

12. Underfrequency Reactor Coolant Pumps (continued)

after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires one Underfrequency RCP channel per bus to be OPERABLE.

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

13. Steam Generator Water Level-Low Low

Loss of the steam generator as a heat sink can be caused by the loss of normal feedwater, a station blackout or a feedline rupture. Feedline ruptures inside containment are protected by the containment high pressure trip Function, based on a 1994 TVA analysis (Ref. 3). Feedline ruptures outside containment and the other causes of the heat sink loss are protected by the SG Water Level Low-Low trip Function.

The SG Water Level—Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Control/protection interaction is addressed by the use of a Median Signal Selector which prevents a single failure of a channel providing input to the control system

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Steam Generator Water Level-Low Low (continued)

from initiating a condition requiring protection function action. The Median Signal Selector performs this by not selecting the channels indicating the highest or lowest steam generator levels as input to the control system.

Because one failed protection instrument channel would not result in an adverse control system action, a second random protection system failure (as otherwise required by IEEE 279-1971) need not be considered.

The Steam Generator Water Level Trip Time Delay (TTD) creates additional operational margin when the plant needs it most, during escalation to power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of vessel DT. Two time delays are calculated based on the number of steam generators indicating less than the Low-Low Trip Setpoint per Note 3 of Table 3.3.1-1. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low Level channel trips.

The algorithm for the TTD, T_s and T_m, determines the trip delay as a function of power level (P) and four constants (A...D for T_s, E...H for T_m). An allowance for the accuracy of the Eagle-21™ time base is included in the determination of the magnitude of the constants. The magnitude of the accuracy allowance is 1%, i.e., the constant values were multiplied by 0.99 to account for this potential error.

In the event of failure of a Steam Generator Water Level Channel, the channel is placed in the trip condition as input to the Solid State Protection System (SSPS) and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation (T_S) to match the multiple steam generator time delay calculation (T_M) for the affected protection set, through the Man-Machine Interface. Failure of the vessel ΔT channel input (failure of more than one T_H RTD or failure of both T_C RTDs) affects the TTD calculation for a

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Steam Generator Water Level-Low Low (continued)

protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

The LCO requires three channels of SG Water Level–Low Low per SG to be OPERABLE. This function initiates a reactor trip and the ESFAS function auxiliary feedwater pump start. The reactor trip feature is required to be OPERABLE in MODES 1 and 2 and the auxiliary feedwater pump start feature is required to be OPERABLE in MODES 1, 2, and 3.

In MODE 3, OPERABILITY of loop DT input to TTD is not required because MODE 3 DT = 0 (by definition). The Eagle-21™ code does not allow anything less than 0.The value of DT is low-limited to 0.0 prior to use in the calculation of the single and multiple trip time delays.

For MODES 1 and 2, ΔT_0, as used in the Vessel ΔT Equivalent to Power represents the 100% RTP value as measured by the plant for each loop. ΔT_0 normalizes each loop's vessel ΔT to the actual operating conditions existing at the time of measurement, thus forcing the TTD to reflect the equivalent full power conditions as assumed in the accident analyses. Differences in RCS loop ΔT can be due to several factors, e.g., measured RCS loop flow greater than minimum measured flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Loop specific values of ΔT_0 must be determined at the beginning of each fuel cycle at full power, steady-state conditions (i.e., power distribution not affected by xenon transient conditions) and will be checked quarterly and updated, if required. Tolerances for ΔT_0 have been included in the determination of the Vessel ΔT Equivalent to Power.

For MODES 1, 2, and 3, channel check surveillance testing on RCS loop DT input to TTD is not required. There are no provisions for performing a channel check on the RCS loop DT for the SG Level TTD Function. The power level can only

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Steam Generator Water Level-Low Low (continued)

be verified by connecting the Eagle-21™ Man-Machine Interface terminal and viewing the Dynamic Information for this channel. The Eagle-21™ system uses a redundant sensor algorithm for the hot leg and cold leg inputs, and will alert the operator if a failure occurs with the sensor or input signal conditioning.

The coefficients (A, B, C, D, E, F, G, and H) shown in the equation of Note 3 represent conservative values for the calculation of the time delay (i.e., the values given are 99% of the values used for the safety analyses). For the Eagle-21™ System, these coefficients are displayed (via the Man-Machine Interface) as A, B, C and D for the single request time delay, and E, F, G and H for the multiple request time delay.

In MODE 1 or 2, when the reactor is critical, the SG Water Level—Low Low trip must be OPERABLE. In MODES 1, 2, and 3 the normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor in these MODES. The ESFAS Function of the SG Water Level-Low Low trip must be OPERABLE in MODES 1, 2, and 3. In MODES 3, 4, 5, and 6, the SG Water Level—Low Low trip Function does not have to be OPERABLE because the reactor is not operating or even critical.

14. Turbine Trip

a. Turbine Trip-Low Fluid Oil Pressure

The Turbine Trip—Low Fluid Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. Three pressure switches monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by

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APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

a. Turbine Trip-Low Fluid Oil Pressure (continued)

two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure–High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip–Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9. Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip–Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip–Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level below the P-9 setpoint, approximately 50% power. This action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure–High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip–Low Fluid Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Turbine Trip-Turbine Stop Valve Closure (continued)

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

The LCO requires four Turbine Trip–Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip–Stop Valve Closure trip Function does not need to be OPERABLE.

15. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. Reactor trip is not credited in the large break LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI Input is provided by solid state logic in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

16. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip Functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel indicates approximately one decade above the minimum channel reading. If both channels decrease below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to increasing power above the source range; and

- on decreasing power, the P-6 interlock automatically enables the NIS Source Range Neutron Flux reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip may be blocked, and this Function would no longer be necessary. In MODE 3, 4, 5, or 6, the P-6 interlock is not required to be OPERABLE because the NIS Source Range is providing core protection.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Auxiliary Feedwater-Steam Generator Water Level-Low Low (continued)

The Steam Generator Water Level Channel Trip Time Delay (TTD) creates additional operational margin when the unit needs it most, during power escalation from low power, by allowing the operator time to recover level when the primary side load is sufficiently small to allow such action. The TTD is based on continuous monitoring of primary side power through the use of vessel DT. Two time delays are calculated, based on the number of steam generators indicating less than the Low-Low Level channel Trip Setpoint per Note 1 of Table 3.3.2-1. The magnitude of the delays decreases with increasing primary side power level, up to 50% RTP. Above 50% RTP there are no time delays for the Low-Low Level channel trips.

The algorithm for the TTD, T_s and T_m, determines the trip delay as a function of power level (P) and four constants (A...D for T_s, E...H for T_m). An allowance for the accuracy of the Eagle-21™ time base is included in the determination of the magnitude of the constants. The magnitude of the accuracy allowance is 1%, i.e., the constant values were multiplied by 0.99 to account for this potential error.

In the event of a failure of a Steam Generator Water Level channel, the channel is placed in the trip condition as input to the Solid State Protection System and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation (T_S) to match the multiple steam generator time delay calculation (T_m) for the affected protection set, through the Man Machine Interface. Failure of the vessel DT channel input (failure of more than one T_H RTD or failure of both T_C RTDs) affects the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

Refer to the Bases for the Steam Generator Water Level

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

c. Auxiliary Feedwater-Safety Injection

Low-Low Reactor Trip, B 3.3.1, for a discussion of the required MODES and normalization of the vessel ΔT input to the TTD.

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

d. Auxiliary Feedwater-Loss of Offsite Power

A loss of offsite power to the RCP buses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each 6.9 kV shutdown board. Loss of power to either 6.9 kV shutdown board will start the turbine driven AFW pump to ensure that enough water is available to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

Functions 6.a through 6.d (except the loop DT input to the trip time delay) must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level—Low Low in any operating SG will cause the motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level—Low Low in any two operating SGs will cause the turbine driven pumps to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

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APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2214 psig and the RCS average temperature limit of 593.2°F correspond to analytical limits of 2185 psig and 594.2°F used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error of 1.6% (process computer) or 1.8% (control board indication) based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to 1.7% (process computer) or 1.9% (control board indication).

Any fouling that might bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

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ENCLOSURE 4

ACCIDENT ANALYSIS DATA

FOR SELECTED

NON-LOCA EVENTS

The table in this enclosure refers to specific documents which are defined as follows:

SECL - Safety Evaluation Checklist - This data represents DNB values consistent with the analyses supporting TS Change 96-013.

AAPC - Accident Analysis Parameters Checklist - This data represents DNB values consistent with the previous analyses of record including any changes prior to TS Change 96-013.

The parameter values provided in this enclosure may be subsequently adjusted by the design process using 10 CFR 50.59 (i.e., by shifting of design margins between parameters). However, the specific limits for each parameter will not be exceeded.

PART B

NON-PROPRIETARY VERSION

ACCIDENT ANALYSIS DATA FOR SELECTED NON-LOCA EVENTS[1]

a,c

Event	DNB Reported in 2/13/96 AAPC		DNB in SECL-97-003	
	DNB Limit	DNB Value	DNB Limit	DNB Value
Uncontrolled RCCA Bank Withdrawal at Power				
Uncontrolled from subcritical (First Span) RCCA Bank Withdrawal				
Uncontrolled RCCA Bank Withdrawal from Subcritical (Remaining Span)				
Partial Loss of RCS Flow				
Complete Loss of RCS Flow (underfrequency)				
Complete Loss of RCS Flow (undervoltage)				
Loss of External Load/Turbine Trip				
Excessive Heat Removal from FW System Malfunction				
Excessive Load Increase Incident				
Accidental Depressurization of RCS				
Accidental Depressurization of Main Steam System and Major Rupture of a Main Steam System Pipe				
Inadvertent ECCS Operation				
Reactor Coolant Pump Locked Rotor	See Note 2			

Notes:

1) The DNB related analyses discussed in SECL-97-003 use a different methodology and correlation than the analyses summarized in the AAPCs. Thus, the "DNB Margin" for the analyses discussed in SECL-97-003 are not directly comparable to the "DNB Margin" for the analyses summarized in the AAPCs.

2) The comparison for the locked rotor event is presented as follows:

	SECL	AAPC	LIMIT
Max RCS pressure (psia)			
Max clad temperature (°F)			
Max $Zr\text{-}H_20$ reaction (%)			

a,c