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March 24, 2003

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Duke Energy Corporation
McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369 and 50-370
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413, 50-414
Proposed Technical Specifications and Bases
Amendment
2.0, Safety Limits; 3.3, Instrumentation; 3.4,
Reactor Coolant System; 5.6, Reporting
Requirements

In accordance with the provisions of 10CFR50.90, Duke Energy Corporation (Duke) proposes to revise the McGuire and Catawba Nuclear Station Facility Operating Licenses and Technical Specifications (TS) and Bases. The proposed amendment reduces the required minimum measured Reactor Coolant System (RCS) flow rate from 390,000 gallons per minute (gpm) to 382,000 gpm for McGuire and Catawba Units 1 and 2. Additionally, the proposed change relocates RCS related cycle-specific parameter limits from the TS to, and thus expands, the Core Operating Limits Reports (COLR) for the McGuire and Catawba Nuclear Stations. These proposed changes will allow Duke the flexibility of enhancing operating and core design margins without the need for cycle-specific license amendment requests.

The proposed amendment changes TS 2.2.1, "Reactor Core Safety limits," and associated Bases, by relocating the reactor core safety limit figure to the COLRs and replacing it with specific fuel Departure from Nucleate Boiling Ratio (DNBR) and peak fuel centerline temperature safety limit requirements. The amendment proposes to change TS Table

ADD1

3.3.1-1, "Reactor Trip System Instrumentation," by relocating Overttemperature ΔT and Overpower ΔT nominal RCS operating pressure, nominal T_{avg} , and constant (K) values to the respective station COLR. The amendment also revises TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and associated Bases, by relocating the pressurizer pressure, RCS average temperature, and RCS total flow rate values to the respective station COLR. The minimum RCS total flow rates are retained in TS 3.4.1.3.

TS 5.6.5, "Core Operating Limits Report (COLR)," would be modified to reflect the above relocations to the COLR.

The requested changes are based upon NRC approved Westinghouse Owners Group (WOG) Technical Specifications Task Force (TSTF) TSTF-339, "Relocate TS Parameters to the COLR Consistent with WCAP-14483," Revision 2, and Westinghouse WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report." The return to a lower RCS minimum flow rate, previously reviewed and approved by the NRC, is similar to that submitted by the Comanche Peak Nuclear Station on May 24, 1999, and approved by the NRC in the Safety Evaluation Report dated August 30, 1999.

Duke requests approval of the proposed changes by August 1, 2003 to preclude the need to generate cycle-specific license amendment requests lowering the RCS total flow rates prior to the commencement of McGuire Unit 2, Cycle 16, and Catawba Unit 1, Cycle 15, operations. Duke has determined that the NRC's standard 30 day grace period will be sufficient for the implementation of this amendment.

The contents of this amendment package are as follows:

Attachment 1A provides marked copies of the affected TS and Bases pages for McGuire showing the proposed changes.
Attachment 1B provides marked copies of the affected TS and Bases pages for Catawba showing the proposed changes.
Attachment 2A, containing reprinted pages of the affected

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TS and Bases pages for McGuire, will be provided to the NRC upon issuance of the approved amendment. Attachment 2B, containing reprinted pages of the affected TS and Bases pages for Catawba, will be provided to the NRC upon issuance of the approved amendment. Attachment 3 provides a description of the proposed changes and technical justification. Pursuant to 10CFR50.92, Attachment 4 contains the results of the No Significant Hazards determination. Pursuant to 10CFR51.22(c)(9), Attachment 5 provides the basis for the categorical exclusion from the performance of an Environmental Assessment/Impact Review.

Implementation of this amendment will impact the McGuire and Catawba Updated Final Safety Analysis Reports (UFSAR). Changes to the UFSARs will be submitted in accordance with 10CFR50.71(e) requirements.

In accordance with Duke administrative procedures and Quality Assurance Program Topical Report requirements, this proposed amendment has previously been reviewed and approved by the McGuire and Catawba Plant Operations Review Committees and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10CFR50.91, a copy of this proposed amendment is being sent to the appropriate state officials.

Inquiries on this matter should be directed to J. A. Effinger at (704) 382-8688.

Very truly yours,


W. R. Mc Collum, Jr.
Senior Vice President
Nuclear Support

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W. R. Mc Collum, Jr., being duly sworn, affirms that he is the person who subscribed his name to the foregoing statement, and that all matters and facts set forth herein are true and correct to the best of his knowledge.



W. R. Mc Collum, Jr., Senior Vice President, Nuclear Support

Subscribed and sworn to me: March 24, 2003
Date

May P. Debus, Notary Public

My commission expires: JAN 22, 2006



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Catawba Owners: NCMPA-1, SREC, PMPA, NCEMC

McGuire Master File (MG01DM)

Catawba Document Control File 801.01 (CN04DM)

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ATTACHMENT 1A

McGUIRE UNITS 1 AND 2 TECHNICAL SPECIFICATIONS
AND
TECHNICAL SPECIFICATION BASES

MARKED COPY

INSERT 1

the COLR for four loop operation; and the following SLs shall not be exceeded:

- 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained \geq the 95/95 DNB criterion for the DNB correlation(s) specified in the thermal-hydraulic analytic method(s) listed in Section 5.6.5(b).
- 2.1.1.2 The peak fuel centerline temperature shall be maintained \leq the 95/95 CFM criterion for the fuel design(s) specified in the fuel mechanical analytic method(s) listed in Section 5.6.5(b).

INSERT 2

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

- a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criteria) that the hot fuel rod in the core does not experience DNB; and
- b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

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 Over Temperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor

core SLs will be satisfied during steady state operation, normal operational transients, and AOs.

INSERT 3

The numerical limits of these variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on previously analyzed maximum steam generator tube plugging, is retained in the TS LCO.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the ~~SLs~~ ^{LIMITS} specified in ~~Figure 2.1.1 for four loop operation.~~ ^{INSERT 1}

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.

DELETE

SLs
2.0

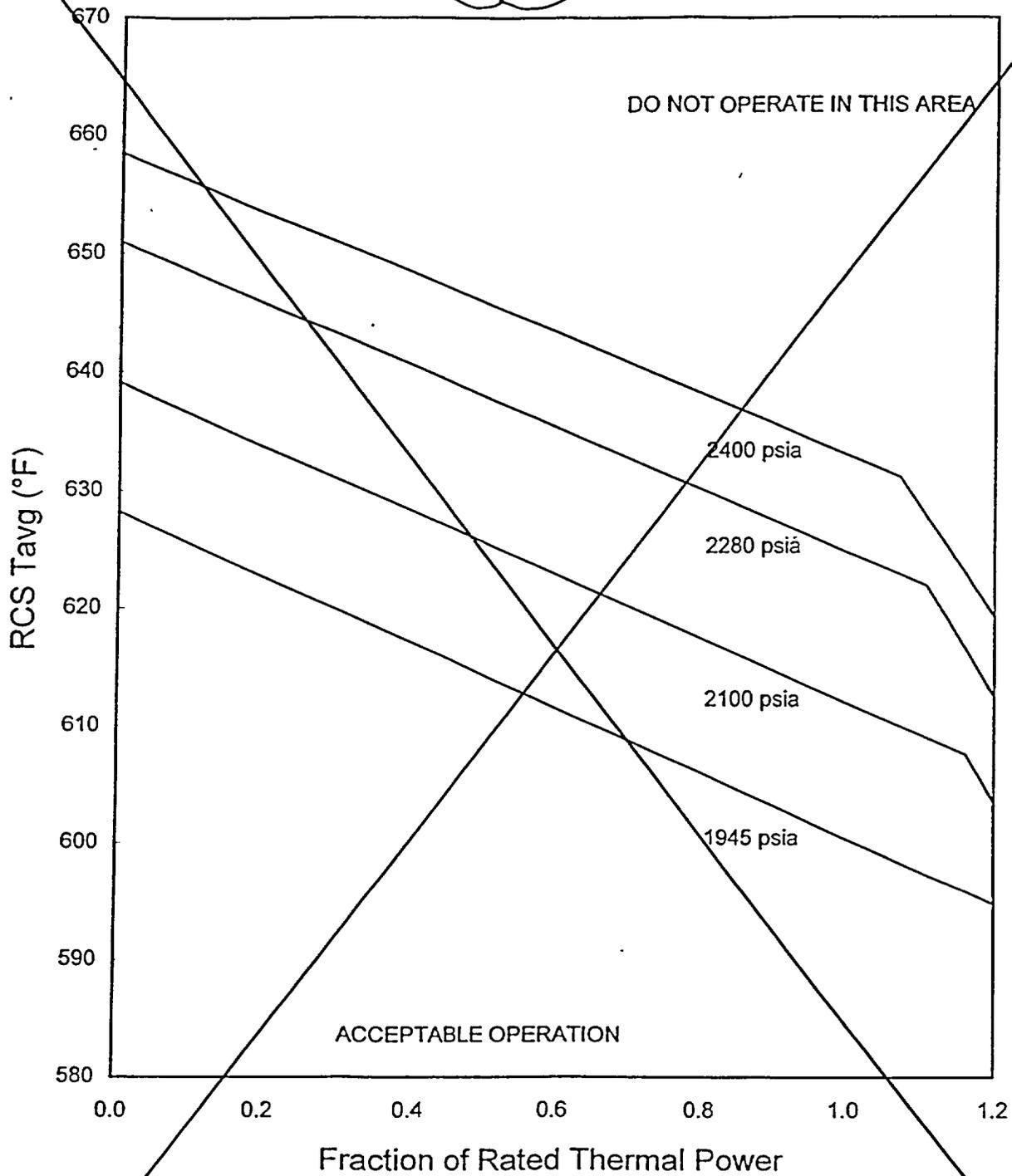


Figure 2.1.1-1

Reactor Core Safety Limits -
Four Loops in Operation

Table 3.3.1-1 (page 5 of 7)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following NOMINAL TRIP SETPOINT by more than 4.4% of RTP.

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

Where: ΔT is measured RCS ΔT by loop narrow range RTDs, °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 nominal T_{avg} at RTP, ~~585.1~~ °F. ←

THE VALUE SPECIFIED IN THE COLR

P is the measured pressurizer pressure, psig

P' is the nominal RCS operating pressure, = ~~2235~~ psig

THE VALUE SPECIFIED IN THE COLR

- K_1 = Overtemperature ΔT reactor trip NOMINAL TRIP SETPOINT, as presented in the COLR,
- K_2 = Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, as presented in the COLR,
- K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, as presented in the COLR,
- τ_1, τ_2 = Time constants utilized in the lead-lag controller for ΔT , as presented in the COLR,
- τ_3 = Time constants utilized in the lag compensator for ΔT , as presented in the COLR,
- τ_4, τ_5 = Time constants utilized in the lead-lag controller for T_{avg} , as presented in the COLR,
- τ_6 = Time constants utilized in the measured T_{avg} lag compensator, as presented in the COLR, and,

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

- (i) for $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the COLR; $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(continued)

Table 3.3.1-1 (page 6 of 7)
Reactor Trip System Instrumentation

- (ii) for each percent imbalance that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the COLR; and
- (iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the COLR.

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following NOMINAL TRIP SETPOINT by more than 3.0% of RTP.

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

Where: ΔT is measured RCS ΔT by loop narrow range RTDs, °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 nominal T_{avg} at RTP, $\leq 585.1^\circ F$. *(THE VALUE SPECIFIED IN THE COLR)*

K_4 = Overpower ΔT reactor NOMINAL TRIP SETPOINT as presented in the COLR,

K_5 = ~~0.021%~~ for increasing average temperature and ~~0~~ for decreasing average temperature, *(THE VALUE SPECIFIED IN THE COLR)*

K_6 = Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the COLR for $T > T'$ and $K_6 = 0$ for $T \leq T'$.

τ_1, τ_2 = Time constants utilized in the lead-lag controller for ΔT , as presented in the COLR,

τ_3 = Time constants utilized in the lag compensator for ΔT , as presented in the COLR,

τ_6 = Time constants utilized in the measured T_{avg} lag compensator, as presented in the COLR,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg} , as presented in the COLR, and

$f_2(\Delta I)$ = 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:

(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 in Table 3.4.1.1.

AND

THE COLR, AND RCS TOTAL FLOW RATE SHALL BE GREATER THAN OR EQUAL TO 362,000 GPM AND THE LIMIT SPECIFIED IN THE COLR

APPLICABILITY:

MODE T.

-----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. Pressurizer pressure or RCS average temperature DNB parameters not within limits.	A.1 Restore DNB parameter(s) to within limit.	2 hours
B. RCS total flow rate ≥ 390,000 gpm but ≥ 386,100 gpm	B.1 Reduce THERMAL POWER to ≤ 98% RTP.	2 hours
<p>< THE VALUE SPECIFIED IN THE COLR BUT ≥ 99% OF THE VALUE SPECIFIED IN THE COLR.</p>	AND	6 hours
	B.2 Reduce the Power Range Neutron Flux - High Trip Setpoint below the nominal setpoint by 2% RTP.	

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.1.1 Verify pressurizer pressure is within limits _x ← <i>SPECIFIED IN THE COLR.</i>	12 hours
SR 3.4.1.2 Verify RCS average temperature is within limits _x ← <i>SPECIFIED IN THE COLR.</i>	12 hours
SR 3.4.1.3 Verify RCS total flow rate is within limits_x <i>GREATER THAN OR EQUAL TO 382,000 GPM AND THE LIMIT SPECIFIED IN THE COLR.</i>	12 hours
SR 3.4.1.4 Perform CHANNEL CALIBRATION for each RCS total flow indicator.	18 months

DELETE

Table 3.4.1-1 (page 1 of 1)
RCS DNB Parameters

PARAMETER	INDICATION	No. OPERABLE CHANNELS	LIMITS
1. Indicated RCS Average Temperature	meter	4	$\leq 587.2\text{ }^{\circ}\text{F}$
	meter	3	$\leq 586.9\text{ }^{\circ}\text{F}$
	computer	4	$\leq 587.7\text{ }^{\circ}\text{F}$
	computer	3	$\leq 587.5\text{ }^{\circ}\text{F}$
2. Indicated Pressurizer Pressure	meter	4	$\geq 2219.8\text{ psig}$
	meter	3	$\geq 2222.1\text{ psig}$
	computer	4	$\geq 2215.8\text{ psig}$
	computer	3	$\geq 2217.5\text{ psig}$
3. RCS Total Flow Rate			$\geq 390,000\text{ gpm}$

BASES

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

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

RCS Flow Rate, ΔI ,

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

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

APPROPRIATE OPERATION OF THE RPS AND THE STEAM GENERATOR SAFETY VALVES.

- a. / High pressurizer pressure trip,
- b. / Low pressurizer pressure trip,
- c. / Overtemperature ΔT trip,
- d. / Overpower ΔT trip,
- e. / Power Range Neutron Flux trip, and
- f. / Steam generator safety valves.

~~The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.~~

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

FIGURE

THE COLR SHOWS

BASES

SAFETY LIMITS

EDITORIAL

FRACTION OF RATED THERMAL POWER,

The curves provided in Figure B 2.1.1-1 show the loci of points of RCS Pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, and that the exit quality is within the limits defined by the DNBR correlation.

INSERT 2

The curves in Figure 2.1.1-1 are based on a reference nuclear enthalpy rise hot channel factor (F_{DH}), a reference axial power shape ($F_z/x/L$), the approved CHF correlation and the Technical Specification minimum flow rate. Therefore, these curves provide limits for which the analyses analyzed at the above reference values will be bounded. The curves in Figure B 2.1.1-1 illustrate the various RFS functions that are designed to prevent the unit from reaching the limit.

The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the Overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the Overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Ref. 3).

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

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

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Section 7.2.
3. ~~DPC-NE-2011PA, March 1990.~~

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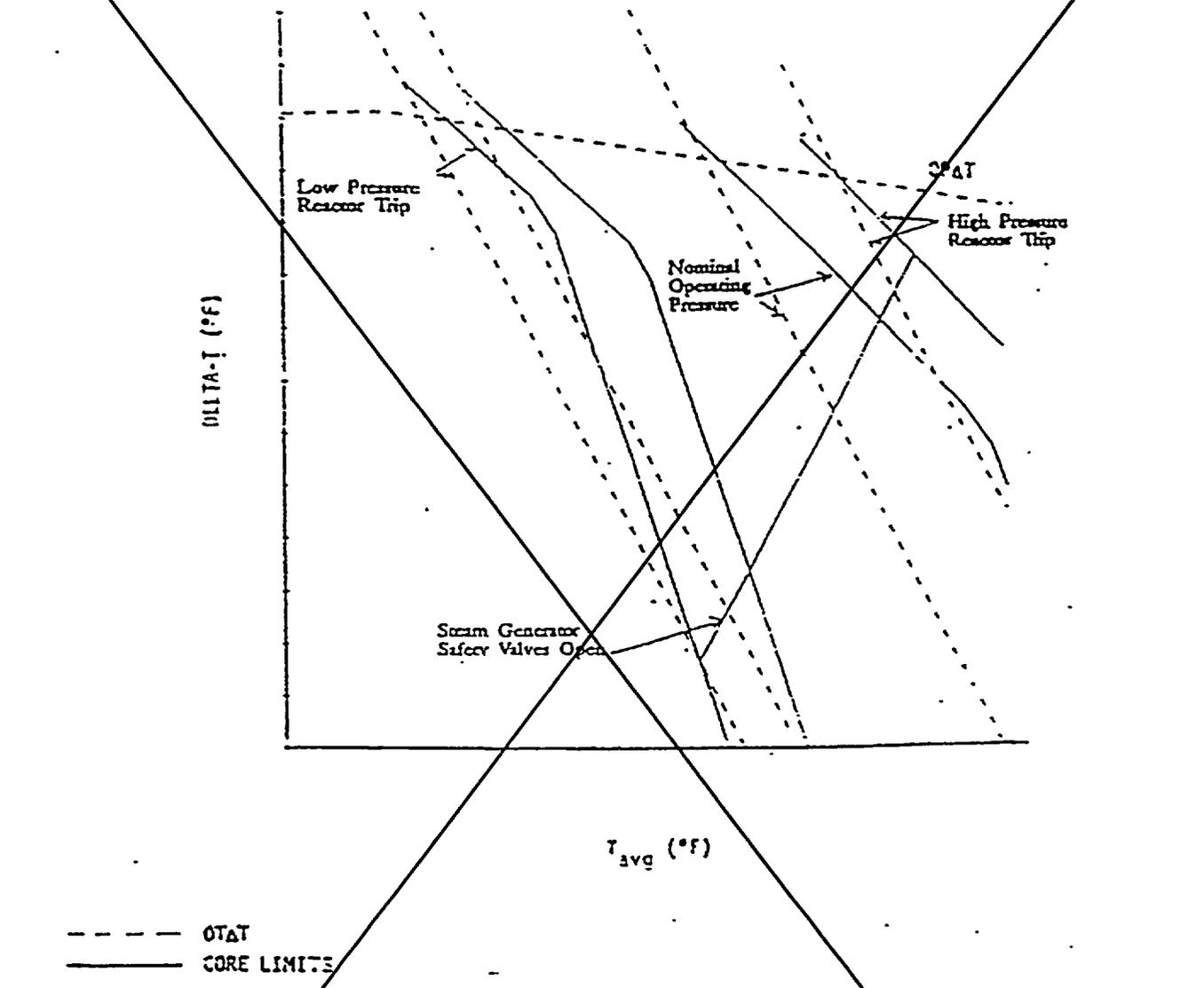


Figure B 2.1.1-1

Illustration of Overtemperature
and Overpower ΔT Protection

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level-High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure—High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The setpoints are based on percent of instrument span. The LCO requires three channels of Pressurizer Water Level—High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level—High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow-Low

a. Reactor Coolant Flow-Low (Single Loop)

The Reactor Coolant Flow-Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48% RTP, a loss of flow in any RCS loop will actuate a reactor trip. The setpoints are based on a minimum ~~measured~~ flow ~~of 1500~~

THE

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

SPECIFIED IN THE COLR.

~~gpm~~. Each RCS 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 P-8.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in 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.

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. The setpoints are based on a minimum ~~measured~~ flow of ~~97,500~~

SPECIFIED IN THE COLR.

~~gpm~~. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

THE

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 power distributions that would cause a DNB concern at this low power level are unlikely. 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

assessed for their impact on the acceptance criteria. 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)."

SPECIFIED IN THE COLR

The pressurizer pressure limits and the RCS average temperature limits correspond to analytical limits of ~~2205 psia~~ and ~~589.1°F~~ used in the safety analyses, with allowance for measurement uncertainty.

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

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 acceptance criteria, including the DNBR criterion.

INSERT 3

RCS total flow rate contains a measurement error of ~~1.1%~~ based on the performance of past precision heat balances and using the result to calibrate the RCS flow rate indicators. Sets of elbow tap coefficients, as determined during these heat balances, were averaged for each elbow tap to provide a single set of elbow tap coefficients for use in calculating RCS flow. This set of coefficients establishes the calibration of the RCS flow rate indicators and becomes the set of elbow tap coefficients used for RCS flow measurement. Potential fouling of the feedwater venturi, which might not have been detected, could have biased the result from these past precision heat balances in a nonconservative manner. Therefore, a penalty of ~~0.1%~~ for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to ~~1.8%~~ for no fouling.

The ~~LCO~~ numerical values in ~~Table 3.A.1.1~~ for pressure and average temperature are given for the measurement location with adjustments for the indication instruments.

SPECIFIED IN THE COLR

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.

BASES

APPLICABILITY (continued)

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 increase > 5% RTP per minute or a THERMAL POWER step increase > 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 CONDITIONS WHICH DEFINE THE DNBR LIMIT

THE DNBR LIMIT

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits 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

Pressurizer 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).

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.

LESS THAN THE VALUE SPECIFIED IN THE COLR, BUT GREATER THAN OR EQUAL TO 99% OF THE VALUE SPECIFIED IN THE COLR,

B.1 and B.2

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 ~~less than 290,000 gpm or 288,100 gpm~~ then THERMAL POWER may not exceed 98% RTP. THERMAL POWER must be reduced within 2 hours. The completion time of 2 hours is consistent with Required Action A.1. In addition, the Power Range Neutron Flux - High Trip Setpoint must be reduced from the nominal setpoint by 2% RTP within 6 hours. The Completion Time of 6 hours to reset the trip setpoints recognizes that, with power reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

BASES

ACTIONS (continued)

C.1, C.2.1, C.2.2, and C.2.3

LESS THAN 99% OF THE VALUE SPECIFIED IN THE COLR,

GREATER THAN OR EQUAL TO 99% OF THE VALUE SPECIFIED IN THE COLR

If the indicated RCS total flow rate is ~~1,386,100 gpm~~, then RCS total flow must be restored to ~~1,386,100 gpm~~ within 2 hours or power must be reduced to less than 50% RTP. The Completion Time of 2 hours is consistent with Required Action A.1. If THERMAL POWER is reduced to less than 50% RTP, the Power Range Neutron Flux - High Trip Setpoint must also be reduced to $\leq 55\%$ RTP. The Completion Time of 6 hours to reset the trip setpoints is consistent with Required Action B.2. This is a sensitive operation that may inadvertently trip the Reactor Protection System. Operation is permitted to continue provided the RCS total flow is restored to ~~1,386,100 gpm~~ within 24 hours. The Completion Time of 24 hours is reasonable considering the increased margin to DNB at power levels below 50% and the fact that power increases associated with a transient are limited by the reduced trip setpoint.

D.1

If the Required Actions are not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This surveillance demonstrates that the pressurizer pressure remains within the required limits. Alarms and other indications are available to alert operators if this limit is approached or exceeded. The frequency of 12 hours is sufficient, considering the other indications available to the operator in the control room for monitoring the RCS pressure and related equipment status. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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 Chapter 16 of the UFSAR 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.

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:

2.X Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3.1.3,

ILLUSTRATION OF REACTOR CORE SAFETY LIMITS
FOR SPECIFICATION 2.1.1.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

- 3. ~~X~~. Shutdown Bank Insertion Limit for Specification 3.1.5,
- 4. ~~X~~. Control Bank Insertion Limits for Specification 3.1.6,
- 5. ~~X~~. Axial Flux Difference limits for Specification 3.2.3,
- 6. ~~X~~. Heat Flux Hot Channel Factor for Specification 3.2.1,
- 7. ~~X~~. Nuclear Enthalpy Rise Hot Channel Factor limits for Specification 3.2.2,
- 8. ~~X~~. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,
- 10. ~~X~~. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,
- 11. ~~X~~. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,
- 12. ~~X~~. Spent fuel pool boron concentration limits for Specification 3.7.14,
- 13. ~~X~~. SHUTDOWN MARGIN for Specification 3.1.1.

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," (W Proprietary).
- 2. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," (W Proprietary).
- 3. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," (B&W Proprietary).

9. REACTOR COOLANT SYSTEM PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS FOR SPECIFICATION 3.4.1,

(continued)

ATTACHMENT 1B

CATAWBA UNITS 1 AND 2 TECHNICAL SPECIFICATIONS
AND
TECHNICAL SPECIFICATION BASES

MARKED COPY

INSERT 1

the COLR for four loop operation; and the following SLs shall not be exceeded:

- 2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained \geq the 95/95 DNB criterion for the DNB correlation(s) specified in the thermal-hydraulic analytic method(s) listed in Section 5.6.5(b).
- 2.1.1.2 The peak fuel centerline temperature shall be maintained \leq the 95/95 CFM criterion for the fuel design(s) specified in the fuel mechanical analytic method(s) listed in Section 5.6.5(b).

INSERT 2

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

- a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criteria) that the hot fuel rod in the core does not experience DNB; and
- b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

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 Over Temperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor

core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

INSERT 3

The numerical limits of these variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on previously analyzed maximum steam generator tube plugging, is retained in the TS LCO.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1.1 for four loop operation.

LIMITS

INSERT 1

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained \leq 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.

SLs
2.0

DELETE

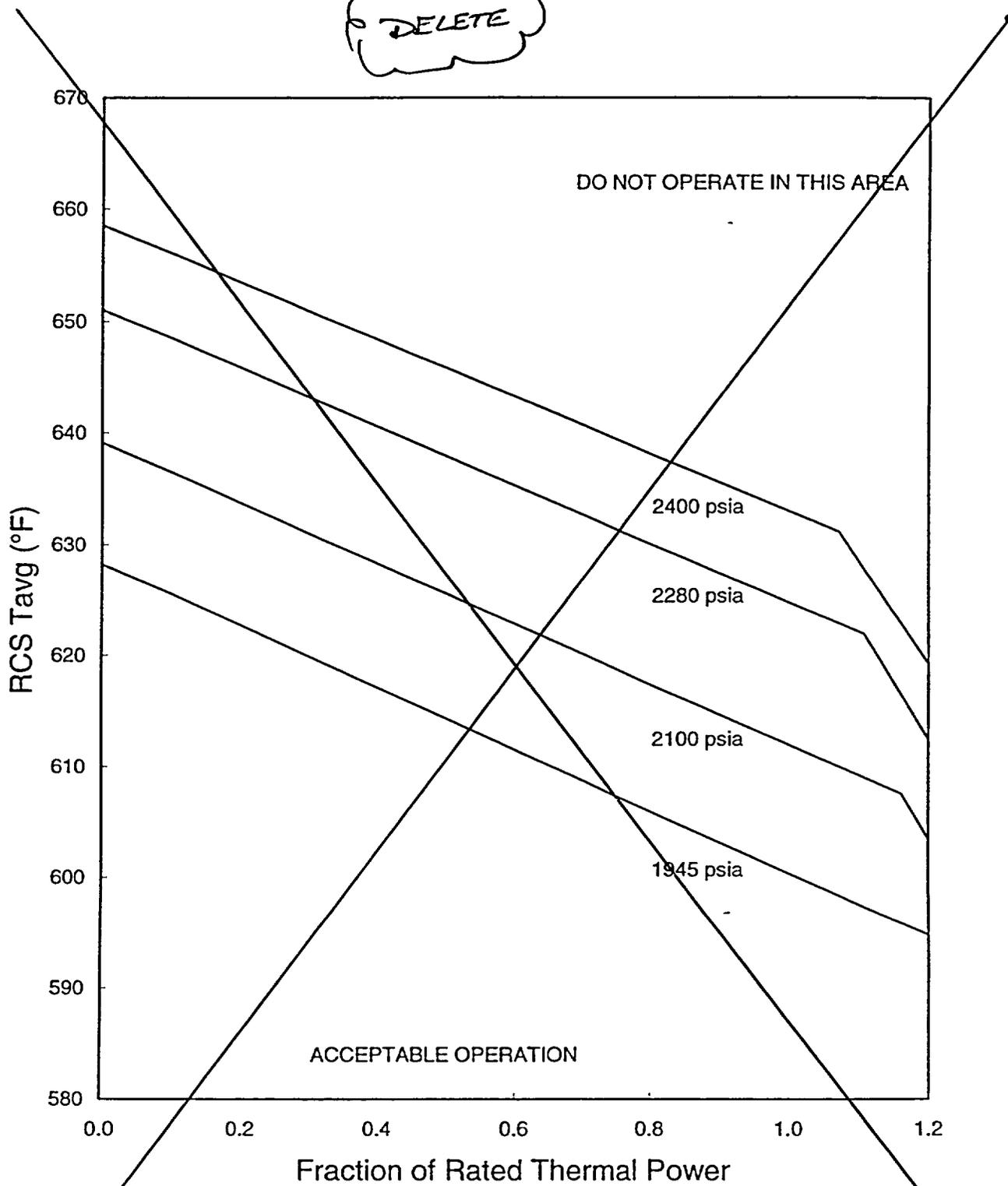


Figure 2.1.1-1
Reactor Core Safety Limits
Four Loops in Operation

Table 3.3.1-1 (page 5 of 7)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following NOMINAL TRIP SETPOINT by more than 4.3% (Unit 1) and 4.5% (Unit 2) of RTP.

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

Where: ΔT is the measured RCS ΔT by loop narrow range RTDs, °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 nominal T_{avg} at RTP (allowed by Safety Analysis), $\leq 585.1^\circ\text{F (Unit 1)}$ ←

$\leq 590.8^\circ\text{F (Unit 2)}$

P is the measured pressurizer pressure, psig

P' is the nominal RCS operating pressure, = 2235 psig

THE VALUES SPECIFIED IN THE COLR

THE VALUE SPECIFIED IN THE COLR

K_1 = Overtemperature ΔT reactor NOMINAL TRIP SETPOINT, as presented in the COLR,

K_2 = Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, as presented in the COLR,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, as presented in the COLR,

τ_1, τ_2 = Time constants utilized in the lead-lag compensator for ΔT , as presented in the COLR,

τ_3 = Time constant utilized in the lag compensator for ΔT , as presented in the COLR,

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg} , as presented in the COLR,

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the COLR, and

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

(i) for $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the COLR; $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the COLR; and

(continued)

Table 3.3.1-1 (page 6 of 7)
Reactor Trip System Instrumentation

- (iii) for each percent ΔI that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the COLR.

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following NOMINAL TRIP SETPOINT by more than 2.6% (Unit 1) and 3.1% (Unit 2) of RTP.

$$\Delta T \frac{(1 + \tau_1 s)}{(1 + \tau_2 s)} \left(\frac{1}{1 + \tau_3 s} \right) \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_7 s}{1 + \tau_7 s} \left(\frac{1}{1 + \tau_6 s} \right) T - K_6 \left[T \frac{1}{1 + \tau_6 s} - T^* \right] - f_2(\Delta I) \right\}$$

Where: ΔT is the measured RCS ΔT by loop narrow range RTDs, °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 nominal T_{avg} at RTP (calibration temperature for ΔT instrumentation).

$\leq 585.1^\circ\text{F}$ (Unit 1) $\leq 590.8^\circ\text{F}$ (Unit 2). THE VALUES SPECIFIED IN THE COLR

K_4 = Overpower ΔT reactor NOMINAL TRIP SETPOINT as presented in the COLR,

K_5 = ~~0.02%~~ for increasing average temperature and ~~0~~ for decreasing average temperature, THE VALUE SPECIFIED IN THE COLR

K_6 = Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the COLR for $T > T^*$ and $K_6 = \text{0}$ for $T \leq T^*$,

τ_1, τ_2 = Time constants utilized in the lead-lag compensator for ΔT , as presented in the COLR,

τ_3 = Time constant utilized in the lag compensator for ΔT , as presented in the COLR,

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the COLR,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg} , as presented in the COLR, and

$f_2(\Delta I)$ = a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

- (i) for $q_t - q_b$ between the "positive" and "negative" $f_2(\Delta I)$ breakpoints as presented in the COLR; $f_2(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(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, ^{SAND}RCS average temperature, ~~and RCS total flow rate~~ shall be within the limits specified in ~~Table B.4.XX~~ ←

THE COLR AND RCS TOTAL FLOW RATE SHALL BE GREATER THAN OR EQUAL TO 382,000 GPM 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. Pressurizer pressure or RCS average temperature DNB parameters not within limits.	A.1 Restore DNB parameter(s) to within limit.	2 hours
B. RCS total flow rate < 380,000 gpm or < 382,100 gpm ← ← THE LIMIT SPECIFIED IN THE COLR BUT ≥ 99% OF THE VALUE SPECIFIED IN THE COLR.	B.1 Reduce THERMAL POWER to ≤ 98% RTP. AND B.2 Reduce the Power Range Neutron Flux – High Trip Setpoint below the nominal setpoint by 2% RTP.	2 hours 6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. RCS total flow rate to $\leq 286,100$ gpm</p> <p><i>< 99% OF THE VALUE SPECIFIED IN THE COLR.</i></p>	<p>C.1 Restore RCS total flow rate to $\leq 286,100$ gpm</p> <p><i>$\geq 99\%$ OF THE VALUE SPECIFIED IN THE COLR.</i></p> <p>OR</p> <p>C.2.1 Reduce THERMAL POWER to $< 50\%$ RTP.</p> <p><u>AND</u></p> <p>C.2.2 Reduce the Power Range Neutron Flux - High Trip Setpoint to $\leq 55\%$ RTP.</p> <p><u>AND</u></p> <p>C.2.3 Restore RCS total flow rate to $\leq 286,100$ gpm</p> <p><i>$\geq 99\%$ OF THE VALUE SPECIFIED IN THE COLR.</i></p>	<p>2 hours</p> <p>2 hours</p> <p>6 hours</p> <p>24 hours</p>
<p>D. Required Action and associated Completion Time not met.</p>	<p>D.1 Be in MODE 2.</p>	<p>6 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.1.1 Verify pressurizer pressure is within limits _x ← <i>(SPECIFIED IN THE CLR.)</i>	12 hours
SR 3.4.1.2 Verify RCS average temperature is within limits _x ← <i>(SPECIFIED IN THE CLR.)</i>	12 hours
SR 3.4.1.3 Verify RCS total flow rate is within limits _x ←	12 hours
SR 3.4.1.4 Perform CHANNEL CALIBRATION for each RCS total flow indicator.	18 months

GREATER THAN OR EQUAL TO 382,000 GPM AND THE LIMIT SPECIFIED IN THE CLR.

DELETE

Table 3.4.1-1 (page 1 of 1)
RCS DNB Parameters

PARAMETER	INDICATION	No. OPERABLE CHANNELS	LIMITS
1. Indicated RCS Average Temperature – Unit 1	meter	4	≤ 587.2 °F
	meter	3	≤ 586.9 °F
	computer	4	≤ 587.7 °F
	computer	3	≤ 587.5 °F
Indicated RCS Average Temperature – Unit 2	meter	4	≤ 592.9 °F
	meter	3	≤ 592.6 °F
	computer	4	≤ 593.4 °F
	computer	3	≤ 593.2 °F
2. Indicated Pressurizer Pressure	meter	4	≥ 2219.8 psig
	meter	3	≥ 2222.1 psig
	computer	4	≥ 2215.8 psig
	computer	3	≥ 2217.5 psig
3. RCS Total Flow Rate			$\geq 390,000$ gpm

BASES

APPLICABLE
SAFETY ANALYSES

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

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

RCS FLOW RATE, ΔT ,

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

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

- a. / High pressurizer pressure trip;
 - b. / Low pressurizer pressure trip;
 - c. / Overtemperature ΔT trip;
 - d. / Overpower ΔT trip;
 - e. / Power Range Neutron Flux trip; and
 - f. / Steam generator safety valves.
- The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

← APPROPRIATE OPERATION OF THE RPS AND THE STEAM GENERATOR SAFETY VALVES.

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

BASES

FIGURE

THE COLOR SHOWS

EDITORIAL

SAFETY LIMITS

FRACTION OF RATED THERMAL POWER

The ~~curves~~ provided in ~~Figure B.2.1.1-1~~ show the loci of points of ~~CHF~~, RCS Pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, and that the exit quality is within the limits defined by the DNBR correlation.

INSERT 2

~~The curves in Figure 2.1.1-1 are based on a reference nuclear enthalpy rise hot channel factor (F_{CH}), a reference axial power shape ($F_z/x/L$), the approved CHF correlation and the Technical Specification minimum flow rate. Therefore, these curves provide limits for which the analyses analyzed at the above reference values will be bounded. The curves in Figure B 2.1.1-1 illustrate the various RPS functions that are designed to prevent the unit from reaching the limit.~~

~~The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta T)$ function of the Overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the Overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Ref. 3).~~

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

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

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Section 7.2.
3. ~~DPC-NE-2011PA, March 1990~~

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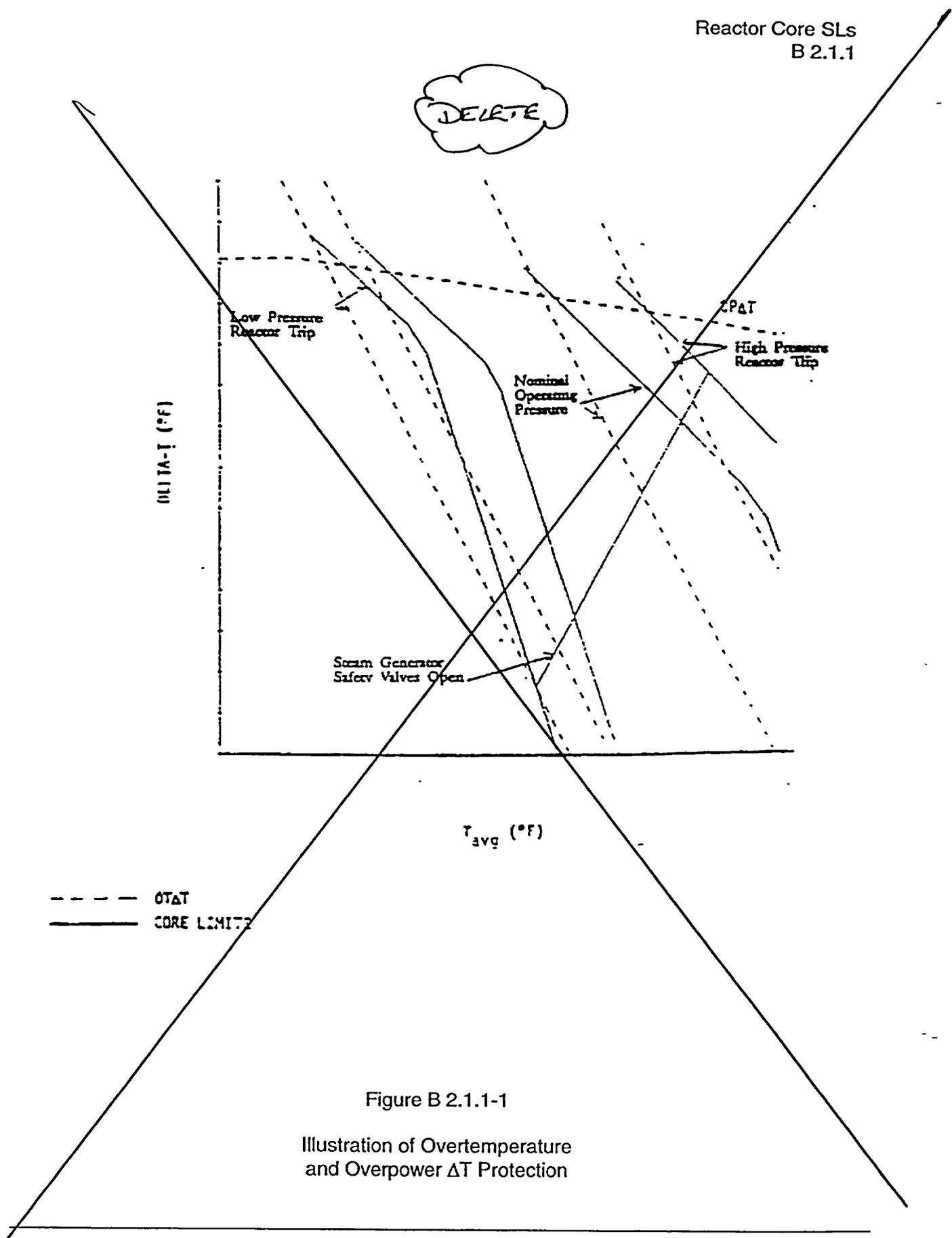


Figure B 2.1.1-1
Illustration of Overtemperature
and Overpower ΔT Protection

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

SPECIFIED IN THE COLR.

setpoints are based on ^{THE} a minimum measured flow of ~~95,500~~ ~~gpm (Unit 1) 96,250 gpm (Unit 2)~~. Each RCS 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 P-8.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in 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.

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.

SPECIFIED IN THE COLR.

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. The setpoints are based on ^{THE} a minimum measured flow of ~~95,500~~ ~~gpm (Unit 1) 96,250 gpm (Unit 2)~~. 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 power distributions that would cause a DNB concern at this low power level are unlikely. 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

(SPECIFIED IN THE COLR)

The pressurizer pressure limits and the RCS average temperature limits correspond to analytical limits of ~~2205 psig~~ and ~~589.1°F (Unit 1) and 594.8°F (Unit 2)~~ used in the safety analyses, with allowance for measurement uncertainty.

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

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 acceptance criteria, including the DNBR criterion. *(INSERT 3)*

RCS total flow rate contains a measurement error of ~~1.2%~~ based on the performance of past precision heat balances and using the result to calibrate the RCS flow rate indicators. Sets of elbow tap coefficients, as determined during these heat balances, were averaged for each elbow tap to provide a single set of elbow tap coefficients for use in calculating RCS flow. This set of coefficients establishes the calibration of the RCS flow rate indicators and becomes the set of elbow tap coefficients used for RCS flow measurement. Potential fouling of the feedwater venturi, which might not have been detected, could have biased the result from these past precision heat balances in a nonconservative manner. Therefore, a penalty of ~~1.0%~~ for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to ~~1.2%~~ for no fouling.

(SPECIFIED IN THE COLR)

The ~~LCO~~ numerical values in ~~Table B.3.4.1.1~~ for pressure and average temperature are given for the measurement location with adjustments for the indication instruments.

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 increase > 5% RTP per minute or a THERMAL POWER step increase > 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.

BASES

APPLICABILITY (continued)

THE DNB LIMIT

THE CONDITIONS WHICH DEFINE THE DNB LIMIT

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits 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

Pressurizer 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).

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.

LESS THAN THE VALUE SPECIFIED IN THE COLR, BUT GREATER THAN OR EQUALS 99% OF THE VALUE SPECIFIED IN THE COLR,

B.1 and B.2

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 ~~1,300,000 gpm~~ ~~or~~ ~~1,386,100 gpm~~, then THERMAL POWER may not exceed 98% RTP. THERMAL POWER must be reduced within 2 hours. The Completion Time of 2 hours is consistent with Required Action A.1. In addition, the Power Range Neutron Flux - High Trip Setpoint must be reduced from the nominal setpoint by 2% RTP within 6 hours. The Completion Time of 6 hours to reset the trip setpoints recognizes that, with power reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

GREATER THAN OR EQUAL TO 99% OF THE VALUE SPECIFIED IN THE COLR

C.1, C.2.1, C.2.2, and C.2.3

LESS THAN 99% OF THE VALUE SPECIFIED IN THE COLR,

If the indicated RCS total flow rate is ~~1,386,100 gpm~~, then RCS total flow must be restored to ~~1,386,100 gpm~~ within 2 hours or power must be reduced to less than 50% RTP. The Completion Time of 2 hours is consistent with Required Action A.1. If THERMAL POWER is reduced to less than 50% RTP, the Power Range Neutron Flux - High Trip Setpoint

BASES

ACTIONS (continued)

GREATER THAN OR
EQUAL TO 99% OF
THE VALUE SPECIFIED
IN THE COLR

must also be reduced to $\leq 55\%$ RTP. The Completion Time of 6 hours to reset the trip setpoints is consistent with Required Action B.2. This is a sensitive operation that may inadvertently trip the Reactor Protection System. Operation is permitted to continue provided the RCS total flow is restored to ~~2,886,100 gpm~~ within 24 hours. The Completion Time of 24 hours is reasonable considering the increased margin to DNB at power levels below 50% and the fact that power increases associated with a transient are limited by the reduced trip setpoint.

D.1

If the Required Actions are not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.1

This surveillance demonstrates that the pressurizer pressure remains within the required limits. Alarms and other indications are available to alert operators if this limit is approached or exceeded. The frequency of 12 hours is sufficient, considering the other indications available to the operator in the control room for monitoring the RCS pressure and related equipment status. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

This surveillance demonstrates that the average RCS temperature remains within the required limits. Alarms and other indications are available to alert operators if this limit is approached or exceeded. The frequency of 12 hours is sufficient, considering the other indications available to the operator in the control room for monitoring the RCS. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. ILLUSTRATION OF REACTOR CORE SAFETY LIMITS FOR SPECIFICATION 2.1.1,

- 2. X. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3.1.3,
 - 3. X. Shutdown Bank Insertion Limit for Specification 3.1.5,
 - 4. X. Control Bank Insertion Limits for Specification 3.1.6,
 - 5. A. Axial Flux Difference limits for Specification 3.2.3,
 - 6. X. Heat Flux Hot Channel Factor for Specification 3.2.1,
 - 7. X. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.2.2,
 - 8. X. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,
 - 10. X. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,
 - 11. X. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,
 - 12. X. Spent fuel pool boron concentration limits for Specification 3.7.15,
 - 13. X. SHUTDOWN MARGIN for Specification 3.1.1.
- 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" (W Proprietary).
 - 2. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE" (W Proprietary).

9. REACTOR COOLANT SYSTEM PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS FOR SPECIFICATION 3.4.1,

(continued)

ATTACHMENT 2A

McGUIRE UNITS 1 AND 2 TECHNICAL SPECIFICATIONS
AND
TECHNICAL SPECIFICATION BASES

(TO BE PROVIDED TO THE NRC UPON ISSUANCE OF APPROVED
AMENDMENT)

ATTACHMENT 2B

CATAWBA UNITS 1 AND 2 TECHNICAL SPECIFICATIONS
AND
TECHNICAL SPECIFICATION BASES

(TO BE PROVIDED TO THE NRC UPON ISSUANCE OF APPROVED
AMENDMENT)

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

Proposed Changes

The proposed changes reduce the required minimum measured Reactor Coolant System (RCS) flow rate from 390,000 gpm to a previously reviewed and approved value of 382,000 gpm and relocate RCS related cycle-specific parameter limits from the Technical Specifications (TS) to, and thus expand, the Core Operating Limits Reports (COLR) for the McGuire and Catawba Nuclear Stations.

McGuire and Catawba Units 1 and 2 were originally licensed with a minimum reactor coolant system (RCS) flow rate of 385,000 gpm. Due to steam generator tube plugging issues, analyses were performed for McGuire and Catawba Units 1 and 2 assuming a bounding tube plugging percentage and a license amendment to reduce the RCS minimum flow rate from 385,000 gpm to 382,000 gpm was requested for McGuire Units 1 and 2 and Catawba Unit 1. Although these analyses were applicable to Catawba Unit 2, a license amendment to lower the RCS minimum flow rate was not requested for this unit because Catawba Unit 2's steam generators had not experienced the same rate of tube plugging as had Catawba Unit 1. This license amendment request was approved by the NRC staff in Safety Evaluation Reports dated December 17, 1993 (for Catawba Unit 1)¹ and March 22, 1994 (for McGuire Units 1 and 2)².

Subsequent to the replacement of steam generators, McGuire Unit 1 and 2 and Catawba Unit 1 were successfully operated with an RCS minimum total flow rate of 382,000 gpm in the technical specifications until those values were changed to 390,000 gpm through NRC Safety Evaluation Reports of March 2, 2000 for McGuire, and March 1, 2000 for Catawba. The NRC Safety Evaluation Report of March 1, 2000, also revised the Catawba Unit 2 RCS minimum total flow rate from 385,000 gpm to 390,000 gpm in the technical specifications. The analyses supporting the RCS minimum total flow rate of 390,000 gpm assumed a minimal steam generator tube plugging

¹ Reference Duke submittal of October 25, 1993, as supplemented by letters of December 3 and 6, 1993

² Reference Duke submittal of October 25, 1993, as supplemented by letters of December 3, 1993, and February 14, 1994

percentage. The RCS minimum total flow rates for McGuire and Catawba Units 1 and 2 were increased to make more effective use of available operating and analytical margins. This 390,000 gpm RCS total flow rate should be considered a cycle-specific minimum value, reflecting the condition of the McGuire and Catawba steam generators at the time the license amendment request was made.

Values for the RCS pressure, temperature, and flow are specified in TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure for Nucleate Boiling (DNB) Limits." The proposed change reduces the required minimum measured RCS flow rate for McGuire and Catawba Units 1 and 2 from 390,000 gpm to 382,000 gpm, and replaces the current minimum RCS flow rate of 390,000 gpm with a reference to the COLR.

The previously reviewed and approved accident evaluation of the replacement steam generators at a minimum RCS flow rate of 382,000 gpm was included as Attachment 1 to the McGuire Unit 1 and 2³, and Catawba Unit 1⁴ steam generator replacement license amendment requests of September 30, 1994.

The proposed revision to TS 3.4.1, and associated Bases, further relocates Table 3.4.1-1, "RCS DNB Parameters," to the COLR. The 382,000 gpm RCS minimum flow rate, will be maintained in TS 3.4.1 so as to assure that a lower flow rate will not be used without prior NRC approval.

TS 5.6.5, "Core Operating Limits Report (COLR)," has been modified to reflect the above relocations to the COLR.

The proposed amendment also relocates TS Figure 2.1.1-1, "Reactor Core Safety Limits," to the COLRs, replacing it with fuel integrity limits (95/95 DNB and 95/95 CFM

³ Reference Duke submittal of September 30, 1994, as supplemented by letters of September 18, 1995, and March 15, April 29, May 16, September 23, and October 28, 1996, and January 16, April 22, and May 2, 1997, and the NRC Safety Evaluation Report dated May 5, 1997, for McGuire Units 1 and 2

⁴ Reference Duke submittal of September 30, 1994, as supplemented by letters of September 18, 1995, January 19, March 15, May 16, and August 27, 1996; and the NRC Safety Evaluation Report of August 29, 1996, for Catawba Unit 1

criteria) to reflect McGuire and Catawba's use of different fuel designs from various vendors and to better represent the safety limits requirement to 10CFR §50.36. As discussed in the Safety Evaluation Reports to WCAP-14483-A, it is necessary to relocate TS Figure 2.1.1-1 to the COLR since cycle-dependent changes to parameters upon which TS Figure 2.1.1-1 is based would require a license amendment request to revise the figure.

The amendment also revises TS Table 3.3.1-1, "Reactor Trip System Instrumentation," by relocating numerical values pertaining to Overtemperature ΔT and Overpower ΔT nominal RCS operating pressure, nominal T_{avg} , and constant (K) values to the COLR. Additionally, the basis for the RCS low flow reactor trip setpoints has been revised by referring to the COLR in the associated Bases document.

TS 5.6.5, "Core Operating Limits Report (COLR)," will be modified to reflect the above relocations to the COLR.

The proposed changes will allow Duke the flexibility of enhancing operating and core design margins without the need for cycle-specific license amendment requests. The relocation of these cycle-specific TS values to the COLR will result in a more complete COLR containing cycle-specific operating conditions and core reload related parameters. The safety and quality of operations at Duke's McGuire and Catawba nuclear stations will not be compromised by the implementation of this amendment request as TS 5.6.5(c) requires that all applicable limits of the safety analyses be met when generating cycle-specific requirements in the COLR.

Basis for Proposed Change

NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameters From Technical Specifications," dated October 4, 1988, provides guidance to licensees for the removal of cycle-dependent variables from the TS provided that these values are included in a COLR and are determined with NRC-approved methodologies referenced in the TS. Westinghouse Electric Company (Westinghouse) subsequently developed WCAP-14483, "Generic Methodology for Expanding Core Operating Limits Report," describing how cycle-specific

parameters may be relocated to the COLR. WCAP-14483 was accepted for referencing by the NRC on January 19, 1999. The Safety Evaluation Report, contained in the January 19, 1999 NRC letter approving WCAP-14483-A, concludes that additional information contained in the TS may be relocated to the COLR.

The limits on the parameters which are removed from the Technical Specifications and added to the COLRs must be developed or justified using NRC-approved methodologies. All accident analyses, performed in accordance with these methodologies, must meet the applicable NRC-approved limits of the safety analysis. The removal of parameter limits from the Technical Specifications and their addition to the COLRs does not obviate the requirement to operate within these limits. Furthermore, any changes to those limits must be performed in accordance with TS 5.6.5(c). If any of the applicable limits of the safety analyses are not met, prior NRC approval of the change is required, just as is the case for a license amendment request. For more routine modifications, where NRC-approved methodologies and limits of the safety analysis remain applicable, the potentially burdensome and lengthy process of amending the Technical Specifications may be avoided. The requested changes are essentially administrative in nature; therefore, the required level of safety will be maintained.

The requested changes are based upon NRC approved Westinghouse Owners Group (WOG) Technical Specifications Task Force (TSTF) TSTF-339, "Relocate TS Parameters to the COLR Consistent with WCAP-14483," Revision 2, and Westinghouse WCAP-14483-A. In accordance with these documents, previously approved RCS minimum total flow rates are retained in the TS to preclude the use of lower flow rates without prior NRC approval. The return to a lower RCS minimum flow rate, previously reviewed and approved by the NRC, is similar to that submitted by the Comanche Peak Nuclear Station on May 24, 1999, and approved by the NRC in the Safety Evaluation Report dated August 30, 1999. Consistent with the Comanche Peak submittal is a generic statement in TS 2.1.1 requiring compliance with the DNBR limit for the correlation(s) used for a specific core design. The same approach has also been made with respect to the CFM limits. Both of these statements more clearly address the requirements of 10CFR §50.36 by stating the

actual safety limits of DNB and CFM. Future changes to TS 2.1.1 are also minimized.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

As required by 10CFR50.91(a)(1), this analysis is provided to demonstrate that the proposed license amendment does not involve a significant hazard.

Conformance of the proposed amendment to the standards for a determination of no significant hazards, as defined in 10CFR50.92, is shown in the following:

- 1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The relocation of Reactor Coolant System (RCS) related cycle-specific parameter limits from the Technical Specifications (TS) to the Core Operating Limits Reports (COLR) proposed by this amendment request does not result in the alteration of the design, material, or construction standards that were applicable prior to the change. The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSARs. Therefore, the proposed amendment does not result in the increase in the probability or consequences of an accident previously evaluated.

- 2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the facility which should introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators.

3) Does the proposed change involve a significant reduction in margin of safety?

No. Implementation of this amendment would not involve a significant reduction in the margin of safety. Previously approved methodologies will continue to be used in the determination of cycle-specific core operating limits appearing in the COLRs. Additionally, previously approved RCS minimum total flow rates for McGuire and Catawba are retained in their respective TS so as to assure that lower flow rates will not be used without prior NRC approval. Consequently, no safety margins will be impacted.

Based on the above, it is concluded that the proposed license amendment request does not result in a reduction in margin with respect to plant safety.

Conclusion

Based on the preceding analysis, it is concluded that the relocation of Reactor Coolant System (RCS) related cycle-specific parameter limits from the TS to the COLR does not involve a Significant Hazards Consideration Finding as defined in 10CFR50.92.

ATTACHMENT 5

ENVIRONMENTAL ANALYSIS

ENVIRONMENTAL ANALYSIS

The proposed amendment has been reviewed against the criteria of 10CFR51.22 for environmental considerations. The proposed amendment does not involve a significant hazards consideration, increase the types and amounts of effluents that may be released offsite, or result in the increase of individual or cumulative occupational radiation exposures. Therefore, the proposed amendment meets the criteria provided by 10CFR51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.