Template = NRR 058

March 2, 2000

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: McGUIRE NUCLEAR STATION, UNITS 1AND 2 - ISSUANCE OF AMENDMENTS RE: (TAC NOS. MA5994 AND MA5995)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 191 to Facility Operating License NPF-9 and Amendment No. 172 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated June 24, 1999, as supplemented by letter dated November 24, 1999.

The amendments revise the minimum reactor coolant system (RCS) flow rate limit, reduce the reactor coolant average temperature and pressurizer pressure limits, restrict operation to a RCS flow deficit of no more than one percent, and change the low RCS flow reactor trip setpoint.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 191 to NPF-9
2. Amendment No. 172 to NPF-17
3. Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 2, 2000

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

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McGuire Nuclear Station

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. NPF-9

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Facility Operating License No. NPF-9 filed by the Duke Energy Corporation (licensee) dated June 24, 1999, as supplemented by letter dated November 24, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 191 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 2, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. NPF-17

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Facility Operating License No. NPF-17 filed by the Duke Energy Corporation (licensee) dated June 24, 1999, as supplemented by letter dated November 24, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 172 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 2, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

ATTACHMENT TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Appendix A Technical Specifications and Bases pages with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2.0-2	2.0-2
3.3.1-15	3.3.1-15
3.3.1-18	3.3.1-18
3.4.1-1	3.4.1-1
3.4.1-2	3.4.1-2
3.4.1-4	3.4.1-4
3.4.1-5	-
B 3.2.2-1	B 3.2.2-1
B 3.2.2-2	B 3.2.2-2
B 3.2.2-3	B 3.2.2-3
B 3.2.2-4	B 3.2.2-4
B 3.2.2-5	B 3.2.2-5
B 3.2.2-6	B 3.2.2-6
B 3.2.2-7	B 3.2.2-7
B 3.2.2-8	B 3.2.2-8
B 3.2.2-9	B 3.2.2-9
B 3.3.1-16	B 3.3.1-16
B 3.3.1-17	B 3.3.1-17
B 3.4.1-1	B 3.4.1-1
B 3.4.1-2	B 3.4.1-2
B 3.4.1-3	B 3.4.1-3
B 3.4.1-4	B 3.4.1-4

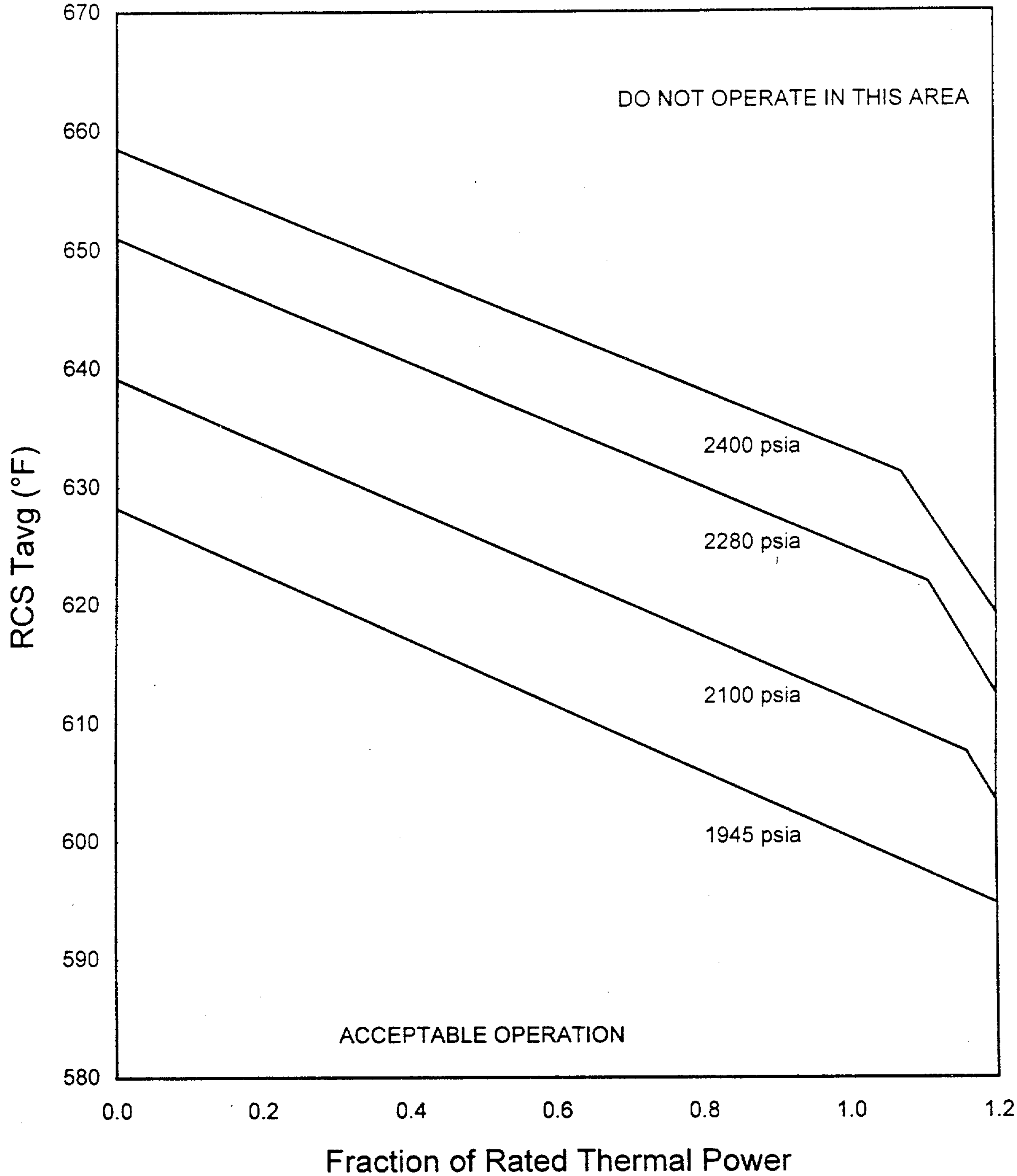

Figure 2.1.1-1

Reactor Core Safety Limits -
Four Loops in Operation

Table 3.3.1-1 (page 2 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16 SR 3.3.1.17	Refer to Note 1 (Page 3.3.1-18)	Refer to Note 1 (Page 3.3.1-18)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16 SR 3.3.1.17	Refer to Note 2 (Page 3.3.1-19)	Refer to Note 2 (Page 3.3.1-19)
8. Pressurizer Pressure						
a. Low	1(f)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1935 psig	≥ 1945 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2395 psig	≤ 2385 psig
9. Pressurizer Water Level - High	1(f)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93%	≤ 92%
10. Reactor Coolant Flow - Low						
a. Single Loop	1(g)	3 per loop	N	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 87%	≥ 88%
b. Two Loops	1(h)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 87%	≥ 88%
11. Undervoltage RCPs	1(f)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 5016 V	≥ 5082 V

(continued)

(f) Above the P-7 (Low Power Reactor Trips Block) interlock.

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

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 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, $\le$ 585.1 °F.

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

K_1 = Overtemperature ΔT reactor 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

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.	B.1 Reduce THERMAL POWER to ≤ 98% RTP.	2 hours
	<u>AND</u>	
	B.2 Reduce the Power Range Neutron Flux – High Trip Setpoint below the nominal setpoint by 2% RTP.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS total flow rate < 386,100 gpm.	C.1 Restore RCS total flow rate to ≥ 386,100 gpm. OR C.2.1 Reduce THERMAL POWER to < 50% RTP. AND C.2.2 Reduce the Power Range Neutron Flux - High Trip Setpoint to ≤ 55% RTP. AND C.2.3 Restore RCS total flow rate to ≥ 386,100 gpm.	2 hours 2 hours 6 hours 24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 2.	6 hours

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	≤ 587.2 °F
		meter	3	≤ 586.9 °F
		computer	4	≤ 587.7 °F
		computer	3	≤ 587.5 °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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}(X,Y)$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors, along with the other applicable LCOs, ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F_{\Delta H}(X,Y)$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F_{\Delta H}(X,Y)$ is a measure of the maximum total power produced in a fuel rod.

$F_{\Delta H}(X,Y)$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F_{\Delta H}(X,Y)$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F_{\Delta H}(X,Y)$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F_{\Delta H}(X,Y)$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency for transients that do not alter the core power distribution. The DNB design basis for operational transients and transients of moderate frequency preclude DNB and is met by limiting the minimum local DNB heat flux ratio to the design limit value using an NRC approved critical heat flux (CHF) correlation. Operation transients and transients of moderate frequency that are DNB limited are assumed to begin with an $F_{\Delta H}(X,Y)$ value that satisfies the LCO requirement, with the exception of accidents such as the uncontrolled RCCA bank withdrawal

BACKGROUND (continued)

(UCBW). For these types of accidents, the event itself causes changes in the power distribution and this LCO alone is not sufficient to preclude DNB. The acceptability of analyses such as the UCBW accident analysis is ensured by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits," in combination with cycle-specific analytical calculations."

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs.

APPLICABLE SAFETY ANALYSES

Limits on $F_{\Delta H}(X,Y)$ preclude core power distributions that exceed the following fuel design limits:

a. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

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

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

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

For transients that may be DNB limited, the Reactor Coolant System flow and $F_{\Delta H}(X,Y)$ are the core parameters of most importance. The limits on $F_{\Delta H}(X,Y)$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency that do not alter the core power distribution. For transients such as uncontrolled RCCA bank withdrawal, which are characterized by changes in the core power distribution, this LCO alone is not sufficient to preclude DNB. The acceptability of the accident analyses is ensured by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.4, "QUADRANT POWER TILT RATIO

APPLICABLE SAFETY ANALYSES (continued)

(QPTR)," and LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits," in combination with cycle-specific analytical calculations. The DNB design basis is met by limiting the minimum DNBR to the design limit value using an NRC approved CHF correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F_{\Delta H}(X,Y)$ limit increases with decreasing power level. This functionality in $F_{\Delta H}(X,Y)$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F_{\Delta H}(X,Y)$ in the analyses.

The LOCA safety analysis models $F_{\Delta H}(X,Y)$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3). The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$)."

$F_{\Delta H}(X,Y)$ and $F_Q(X,Y,Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Control Bank Insertion Limits.

$F_{\Delta H}(X,Y)$ satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

$F_{\Delta H}(X,Y)$ shall be limited by the following relationship:

$$F^{M}_{\Delta H}(X,Y) \le F^{L}_{\Delta H}(X,Y)^{LCO}$$

where: $F^{M}_{\Delta H}(X,Y)$ is defined as the measured radial peak, and

$F^{L}_{\Delta H}(X,Y)^{LCO}$ is defined as the steady state maximum allowable radial peak defined in the COLR.

LCO (continued)

The $F^{L}_{\Delta H}(X,Y)^{LCO}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB.

$F^{L}_{\Delta H}(X,Y)^{LCO}$ limits are maximum allowable radial peak (MARP) limits which are developed in accordance with the methodology outlined in Reference 5. MARP limits are constant DNBR limits which are a function of both the magnitude and location of the axial peak F(Z), therefore, justifying the X,Y dependence of the $F^{L}_{\Delta H}(X,Y)^{LCO}$ limit.

The limiting value, $F^{L}_{\Delta H}(X,Y)^{LCO}$, is also power dependent and can be described by the following relationship:

$$F^{L}_{\Delta H}(X,Y)^{LCO} = MARP(X,Y) * [1.0 + (1/RRH) * (1.0 - P)]$$

where: MARP(X,Y) is the maximum allowable radial peaks provided in the COLR,

P is the ratio of THERMAL POWER to RATED THERMAL POWER, and

RRH is the amount by which allowable THERMAL POWER must be reduced for each 1% that $F^{M}_{\Delta H}(X,Y)$ exceeds the limit. The specific value is contained in the COLR.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value, $F^{L}_{\Delta H}(X,Y)^{LCO}$, is allowed to increase approximately 0.3% for every 1% RTP reduction in THERMAL POWER. This increase in the $F^{L}_{\Delta H}(X,Y)^{LCO}$ limit is due to the reduced amount of heat removal required at lower powers.

APPLICABILITY

The $F_{\Delta H}(X,Y)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that might be expected to be sensitive to $F_{\Delta H}(X,Y)$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F_{\Delta H}(X,Y)$ in these modes. The exceptions to this are the steam line break, APPLICABILITY (continued)

uncontrolled RCCA bank withdrawal from zero power and rod ejection from zero power events, which are assumed, for analysis purposes, to occur from very low power levels. At these low power levels, measurements of $F_{\Delta H}$ are not sufficiently reliable. Operation within analysis limits at these conditions is inferred from startup physics testing verification of design predictions of core parameters in general.

ACTIONS

A.1

If $F^M_{\Delta H}(X,Y)$ is not within limit, THERMAL POWER must be reduced at least RRH% from RTP for each 1% $F_{\Delta H}(X,Y)$ exceeds the limit. Reducing power increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The Completion Time of 2 hours provides an acceptable time to reach the required power level without allowing the plant to remain in an unacceptable condition for an extended period of time.

Condition A is modified by a Note that requires that Required Actions A.3.2.2 and A.4 must be completed whenever Condition A is entered. Thus, if compliance with the LCO is restored, Required Action A.3.2.2 and A.4 nevertheless requires another measurement and calculation of $F_{\Delta H}(X,Y)$ in accordance with SR 3.2.2.1.

A.2.1 and A.2.2

Upon completion of the power reduction in Required Action A.1, the unit is allowed an additional 6 hours to restore $F_{\Delta H}(X,Y)$ to within its RTP limits. This restoration may, for example, involve realigning any misaligned rods enough to bring $F_{\Delta H}(X,Y)$ within its limit. When the $F_{\Delta H}(X,Y)$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F_{\Delta H}(X,Y)$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 8 hours provides an acceptable time to restore $F_{\Delta H}(X,Y)$ to within its RTP limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

If the value of $F_{\Delta H}(X,Y)$ is not restored to within its specified RTP limit, the alternative option is to reduce the Power Range Neutron Flux—High Trip Setpoint ≥ RRH% for each 1% $F^M_{\Delta H}(X,Y)$ exceeds the limit in accordance with Required Action A.2.2. The reduction in trip setpoints ensures that

ACTIONS (continued)

continuing operation remains at an acceptable low power level with adequate DNBR margin and limits the consequences of a transient by limiting the transient power level which can be achieved during a postulated event.

The allowed Completion Time of 8 hours to reset the trip setpoints per Required Action A.2.2 recognizes that, once power is 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.

A.3.1, A.3.2.1, and A.3.2.2

If $F^M_{\Delta H}(X,Y)$ was not restored to within the RTP limits, and the Power Range Neutron Flux-High Trip Setpoints were subsequently reduced, an additional 64 hours are provided to restore $F^M_{\Delta H}(X,Y)$ within the limit for RTP. Alternatively, the Overtemperature ΔT setpoint (K_1 term) must be reduced by $\geq$ TRH for each 1% $F^M_{\Delta H}(X,Y)$ exceeds the limit. TRH is the amount of overtemperature ΔT K_1 setpoint reduction required to compensate for each 1% that $F^M_{\Delta H}(X,Y)$ exceeds the limit and is provided in the COLR. This action ensures that protection margin is maintained in the reduced power level for DNB related transients not covered by the reduction in the Power Range Neutron Flux-High Trip Setpoint. Once the Overtemperature ΔT Trip Setpoint has been reduced per Required Action A.3.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F_{\Delta H}(X,Y)$ verified not to exceed the allowed limit at the lower power level.

The unit is provided 64 additional hours to perform these tasks over and above the 8 hours allowed by either Action A.2.1 or Action A.2.2. The Completion Time of 72 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 72 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F_{\Delta H}(X,Y)$.

A.4

Verification that $F_{\Delta H}(X,Y)$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F_{\Delta H}(X,Y)$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO

ACTIONS (continued)

limit. This Action demonstrates that the $F_{\Delta H}(X,Y)$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1 through A.4 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 and SR 3.2.2.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F^M_{\Delta H}(X,Y)$ is within the specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which it was last verified to be within specified limits. Because $F^M_{\Delta H}(X,Y)$ could not have previously been measured in this reload core, power may be increased to RTP prior to an equilibrium verification of $F_{\Delta H}(X,Y)$ provided nonequilibrium measurements of $F_{\Delta H}(X,Y)$ are performed at various power levels during startup physics testing. This ensures that some determination of $F_{\Delta H}(X,Y)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. The Frequency condition is not intended to require verification of the parameter after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_{\Delta H}(X,Y)$ was last measured.

SR 3.2.2.1

The value of $F^M_{\Delta H}(X,Y)$ is determined by using the movable incore detector system to obtain a flux distribution map at any THERMAL POWER greater than 5% RTP. A computer program is used to process

SURVEILLANCE REQUIREMENTS (continued)

the measured 3-D power distribution to calculate the steady state $F^{L}_{\Delta H}(X,Y)^{LCO}$ limit which is compared against $F^{M}_{\Delta H}(X,Y)$.

$F^{M}_{\Delta H}(X,Y)$ is verified at power levels $\geq$ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F^{M}_{\Delta H}(X,Y)$ is within its limit at high power levels.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F_{\Delta H}(X,Y)$ limit cannot be exceeded for any significant period of operation.

SR 3.2.2.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_{\Delta H}(X,Y)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values is a limit called $F^{L}_{\Delta H}(X,Y)^{SURV}$. This Surveillance compares the measured $F^{M}_{\Delta H}(X,Y)$ to the Surveillance limit to ensure that safety analysis limits are maintained.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F^{M}_{\Delta H}(X,Y)$ is evaluated and found to be within its surveillance limit, an evaluation is required to account for any increase to $F^{M}_{\Delta H}(X,Y)$ that may occur and cause the $F_{\Delta H}(X,Y)^{SURV}$ limit to be exceeded before the next required $F_{\Delta H}(X,Y)^{SURV}$ evaluation.

In addition to ensuring via surveillance that the enthalpy rise hot channel factor is within its steady state and surveillance limits when a measurement is taken, there are also requirements to extrapolate trends in both the measured hot channel factor and in its surveillance limit. Two extrapolations are performed for this limit:

1. The first extrapolation determines whether the measured enthalpy rise hot channel factor is likely to exceed its surveillance limit prior to the next performance of the SR.

2. The second extrapolation determines whether, prior to the next performance of the SR, the ratio of the measured enthalpy rise hot

BASES

SURVEILLANCE REQUIREMENTS (continued)

channel factor to the surveillance limit is likely to decrease below the value of that ratio when the measurement was taken.

Each of these extrapolations is applied separately to the enthalpy rise hot channel factor surveillance limit. If both of the extrapolations are unfavorable, i.e., if the extrapolated factor is expected to exceed the extrapolated limit and the extrapolated factor is expected to become a larger fraction of the extrapolated limit than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F^M_{\Delta H}(X,Y)$ limit with the last $F^M_{\Delta H}(X,Y)$ increased by the appropriate factor as specified in the COLR, or to evaluate $F^M_{\Delta H}(X,Y)$ prior to the point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F^M_{\Delta H}(X,Y)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_{\Delta H}(X,Y)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending.

$F^M_{\Delta H}(X,Y)$ is verified at power levels 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F^M_{\Delta H}(X,Y)$ is within its limit at high power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F^M_{\Delta H}(X,Y)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day st

B 3.2.2-9

Rev. 2

REFERENCES

1. UFSAR Se

2. 10 CFR 5

3. 10 CFR 5

4. 10 CFR 50.36, Technical Spec... s, (c)(2)(ii).

5. DPC-NE-2004P-A, Rev. 1, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," SER Dated February 20, 1997 (DPC Proprietary)

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 97,500

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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 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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS volumetric flow rate normally remains constant during an operational fuel cycle with all pumps running. Flow rate indications are averaged within a loop and then summed among the four loops to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits. RCS flow rate may be slightly reduced provided THERMAL POWER is also reduced to ensure that the calculated DNBR will not be below the design DNBR value.

Operation outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the acceptance criteria, including the DNBR criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be

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)."

The pressurizer pressure limits and the RCS average temperature limits correspond to analytical limits of 2205 psig 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.

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.

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.

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

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.

BASES

ACTIONS (continued)

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

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated June 24, 1999 (Ref.1), Duke Energy Corporation (DEC, the licensee), submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would increase the minimum reactor coolant system (RCS) total flow rate limit to 390,000 gpm to provide more margin in the core design limits. The increase of the RCS flow is realized as a result of: (1) the use of an approved elbow tap RCS flow measurement method which eliminated impacts of hot leg streaming on the RCS flow measurement and (2) the steam generator replacement for the McGuire units. Also, the licensee has proposed to: (1) revise the reactor coolant average temperature and pressurizer pressure limits to be consistent with the assumptions made in the re-analyses of the design basis transients and accidents in Chapter 15 of Updated Final Safety Analysis Report (UFSAR); (2) revise the safety limit curves; and (3) restrict plant operation with a RCS flow deficit of no more than 1%. Further, the licensee has proposed to reduce the set-points of the low RCS flow reactor trip function, correct a typographic error in the McGuire TS, and make appropriate changes to the Bases Section.

Specifically, the proposed TS changes are as follows:

(1) Table 3.4.1-1, "RCS DNB Parameters," in LCO 3.4.1 will be revised by specifying the minimum RCS total flow rate limit of 390,000 gpm to replace Figure 3.4.1-1, which specifies the RCS total flow rate versus rated thermal power (RTP) for four loop operation. The RCS average temperature and pressurizer pressure limits in Table 3.4.1-1 will be revised. LCO 3.4.1 Actions B and C will be revised to allow for operation with a RCS flow deficit of no more than one percent at no more than 98% RTP, replacing Figure 3.4.1-1. Figure 3.4.1-1 is deleted. TS Bases B3.4.1 will also be revised to reflect the proposed changes.

The Safety Limit curves in Figure 2.1.1-1 will be revised due to the proposed change in minimum RCS total flow rate limit to 390,000 gpm.

(3) The trip setpoint and allowable values for Function 10, "RCS Flow - Low," in Table 3.3.1-1, "Reactor Trip System Instrumentation," will be reduced from 91% and 90% to 88% and 87%, respectively. TS Bases B3.3.1 will be revised to reflect these changes. The licensee also proposed to correct a typographical error in Note 1 of the Overtemperature ΔT trip function of Table 3.3.1-1, where T', the nominal T-average at RTP, will be changed from < 585.1°F to ≤ 585.1°F.

In a letter dated November 24, 1999 (Ref. 8), DEC, in response to staff's questions raised during a meeting on November 16, 1999, made changes to the TS Bases and provided clarifying information that did not change the scope of the June 24, 1999, application and the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

Current McGuire TS Figure 3.4.1-1, "RCS Total Flow Rate Versus Rated Thermal Power - Four Loops In Operation," specifies the relationships of the RCS flow rates versus rated thermal power (RTP) for the "permissible" and "restricted" regions of reactor operation. The permissible operation region requires the RCS flow to be greater than 382,000 gpm for operation up to 102% RTP. The restricted operation region allows for reactor operation with a RCS flow deficit of up to 5% with a power reduction of 2% for every percent of RCS flow reduction. The proposed TS changes will delete Figure 3.4.1-1 and replace it with (1) a specification of the minimum RCS total flow rate limit of 390,000 gpm in Table 3.4.1-1, "RCS DNB Parameters," and (2) a revision of LCO 3.4.1 Actions B and C to allow for operation with a RCS flow deficit of no more than 1% at no more than 98% RTP.

The RCS flow rate is an important parameter in the reactor operation to assure that the departure from nucleate boiling ratio (DNBR) limit is not exceeded during normal operation and anticipated operational occurrences (AOOs). The proposed change to increase the RCS flow to 390,000 gpm would improve margin with regard to the DNBR. Also, as discussed in Section 2.1 of this report, increasing the minimum RCS flow limit has no or negligible effect on LOCA blowdown forces and containment functional design.

The current TSs allow for reactor operation with a RCS flow deficit of up to 5% with a reactor power reduction of a stair-step tradeoff of 2% power per 1% flow. The licensee's recent engineering evaluations raised concerns regarding the validity of the power/flow stair-step tradeoff for a RCS flow deficit of up to 5% because the original safety analyses justifying the reduced flow operation had not considered all possible initiating conditions for design basis transients. Therefore, the licensee proposed to reduce the allowable RCS flow deficit from 5 percent to 1 percent, subject to the same power/flow tradeoff. This proposed change to restrict operation to a RCS flow deficit of no more than 1% is more restrictive and conservative. As discussed in Section 2.2 of this report, the adequacy of reactor operation with a RCS flow deficit of 1% and 98% RTP is verified by specifically using them as initial conditions in the reanalyses of UFSAR Chapter 15 design basis transients.

The licensee also proposed to revise the reactor coolant average temperatures and pressurizer pressures in Table 3.4.1-1 with slightly lower values to be consistent with the assumptions made in the re-analyses of the UFSAR Chapter 15 design basis events. Lowering the RCS average temperature would improve thermal margin, whereas lower pressurizer pressure would have an adverse effect on DNBR. However, the overall acceptability of these changes are verified with the Chapter 15 safety analyses.

The safety limit curves in TS Figure 2.1.1-1, "Reactor Core Safety Limits - Four Loops in Operation," are revised. The safety limit curves show the loci of points of thermal power, reactor coolant system pressure and average temperature below which (1) the calculated DNBR is not less than the design DNBR limit or (2) the average enthalpy at the vessel exit is less than or equal to the enthalpy of the saturated liquid. These curves are revised slightly higher as a result of the increase in the RCS flow limit to 390,000 gpm. The safety limits do not affect the normal operation of the facility, but are only used to determine the need for further safety evaluations following postulated overheating or power excursion transients. Since the axial flux difference limits of LCO 3.2.3 are unchanged, all the current thermal hydraulic design criteria continue to be satisfied. The licensee has determined that the constants in the overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) setpoint equations are conservative and provide the necessary reactor protection. These OTΔT and OPΔT reactor trips are assumed in the Chapter 15 analyses.

The licensee also proposed to reduce the flow trip setpoint and allowable values for the "RCS Flow - Low" reactor trip in Table 3.3.1-1 of the McGuire TS from 91% to 88% and from 90% to 87%, respectively. These changes are intended to preclude spurious reactor trips that might occur following the increase in the minimum RCS total flow rate limit due to normal flow noise and the lower flow indication historically observed in Loop A at McGuire Unit 1. The RCS low flow trip functions in the mitigation of the partial loss of forced RCS flow and the reactor coolant pump shaft seizure (locked rotor) accidents. To verify the acceptability of the reduced Low RCS Flow setpoint and allowable values, those transients and accidents, such as partial loss of forced reactor coolant flow transient and the reactor coolant pump shaft seizure accident, which rely on the low RCS flow trip for reactor protection are re-analyzed with the reduced trip setpoint value.

In summary, the acceptability of these changes are verified through (1) the evaluation of relevant UFSAR chapters that may be affected by the increased RCS flow and (2) the re-analyses of the UFSAR Chapter 15 design basis events to demonstrate compliance with the acceptance criteria, including the DNBR limit and the overpressure limit. The licensee provided these evaluations in Attachment 3, "Description of the Proposed Changes and Technical Justification," of the June 24, 1999, letter (Ref. 1). The staff has reviewed this information as summarized below.

2.1 LOCA Blowdown Force and Containment Analyses

Because the revised minimum RCS flow rate limit of 390,000 gpm is within the upper and lower bounds of 420,000 and 382,000 gpm, respectively, supported by the existing LOCA blowdown forces analyses on the reactor vessel and loop in UFSAR Section 3.6 and 3.9, it would have no adverse effect on vessel internals, core components, and coolant loop piping structural adequacy. With regard to the containment functional design, the increase in the minimum total RCS flow limit and a small change in RCS temperature limit will have no or negligible effect on the assumptions used in the containment analyses with respect to (1) the mass and energy release analyses for postulated LOCA and secondary system pipe rupture inside containment and (2) the minimum containment pressure analysis for performance capability studies of emergency core cooling system.

2.2 UFSAR Chapter 15 Accident Analyses

The licensee performed re-analyses of pertinent UFSAR Chapter 15 design basis transients and accidents to demonstrate that the increase in the TS minimum RCS total flow rate, the reduced pressurizer pressure and average reactor coolant temperature, the reduced RCS Flow - Low trip setpoint, and the operation with a RCS deficit of 1% and 98% RTP will not have any adverse impact on any of the design basis event analyses. Some of the UFSAR Chapter 15 transients and accidents transients were not re-analyzed. This is because either (1) the analysis is unaffected by the TS changes, (2) the transient is non-limiting and any changes will have a favorable positive impact on the analysis results, or (3) the transient is bounded by a more limiting transient of the same ANS Condition which is being reanalyzed. For example, In the event categories of "increase in reactor coolant inventory" and "decrease in reactor coolant inventory," it is found that each event is either bounded by other transients or the RCS flow rate has no effect because it is not explicitly assumed in the analysis.

The re-analyses of the design basis events were performed with the approved methods described in DPC-NE-3000-PA (Ref. 2), DPC-NE-3001-PA (Ref. 3), DPC-NE-3002-A (Ref. 4), and DPC-NE-2005-A (Re. 5). The hot channel DNBR was calculated with the BWUZ CHF correlation and the statistical core design (SCD) methodology (Ref. 5) with the BWUZ correlation design DNBR limit of 1.31 and the SCD design limit of 1.5. The initial conditions of the RCS flow, temperature, and pressure assumed in the analyses were consistent with the proposed TS changes, and the reduced trip setpoint value for the low-RCS flow trip was used for partial loss of force flow and locked rotor events. The impact of the operation with 99% RCS flow and 98% RTP was also specifically evaluated. The evaluation is summarized below.

In the "increase in heat removal by the secondary system" category of events, the reanalyses were made for (1) feedwater system malfunction causing an increase in feedwater flow, (2) excessive increase in secondary steam flow, and (3) steam system piping failure. The results showed that the calculated minimum DNBRs were well above the SCD design limit or the BWUZ design DNBR limit. In the case of excessive increase in secondary steam flow, the reactor reached an equilibrium condition that did not challenge the OTΔT or OPΔT reactor trip functions which are designed to protect the core against DNB.

In the "decrease in heat removal by the secondary system" category of events, the reanalyses were made for (1) turbine trip, (2) loss of normal feedwater, and (3) feedwater system pipe break. The results showed that the minimum DNBRs calculated for these events were above the design DNBR limit, and the peak primary pressures were below the acceptance criterion of 110% of the design pressure. In addition, the adequacy of the long-term core cooling capability is verified by the prevention of hot leg boiling with a sufficient minimum sub-cooling reached during the post-trip overheating phase of the transients.

In the event category of "decrease in reactor coolant system flow rate," the reanalyses were made for (1) both partial and complete loss of forced reactor coolant flow and (2) a reactor coolant pump shaft seizure. These transients were analyzed with the revised Low RCS flow reactor trip setpoint in accordance with the proposed TS changes for the McGuire units. The minimum DNBRs calculated for both the partial and complete loss of coolant flow events were well above the SCD design limit. For the pump shaft seizure event, the calculated peak primary pressure was found to be significantly below the acceptance criterion of 110 percent of the design pressure. The minimum DNBR calculate with a standard axial power shape was found to fall below the 1.50 SCD design limit. Because the pump shaft seizure is an ANS Condition IV event, the acceptance criterion is that fuel failure percentage must be low enough to ensure that the radiological consequences do not exceed a small fraction of the 10 CFR Part 100 guidelines. The maximum allowable radial peaking (MARP) curves were generated in order to determine the number of fuel rods, if any, that experience DNB and are therefore assumed to experience fuel failure. These revised MARP curves allow greater radial peaking for all axial peaks and locations than the existing MARP curves. Therefore, the fuel failure assumption in the current offsite dose calculation remains valid.

In the event category of "reactivity and power distribution anomalies," the analyses were made for (1) uncontrolled RCCA bank withdrawal from a subcritical condition, (2) RCCA bank withdrawal at power, (3) dropped RCCA rod, (4) single uncontrolled rod withdrawal, and (5) a spectrum of RCCA rod ejection. The minimum DNBRs calculated for bank withdrawal from subcritical condition, dropped RCCA rod, and single rod withdrawal were above the 1.50 SCD design limit, and the calculated peak primary pressure is below 110 percent of the design pressure.

For the RCCA ejection accidents, the peak fuel pellet enthalpy is determined to be significantly below the acceptance criterion of 280 cal/gm. The calculated peak primary pressure is 2693.7 psig, which is below the acceptance criterion. Since the minimum DNBR calculated with a standard axial power shape was found to fall below the 1.31 design limit, MARP curves were generated in order to determine the number of fuel rods, if any, that experience DNB. The approach for generating MARP limits was described in topical report DPC-NE-3001-PA (Ref. 3) (approved by the NRC) as applicable to the control rod ejection analysis. The revised MARP curves allow greater radial peaking for all axial peaks and locations than the existing MARP curves. Therefore, the fuel failure assumption in the current offsite dose calculation remains valid.

For the uncontrolled RCCA bank withdrawal at power, the calculated peak primary and secondary pressures were below the acceptance criteria of 110% of the corresponding design pressures. Since the minimum DNBR calculated with a standard axial power shape was found to fall below the 1.50 design limit, the licensee generated MARP curves in order to determine the number of fuel rods, if any, that experience DNB. The use of the MARP approach for the uncontrolled RCCA bank withdrawal event at power were presented in a meeting on October 7 and 8, 1991, with the NRC staff (Ref. 6) as a part of NRC review for DPC-NE-3002 (Ref. 4) and the McGuire Station Cycle 8 reload, which was approved (Ref. 7). In response to a staff question, the licensee in Attachments 3a and 3b of its November 24, 1999, letter (Ref. 8) described the technical details and bases for the use of the MARP approach to determine whether the DNBR limit was exceeded. The staff has reviewed the MARP analysis of the uncontrolled RCCA bank withdrawal event and agreed with the licensee's conclusion that the revised MARP curves allow greater radial peaking for all axial peaks and locations and that no fuel failures occur. In Reference 8, the licensee also revised the McGuire TS Bases and UFSAR. These revisions are necessary to provide (1) clarification for discussions on transients that can cause changes in the power distribution in the Bases for LCO 3.2.2 and (2) clarification for initial conditions and power distribution assumed in the accident analysis. The staff has reviewed these changes and found them acceptable.

The staff has reviewed and evaluated the licensee's proposed TS changes to increase the minimum RCS flow limit, reduce the reactor coolant average temperature and pressurizer pressure limits, restrict reactor operation to a RCS flow deficit of 1% at 98% RTP, and reduce the RCS flow - Low reactor trip setpoint. The staff finds that the proposed TS changes would not exceed any acceptance criteria and are therefore acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding [64 FR 43772]. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: March 2, 2000

6.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to USNRC, "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 and 50-414, Proposed Amendments to Technical Specifications (TS), TS 2.0 - Safety Limits, TS 3.3.1 - Reactor Trip System Instrumentation, TS 3.4.1 - Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," June 24, 1999.

2. DPC-NE-3000-PA, "Duke Power Company, Oconee Nuclear Station, McGuire Nuclear Station, Catawba Nuclear Station, Thermal-Hydraulic Transient Analysis Methodology," Duke Power Company, Revision 2 (SER dated October 14, 1998).

3. DPC-NE-3001-PA, "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," Duke Power Company, November 1991.

4. DPC-NE-3002-A, "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, FSAR Chapter 15 System Transient Analysis Methodology," Duke Power Company, Revision 3, May 19, 1999.

5. DPC-NE-2005-A, "Duke Power Company, Thermal-Hydraulic Statistical Core Design Methodology," Duke Power Company, Revision 1, November 1996.

6. Letter from M. S. Tuckman, Duke Power Company, to USNRC, "McGuire Nuclear Station, Docket Numbers 50-369 and -370, Catawba Nuclear Station, Docket Numbers 50-413 and -414, Oconee Nuclear Station, Docket Numbers 50-269, -270, and 287, Handouts Presented in October 7 and 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.

7. Letter from Timothy A. Reed, USNRC, to T. C. McMeekin, Duke Power Company, "Issuance of Amendment No. 128 to Facility Operating License NPF-9 and Amendment No. 110 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2 (TAC No. 80694)," November 27, 1991, Docket Nos. 50-369 and 50-370.

8. Letter from M. S. Tuckman, Duke Power Company, to USNRC, "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 and 50-414, Request for Additional Information Regarding Proposed Amendments to Technical Specifications TS 2.0 - Safety Limits, TS 3.3.1 - Reactor Trip System Instrumentation, TS 3.4.1 - Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits (TAC Nos. MA5989, MA5990, MA5994, MA5995)," November 24, 1999.