2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained within the 95/95 DNB criterion correlation specified in the COLR.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU.

2.1.2 RCS Pressure SL

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

2.2 SL Violations

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

2.2.2 If SL 2.1.2 is violated:

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

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

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.4% of ΔT span.

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

Where: ΔT is measured loop ΔT, °F.
ΔT_O is the indicated loop ΔT at RTP and reference T_{avg}, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured loop average temperature, °F.
T' is the reference T_{avg} at RTP, $\leq$ * °F.

P is the measured pressurizer pressure, psig.
P' is the nominal pressurizer operating pressure = * psig.

K_1 = *	K_2 = * /°F	K_3 = * /psi
$\tau_1 \geq$ * sec	$\tau_2 \leq$ * sec	
τ_4 = * sec	$\tau_5 \leq$ * sec	$\tau_6 \leq$ * sec

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

$f_1(\Delta I)$ =	*{* + (q_t - q_b)}	when (q_t - q_b) $\leq$ * % RTP
	* % of RTP	when * % RTP < (q_t - q_b) $\leq$ * % RTP
	*{(q_t - q_b) - *}	when (q_t - q_b) > * % 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.

* as specified in the COLR

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

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.4% of ΔT span.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T_O \left\{ K_4 - K_5 \frac{\tau_3 s}{1+\tau_3 s} \left(\frac{1}{1+\tau_6 s} \right) T - K_6 \left[T \frac{1}{1+\tau_6 s} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured loop ΔT, °F.
ΔT_O is the indicated loop ΔT at RTP and reference T_{avg}, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured loop average temperature, °F.
T" is the reference T_{avg} at RTP, ≤ * °F.

K_4 = *

K_5 = */°F for increasing T_{avg}
K_5 = */°F for decreasing T_{avg}

K_6 = */°F when T > T"
K_6 = */°F when T ≤ T"

$\tau_3 \geq$ * sec

$\tau_4 =$ * sec

$\tau_5 \leq$ * sec

$\tau_6 \leq$ * sec

$f_2(\Delta I)$ = *% RTP for all ΔI.

* as specified in the COLR

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 in the COLR. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method, ≥ 264,200 GPM when using the elbow tap method, and ≥ the limit specified in the COLR.

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 within the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within the limits.	12 hours
SR 3.4.1.4	---------------------------NOTE--------------------- Not required to be performed until 7 days after ≥ 90% RTP. --- Verify by measurement that RCS total flow rate is within the limits.	18 months

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, the monthly operating report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient, and any corrective action necessary to prevent recurrence.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Reactor Core Safety Limits for THERMAL POWER, Reactor Coolant System highest loop average temperature and pressurizer pressure for Safety Limit 2.1.1,

2. SHUTDOWN MARGIN limit for MODES 2 (with $k_{eff} < 1$), 3, 4, and 5 for LCO 3.1.1,

3. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm and 100 ppm surveillance limits for LCO 3.1.3,

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. Shutdown Bank Insertion Limits for LCO 3.1.5,

5. Control Bank Insertion Limit for LCO 3.1.6,

6. Heat Flux Hot Channel Factor F_Q^{RTP} limits, K(Z) figure, W(Z) values, and $F_Q(Z)$ Penalty Factors for LCO 3.2.1,

7. Nuclear Enthalpy Rise Hot Channel Factor limits, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, for LCO 3.2.2,

8. Axial Flux Limits for LCO 3.2.3,

9. Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for Table 3.3.1-1,

10 Reactor Coolant System pressure, temperature, and flow in LCO 3.4.1,

11. Refueling Operations Boron Concentration for LCO 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

   (Methodology for LCOs 3.1.1 - SHUTDOWN MARGIN, 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limit, 3.1.6 - Control Bank Insertion Limits, 3.2.3 - Axial Flux Difference, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor and 3.9.1 – Boron Concentration.)

2. WCAP-10216-P-A, Rev.1A, "Relaxation of Constant Axial Offset Control / F_Q Surveillance Technical Specification," February 1994 (W Proprietary).

   (Methodology for LCOs 3.2.3 - Axial Flux Difference and 3.2.1 - Heat Flux Hot Channel Factor.)

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3a. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (W Proprietary).

3b. WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

(Methodology for LCO 3.2.1 - Heat Flux Hot Channel Factor and LCO 3.4.1- RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

4. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary)

(Methodology for Overpower ΔT and Thermal Overtemperature ΔT Trip Functions)

5. WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs. (Westinghouse Proprietary)

(Methodology for minimum RCS flow determination using the elbow tap measurement)

6. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988

(Methodology for LCO 3.9.1 – Boron Concentration)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

(Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.3.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated March 31, 1998 and April 3, 1998.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures shall be reported within 30 days. Reports on EDG failures shall include a description of the failures, underlying causes, and corrective actions taken per the Emergency Diesel Generator Reliability Monitoring Program.

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance

(continued)

5.6 Reporting Requirements

5.6.9 Tendon Surveillance Report (continued)

Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.10 Steam Generator Tube Inspector Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days of the completion of the plugging effort.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission within 12 months following the completion of the inspection. This Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be considered a Reportable Event and shall be reported pursuant to 10 CFR 50.73 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

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 hottest fuel rod in the core does not experience DNB; and

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

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, 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.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNB limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The Reactor Trip System Functions (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions (i.e., resulting from a Condition I or II event) for Reactor Coolant System (RCS) temperature, pressure, 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 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 provide more restrictive limits to ensure that the SLs are not exceeded.

BASES

SAFETY LIMITS

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 hottest fuel rod in the core does not experience DNB; and

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

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the 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 operations, 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 main steam 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

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.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 7.2.

BASES

APPLICABLE SAFETY ANALYSES (continued)

each analyzed transient. 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 3.1.6, "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 and the RCS average temperature limit specified in the COLR correspond to analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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 DNB design criterion in the event of a DNB limited transient.

RCS total flow rate is based on two elbow tap measurements from each loop and contains a measurement error of 2.3% based on Δp measurements from the cold leg elbow taps, which are correlated to past precision heat balance measurements or performing a precision heat balance at the beginning of the current cycle. 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 2.4%.

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, action shall be taken before performing subsequent precision heat balance measurements, i.e., 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.

BASES

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp > 5% RTP per minute or a THERMAL POWER step > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." The conditions that define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

(continued)