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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTENTION: Document Control Desk

SUBJECT: Duke Energy Corporation
McGuire Nuclear Station Units 1 and 2
Docket Numbers 50-369 and 50-370
Technical Specification Amendments 191/172
Bases Pages Revision

Technical Specification Amendments 191 and 172 were transmitted to McGuire by NRC letter dated March 20, 2000. The Technical Specification Bases pages that were included with these amendments had Revision 1 on the bottom of each page. This revision number was incorrect. Attached are the same Technical Specification Bases pages with only the revision number changed. The correct revision number is Revision 8. Please replace affected pages with these new pages.

Please contact P. T. Vu at (704) 875-4302 if you have any questions.

Very truly yours,

H B Barron

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HBB/PTV/s

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A001

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Master File 1.3.2.9
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ATTACHMENT TO DUKE ENERGY LETTER DATED MAY 17, 2000

TECHNICAL SPECIFICATION AMENDMENTS 191/172

BASES PAGES REVISION

Replace the following Technical Specification Bases pages with the attached revised pages. The revised pages are identified by Revision 8 at the bottom of the pages.

Remove	Insert
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

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:

$$F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{LCO}$$

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

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 surveillances.

REFERENCES

1. UFSAR Section 15.4.8
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2005P "Duke Power Company Thermal Hydraulic Statistical Core Design Methodology", September 1992.
6. 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 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.

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.