Duke Power.
A Duke Energy Company

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June 25, 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

Duke License Amendment Request of March 24, 2003

On March 24, 2003 Duke Energy Corporation (Duke) submitted proposed amendments to the McGuire and Catawba Nuclear Station Facility Operating Licenses and Technical Specifications (TS) and Bases. On June 9, 2003 Duke personnel met with members of the NRC staff to discuss issues related to that submittal. This letter transmits a revised TS amendment package based on the results of that public meeting.

The proposed amendment reduces the required minimum measured Reactor Coolant System (RCS) flow rate from 390,000 gallons per minute (gpm) to a previously approved value of 382,000 gpm for McGuire Units 1 and 2, and Catawba Unit 1; and reduces the required minimum measured RCS flow rate from 390,000 gpm to the originally licensed value of 385,000 gpm for Catawba Unit 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

A001

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 3.3.1-1, "Reactor Trip System Instrumentation," by relocating Overtemperature 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 Table 3.4.1-1. 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 on NRC approval of 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.

During the June 9, 2003 meeting, Duke requested approval of the proposed changes by November 2003. Upon reconsideration, Duke would prefer that this license amendment request be approved in time to support the commencement of McGuire Unit 2, Cycle 16 operations currently scheduled to begin October 6, 2003. 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 revised amendment package are as follows:

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

The revisions provided in this letter do not change the conclusions reached in the original March 24, 2003 No Significant Hazards determination. Additionally, the basis for the categorical exclusion from the performance of an Environmental Assessment/Impact Review in the March 24, 2003 is unaffected by this revision.

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.

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

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: June 25, 2003
Date

Mary P. Dehms, Notary Public

My commission expires: JAN 22, 2006

SEAL

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K. L. Crane
R. L. Gill
Catawba Owners: NCMPA-1, SREC, PMPA, NCEMC
McGuire Master File (MG01DM)
Catawba Document Control File 801.01 (CN04DM)
ELL

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 ≥1.14 for the WRB-2M CHF correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained <5080 degrees F, decreasing 58 degrees F for every 10,000 MWd/mtU of fuel burnup.

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~~ LIMITS specified in ~~Figure 2.1.1-1 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.

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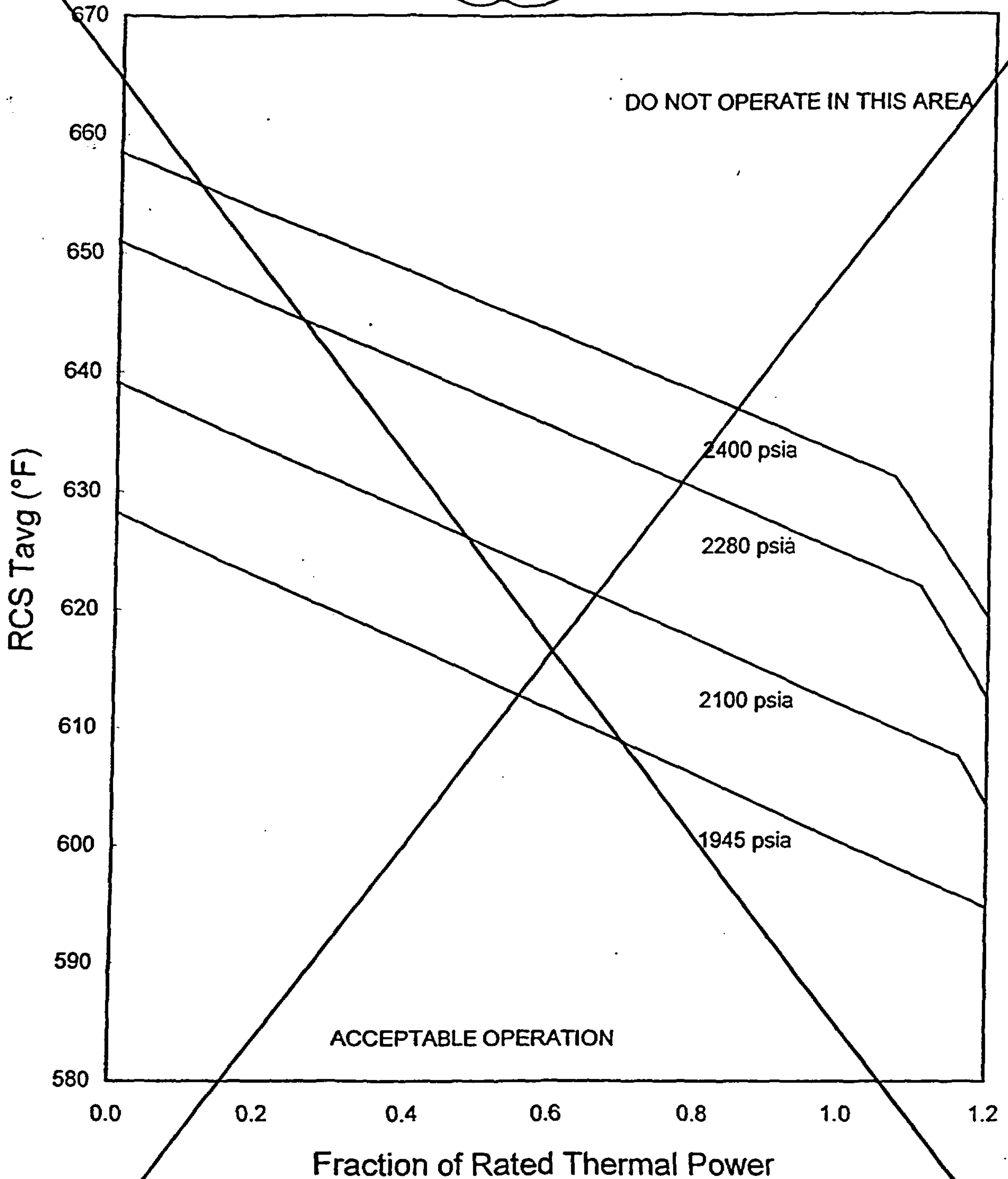

Figure 2.1.1-1

Reactor Core Safety Limits -
Four Loops in Operation

BASES

APPLICABLE SAFETY ANALYSES

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

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

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

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

BASES

FIGURE

THE COLR SHOWS

SAFETY LIMITS

FRACTION OF RATED THERMAL POWER,

The ~~curves~~ provided in ~~Figure B 2.1.1-1 show~~ the loci of points of ~~ΔT(%F),~~ 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 ($F_{\Delta H}$), 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 $F_1(\Delta I)$ 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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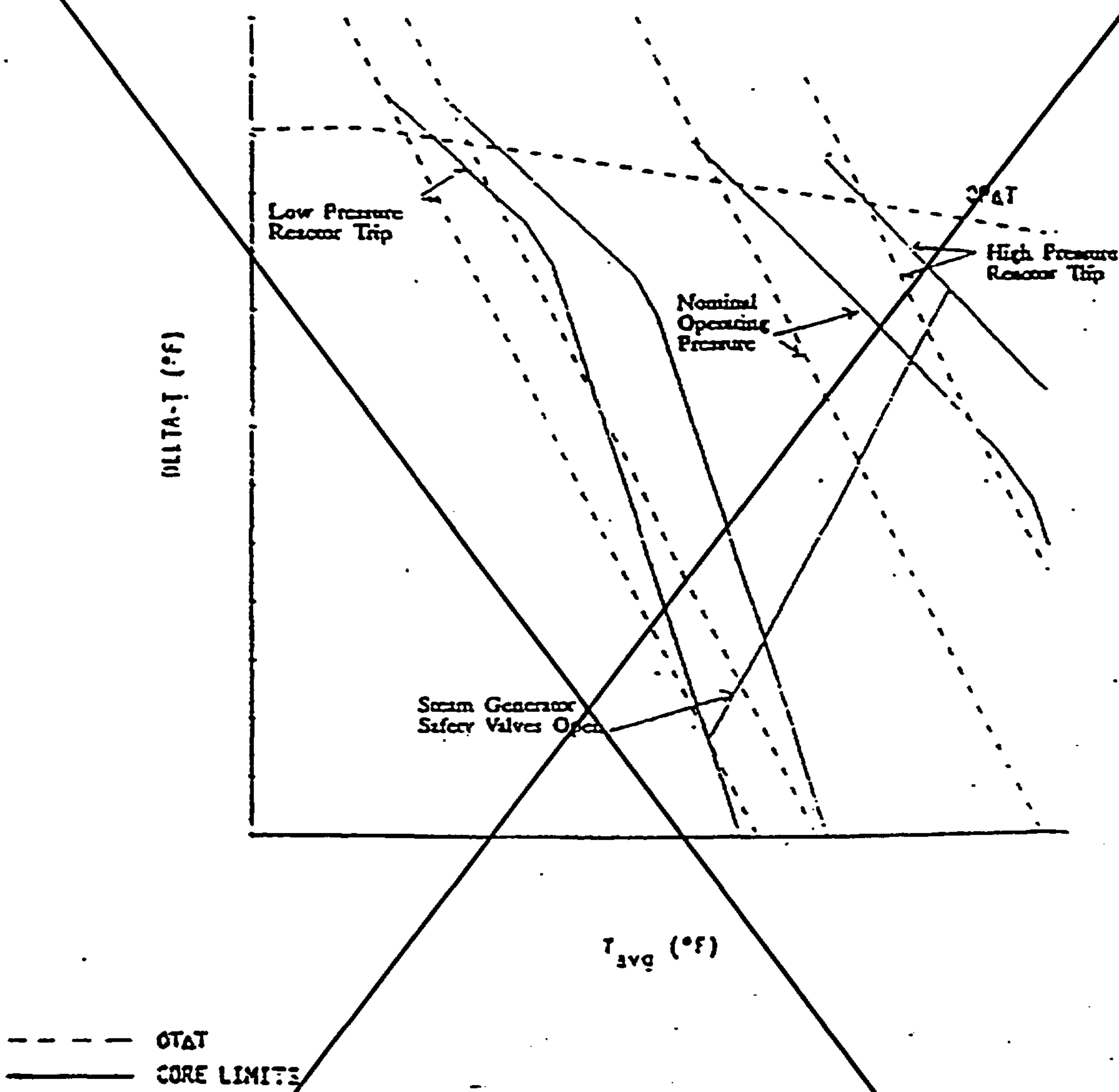

Figure B 2.1.1-1

Illustration of Overtemperature and Overpower ΔT Protection

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.

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

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 Tavg 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.

<u>Note 2: Overpower ΔT</u>

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

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

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, ≤ ~~585.1°F~~. (≤ THE VALUE SPECIFIED IN THE COLR)

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

K_5 = ~~0.02/°F~~ 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 $T \le T'$,

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)

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~~ THE minimum ~~measured~~ flow ~~of 97,500~~

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

THE

SPECIFIED IN THE COLR.

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.

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.

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 ~~< 390,000 gpm but ≥ 386,100 gpm~~ ≥ 99%, but < 100% of the limit 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
C. RCS total flow rate ~~< 386,100 gpm~~ < 99% OF THE VALUE SPECIFIED IN THE COLR.	C.1 Restore RCS total flow rate to ~~≥ 386,100 gpm~~ ≥ 99% OF THE VALUE SPECIFIED IN THE COLR.	2 hours
	OR	
	C.2.1 Reduce THERMAL POWER to < 50% RTP.	2 hours
	AND	
	C.2.2 Reduce the Power Range Neutron Flux - High Trip Setpoint to ≤ 55% RTP.	6 hours
	AND	
	C.2.3 Restore RCS total flow rate to ~~≥ 386,100 gpm~~ ≥ 99% OF THE VALUE SPECIFIED IN THE COLR.	24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within limits.	12 hours
SR 3.4.1.2	Verify RCS average temperature is within limits.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within limits.	12 hours
SR 3.4.1.4	Perform CHANNEL CALIBRATION for each RCS total flow indicator.	18 months

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

THE LIMIT SPECIFIED IN THE COLR

PARAMETER		INDICATION	No. OPERABLE CHANNELS	LIMITS
1.	Indicated RCS Average Temperature	meter	4	≤ ~~587.2°F~~
		meter	3	≤ ~~586.9°F~~
		computer	4	≤ ~~587.7°F~~
		computer	3	≤ ~~587.6°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			≥ ~~390,000 gpm~~

382,000 gpm AND GREATER THAN OR EQUAL TO THE LIMIT SPECIFIED IN THE COLR

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 1.7%~~ 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.

SPECIFIED IN THE COLR

The ~~LCO~~ numerical values ~~in Table 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.

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 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.

THE CONDITIONS WHICH DEFINE THE DNBR LIMIT

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.

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 ~~< 390,000 gpm but ≥ 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.

≥ 99%, BUT < 100% OF THE LIMIT SPECIFIED IN THE COLR,

BASES

ACTIONS (continued)

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

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

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

If the indicated RCS total flow rate is ~~< 386,100 gpm~~ then RCS total flow must be restored to ~~≥ 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 ≤ 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 ~~≥ 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:

ILLUSTRATION OF REACTOR CORE SAFETY LIMITS FOR SPECIFICATION 2.1.1

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

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3\. ~~2.~~ Shutdown Bank Insertion Limit for Specification 3.1.5,

4\. ~~3.~~ Control Bank Insertion Limits for Specification 3.1.6,

5\. ~~4.~~ Axial Flux Difference limits for Specification 3.2.3,

6\. ~~5.~~ Heat Flux Hot Channel Factor for Specification 3.2.1,

7\. ~~6.~~ Nuclear Enthalpy Rise Hot Channel Factor limits for Specification 3.2.2,

8\. ~~7.~~ Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

10\. ~~8.~~ Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

11\. ~~9.~~ Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

12\. ~~10.~~ Spent fuel pool boron concentration limits for Specification 3.7.14,

13\. ~~11.~~ 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 ≥1.14 for the WRB-2M CHF correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained <5080 degrees F, decreasing 58 degrees F for every 10,000 MWd/mtU of fuel burnup.

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 for Unit 1 and the original licensed value for Unit 2, 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 2.1.1-1 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 $\leq$ 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.

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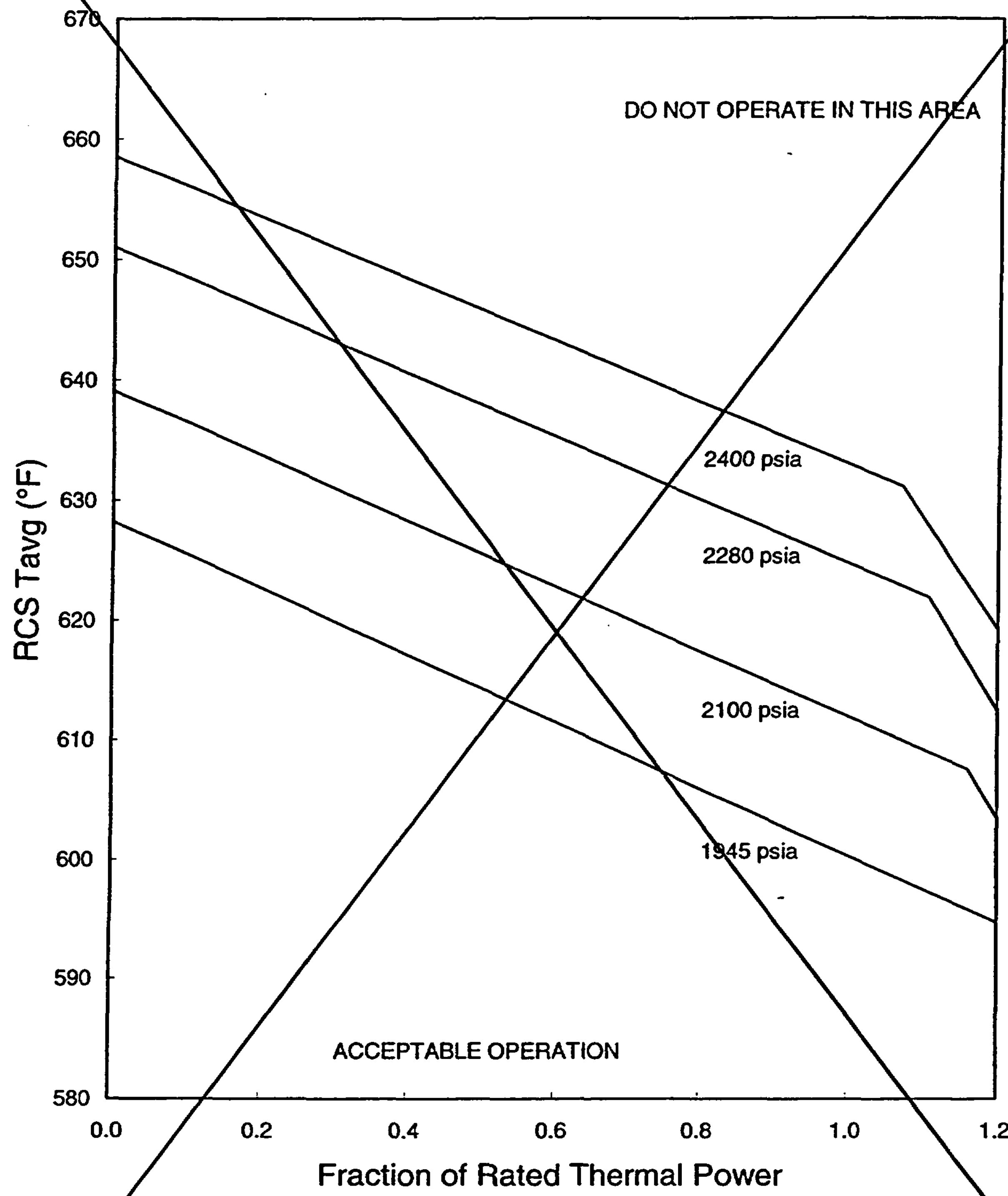

Figure 2.1.1-1
Reactor Core Safety Limits
Four Loops in Operation

BASES

APPLICABLE SAFETY ANALYSES

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

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

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

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

RCS FLOW RATE, ΔI,

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.

BASES

FIGURE   THE COLR 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 ~~ΔT, P~~, 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 Z

~~The curves in Figure 2.1.1-1 are based on a reference nuclear enthalpy rise hot channel factor ($F_{\Delta H}$), 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 $F_1(\Delta I)$ 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

EDITORIAL

1. 10 CFR 50, Appendix A, GDC 10.

2. UFSAR, Section 7.2.

~~3. DPC-NE-2011PA, March 1990.~~

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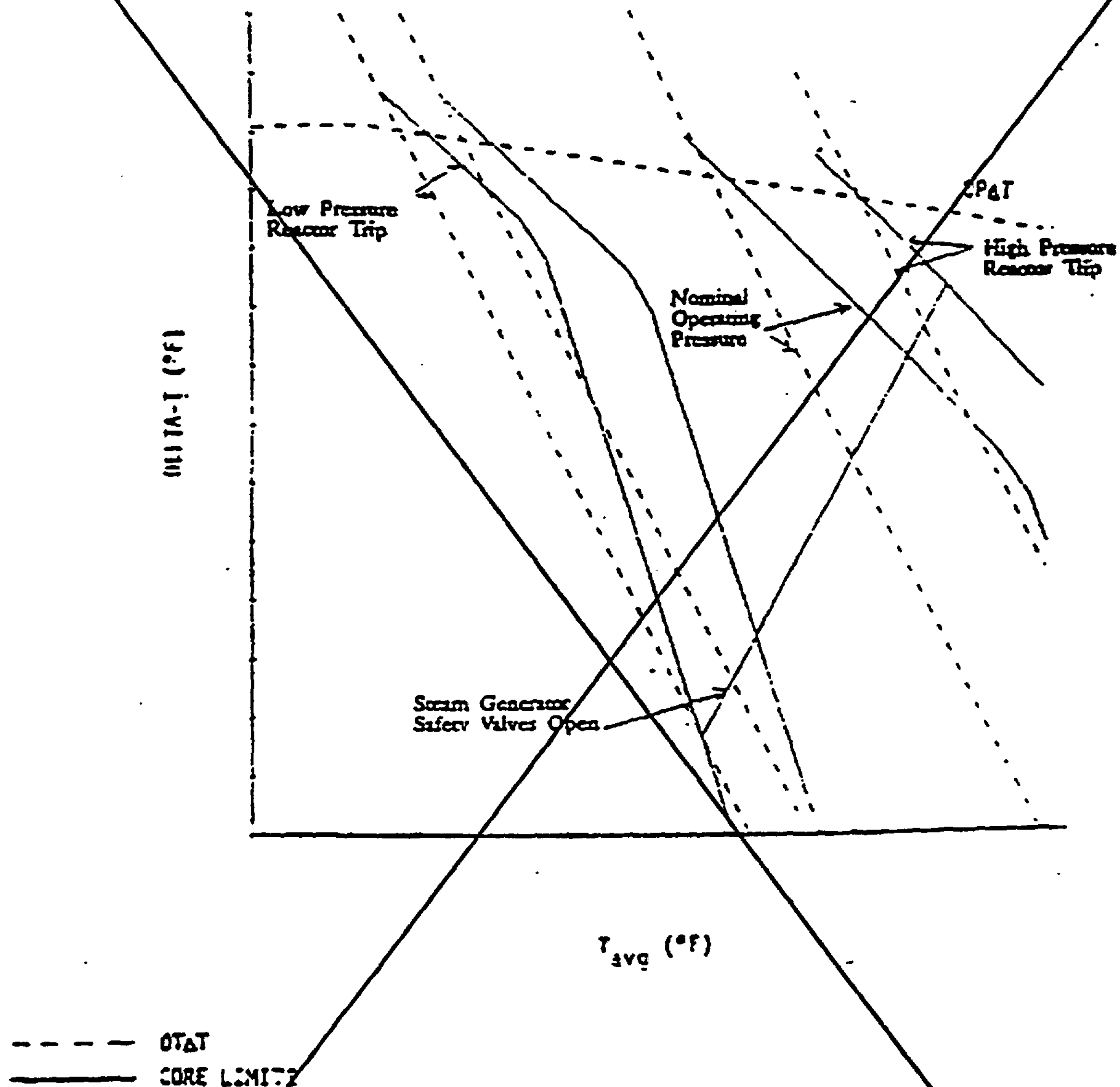

Figure B 2.1.1-1

Illustration of Overtemperature and Overpower ΔT Protection

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.

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

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), ~~≤ 585.1°F (Unit 1)~~ THE VALUES SPECIFIED IN THE COLR
~~≤ 590.8°F (Unit 2)~~
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 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.

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

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), ~~≤ 585.1°F (Unit 1) ≤ 590.8°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/°F~~ 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 $T \le T''$,

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)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

setpoints are based on ~~a~~ THE minimum ~~measured~~ flow ~~of 95,500 gpm (Unit 1) 96,250 gpm (Unit 2)~~ SPECIFIED IN THE COLR. 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. <u>Reactor Coolant Flow-Low (Two Loops)</u>

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~~ THE minimum ~~measured~~ flow ~~of 95,500 gpm (Unit 1) 96,250 gpm (Unit 2)~~ SPECIFIED IN THE COLR. 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.

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.

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 ~~< 390,000 gpm but ≥ 386,100 gpm.~~ ≥ 99%, BUT < 100% OF THE LIMIT 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
C. RCS total flow rate ~~< 386,100 gpm.~~ < 99% OF THE VALUE SPECIFIED IN THE COLR.	C.1 Restore RCS total flow rate to ~~≥ 386,100 gpm.~~ ≥ 99% OF THE VALUE SPECIFIED IN THE COLR.	2 hours
	OR	
	C.2.1 Reduce THERMAL POWER to < 50% RTP.	2 hours
	AND	
	C.2.2 Reduce the Power Range Neutron Flux - High Trip Setpoint to ≤ 55% RTP.	6 hours
	AND	
	C.2.3 Restore RCS total flow rate to ~~≥ 386,100 gpm.~~ ≥ 99% OF THE VALUE SPECIFIED IN THE COLR.	24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within limits.	12 hours
SR 3.4.1.2	Verify RCS average temperature is within limits.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is within limits.	12 hours
SR 3.4.1.4	Perform CHANNEL CALIBRATION for each RCS total flow indicator.	18 months

THE VALUE SPECIFIED IN THE COLR

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	≤ ~~587.1°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	≥ ~~2218.8 psig~~
		meter	3	≥ ~~2222.1 psig~~
		computer	4	≥ ~~2215.8 psig~~
		computer	3	≥ ~~2217.5 psig~~
3.	RCS Total Flow Rate			~~≥ 390,000 gpm~~

≥ 382,000 gpm AND GREATER THAN OR EQUAL TO THE LIMIT SPECIFIED IN THE COLR (UNIT 1), ≥ 385,000 gpm AND GREATER THAN OR EQUAL TO THE LIMIT SPECIFIED IN THE COLR (UNIT 2)

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. (INSERT 3) Operating within these limits will result in meeting the acceptance criteria, including the DNBR criterion.

RCS total flow rate contains a measurement error ~~of 2.1%~~ 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 2.2%~~ for no fouling.

(SPECIFIED IN THE COLR)

The ~~LCO~~ numerical values ~~in Table 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 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.

B.1 and B.2

≥ 99%, BUT < 100% OF THE LIMIT SPECIFIED IN THE COLR,

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 ~~< 390,000 gpm but ≥ 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.

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 ~~< 386,100 gpm~~, then RCS total flow must be restored to ~~≥ 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 ≤ 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 ~~≥ 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.

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. ~~1.~~ Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3.1.3,

3. ~~2.~~ Shutdown Bank Insertion Limit for Specification 3.1.5,

4. ~~3.~~ Control Bank Insertion Limits for Specification 3.1.6,

5. ~~4.~~ Axial Flux Difference limits for Specification 3.2.3,

6. ~~5.~~ Heat Flux Hot Channel Factor for Specification 3.2.1,

7. ~~6.~~ Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.2.2,

8. ~~7.~~ Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

9. Reactor Coolant System Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits for Specification 3.4.1,

10. ~~8.~~ Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

11. ~~9.~~ Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

12. ~~10.~~ Spent fuel pool boron concentration limits for Specification 3.7.15,

13. ~~11.~~ 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).

(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 JUSTIFICATON

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 for McGuire Units 1 and 2 and Catawba Unit 1; reduces the required minimum measured Reactor Coolant System (RCS) flow rate from 390,000 gpm to the originally licensed value of 385,000 gpm for Catawba Unit 2; 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

[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

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 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 Units 1 and 2, and Catawba Unit 1 from 390,000 gpm to 382,000 gpm; and reduces the required minimum measured RCS flow rate for Catawba Unit 2 from 390,000 gpm to the originally licensed value of 385,000 gpm, and further 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 2[3], and Catawba Unit 1[4] steam generator replacement license amendment requests of September 30, 1994.

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

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[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

minimum RCS total flow rates of 382,000 gpm for McGuire Units 1 and 2, and Catawba Unit 1; and 385,000 gpm for Catawba Unit 2, will be maintained in TS Table 3.4.1-1 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 more specific requirements regarding the safety limits (i.e. fuel DNB design basis and the fuel centerline melt design basis) conforming with WCAP-14483-A. 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.