ATTACHMENT 3

MARKED-UP TECHNICAL SPECIFICATIONS BASES
(FOR INFORMATION ONLY)

The following pages depict the changes proposed to the existing Technical Specification Bases. These pages are provided for information only with the final changes processed in accordance with the provisions of TS 5.5.14, Technical Specification (TS) Bases Control Program.

67 pages follow this cover sheet

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which requires that the minimum departure from nucleate boiling ratio (DNBR) of the limiting rod during Condition I and II events is greater than or equal to the DNBR correlation limits.

To meet this correlation limit design basis while accounting for uncertainties for Westinghouse fuel, for Revised Thermal Design Procedure (RTDP) analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters,computer codes, and DNB correlation (WRB-2) predictions are combined statistically to obtain the overall DNBR uncertainty factor. This DNBR uncertainty factor is used to define the design limit DNBR, which corresponds to a 95% probability with 95% confidence that DNB will not occur on the limiting fuel rods during Condition I and II events. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their normal values. The design limit DNBR values are 1.21 and 1.22 for thimble and typical cells, respectively, for VANTAGE 5 fuel. In addition, margin has been maintained by meeting safety analysis DNBR limits above the design limit DNBR to offset known DNBR penalties and to provide DNBR margin for operating and design flexibility. Reference 3 discusses non-RTDP transients. These transients are analyzed using the WRB-2, W-3, ABB-NV, or WLOP DNB correlation, as applicable for the specific transient. The correlation limits for WRB-2, W-3, ABB-NV, and WLOP are 1.17, 1.30, 1.13, and 1.18, respectively.

For Framatome GAIA fuel, uncertainties are statistically applied to the boundary conditions analyzed rather than to the DNBR design limit. The DNBR safety limit for GAIA fuel assemblies is 1.12 for the ORFEO-GAIA Critical Heat Flux (CHF) correlation and 1.15 for the ORFEO-NMGRID CHF correlation with COBRA-FLX using the P-SCHEME Solver. The ORFEO-NMGRID CHF correlation DNBR safety limit is 1.18 in COBRA-FLX with the PV Solver. The DNBR safety limit is 1.12 for the ORFEO-GAIA CHF correlation and 1.15 for the ORFEO-NMGRID CHF correlation with XCOBRA-IIIC. Reference 5, 6, 7

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is

(continued)

BASES

BACKGROUND (CONTINUED)

prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. Reference 4 further discusses the fuel centerline temperature design basis.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Trip System (RTS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

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

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

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

The Reactor Trip System Allowable Values in Table 3.3.1-1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (CONTINUED)

Protection for these reactor core SLs is provided by the proper operation of the steam generator safety valves and the following automatic reactor trip functions:

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Low reactor coolant system flow;

d. Overtemperature ΔT trip;

e. Overpower ΔT trip; and

f. Power Range Neutron Flux trip.

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

SAFETY LIMITS

The reactor core safety limits figure provided in the COLR shows the loci of points of THERMAL POWER, pressurizer pressure, and average temperature below which the calculated DNBR is not less than the design limit DNBR values, the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation.

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

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

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RTS functions that the above criteria are satisfied during steady state operation, normal operating transients, and anticipated operational occurrences (AOOs). To ensure

(continued)

BASES

SAFETY LIMITS (CONTINUED)

that the RTS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation.

Appropriate functioning of the RTS ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔD I that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

Reference 4 discusses the fuel temperature design basis. Figure 15.0-1 of Reference 2 depicts the protection provided by the Overpower ΔD T reactor trip function against fuel centerline melting.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Allowable Values for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is notgenerating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Chapter 15.
3. FSAR Section 4.4.1.1.
4. FSAR ~~Section 4.4.1.2.~~
5. FS1-0026874, Revision 2.0, "CHF Correlation Applicability Limits"
6. FS1-0050690, Revision 3.0, "ORFEO-GAIA and ORFEO-NMGRID Design Limit Validation for XCOBRA-IIIC"
7. FS1-0058371, Revision 1.0, "Validation of the ORFEO-NMGRID CHF Correlation with the PV Solver"

FSAR Sections 4.4.1.2 and 4.7.4.1.2

Reviewer Note: FSAR Section 4.7.4 is a new FSAR Section to capture Framatome GAIA fuel-specific content.

CHAPTER TABLE OF CONTENTS

CHAPTER B 3.2

POWER DISTRIBUTION LIMITS

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1A Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology) Westinghouse COLR Methods

BASES

BACKGROUND

The purpose of the limits on the values of $F_{Q(Z)}$ is to limit the local (i.e., pellet) peak power density. The value of $F_{Q(Z)}$ varies along the axial height (Z) of the core.

$F_{Q(Z)}$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_{Q(Z)}$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3A, "Axial Flux Difference (AFD)," and LCO 3.2.4, "Quadrant Tilt Power Ratio (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_{Q(Z)}$ varies with fuel loading patterns, bank insertion, fuel burnup, and changes in axial power distribution.

$F_{Q(Z)}$ is not directly measurable but is inferred from a power distribution measurement obtained with either the movable incore detector system or from an OPERABLE power distribution monitor~~i~~ring system (PDMS) (Reference 5). The results of the three-dimensional power distribution measurement are analyzed to derive a measured value for $F_{Q(Z)}$. These measurements are generally taken with the core at or near equilibrium conditions.

However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_{Q(Z)}$ that are present during nonequilibrium situations, such as load following. To account for these possible variations, the steady state value of $F_{Q(Z)}$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate

(continued)

BASES

BACKGROUND (continued)

LCOs, including the limits on AFD, QPTR, and control and shutdown bank insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot in irradiated fuel must not exceed 200 cal/gm (Ref. 2); and

d. The control and shutdown rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth rod stuck fully withdrawn (Ref. 3).

Limits on $F_{Q(Z)}$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the LOCA peak cladding temperature is typically most limiting.

$F_{Q(Z)}$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_{Q(Z)}$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_{Q(Z)}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_{Q(Z)}$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} K(Z) \quad \text{for } P>0.5$$

(continued)

BASES

LCO (continued)

$$F_Q(Z) \leq \frac{CFQ}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

where: CFQ = F_Q^{RTP} is the $F_{Q(Z)}$ limit at RTP provided in the COLR,
K(Z) is the normalized $F_{Q(Z)}$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The actual values of CFQ and K(Z) are given in the COLR.

For Relaxed Axial Offset Control operation, $F_{Q(Z)}$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_{Q(Z)}$.

An $F_Q^C(Z)$ evaluation requires obtaining a power distribution measurement in MODE 1. From the power distribution measurement results the measured value ($F_Q^M(Z)$) of $F_{Q(Z)}$. is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z) \; U_{FQ}$$

where U_{FQ} is a factor that accounts for fuel manufacturing tolerances and measurement uncertainty.

$F_Q^C(Z)$ is an excellent approximation for $F_{Q(Z)}$ when the reactor is at the steady state power at which the power distribution measurement was taken.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^M(Z) \; W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

The $F_{Q(Z)}$ limits define limiting values for core power peaking that

(continued)

precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

LCO 3.2.1A works in conjunction with LCO 3.2.2A and LCO 3.2.3A to define the power distribution limits when Westinghouse analysis methods govern COLR development.

(continued)

BASES

LCO (continued)

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_{Q(Z)}$ limits. If $F_{Q(Z)}$ cannot be maintained within the LCO limits, reduction of the core power is required. If the power distribution measurements are performed at a power level less than 100% RTP, then the $F_Q^C(Z)$ and $F_Q^W(Z)$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F_Q^{RTP} assures compliance with the LCO at all power levels.

Violating the LCO limits for $F_{Q(Z)}$ may produce unacceptable consequences if a design basis event occurs while $F_{Q(Z)}$ is outside its specified limits.

APPLICABILITY

The $F_{Q(Z)}$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

The Applicability is modified by a Note that states that this LCO is not applicable when Framatome COLR methods govern COLR development. When the Framatome analysis methods are used to develop the COLR limits, LCO 3.2.1B is the governing TS. The LCO 3.2.1A limits on $F_Q(Z)$ are the governing requirements when Westinghouse analysis methods govern the development of the COLR limits.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor which accounts for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_{Q(Z)}$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

The maximum allowable power level initially determined by Required

(continued)

BASES

Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

(continued)

BASES

ACTIONS

A.1 (continued)

Calculate the percent $F_Q^C(Z)$ exceeds its limit by the following expression:

$$\left[\left\{\max \text{ over } z \text{ of } \left(\frac{F_Q^C(Z)}{\frac{CFQ}{P} \times K(Z)}\right)\right\} - 1\right] \times 100 \text{ for } P \geq 0.5$$

$$\left[\left\{\max \text{ over } z \text{ of } \left(\frac{F_Q^C(Z)}{\frac{CFQ}{0.5} \times K(Z)}\right)\right\} - 1\right] \times 100 \text{ for } P < 0.5$$

A.2

A reduction of the Power Range Neutron Flux - High trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

The maximum allowable Power Range Neutron Flux - High trip, setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux - High trip setpoints.

A.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient consideringthe small likelihood of a severe transient in this time period, and the

(continued)

BASES

ACTIONS

A.3 (continued)

preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum Overpower ΔT trip setpoints.

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1A.1 prior to increasing THERMAL POWER above the limit imposedby Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions. Inherent in this action is identification of the cause of the out of limit condition, and the correction of the cause, to the extent necessary to allowsafe operation at the higher power level. The allowable power level is determined by evaluating $F_Q^C(Z)$ for the higher power level. SR 3.2.1A.1 must be satisfied prior to increasing power above the higher allowable power level or restoration of any reduced Reactor Trip System Setpoints.

B.1

If it is found that the maximum calculated value of $F_{Q(Z)}$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing both the positive and negative AFD limits by $\geq$ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

(continued)

BASES

ACTIONS

B.1 (continued)

Calculate the percent $F_Q^W(Z)$ exceeds its limit by the following expression:

$$\left[\left\{\max \text{ over } z \text{ of } \left(\frac{F^C_Q(Z) \times W(Z)_Q}{\frac{CFQ}{P} \times K(Z)}\right)\right\} - 1\right] \times 100 \text{ for } P \geq 0.5$$

$$\left[\left\{\max \text{ over } z \text{ of } \left(\frac{F^C(Z) \times W(Z)_Q}{\frac{CFQ}{0.5} \times K(Z)}\right)\right\} - 1\right] \times 100 \text{ for } P < 0.5$$

C.1

If Required Actions A.1 through A.4 or B.1 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1A.1 and SR 3.2.1A.2 are modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving MODE 1). The Note allows for power ascensions if the surveillances are not current. 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 $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been

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BASES

SURVEILLANCE REQUIREMENTS (continued)

measured in a reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified within 24 hours from when equilibrium conditions at RTP (or any other power level for extended operation) are achieved. Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainty allowances associated with the measurement are valid. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for a duration allowed by the Surveillance Frequency Control Program without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verificationof these parameters 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_Q was last measured.

SR 3.2.1A.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_{Q(Z)}$ obtained from core power distribution measurement results and $F_Q^C(Z) = F_Q^M(Z)U_{FQ}$ (Ref. 4). The value of U_{FQ} is determined using the formulation provided in the COLR. $F_Q^C(Z)$ is then compared to its specified limits.

The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP $F_Q(Z)$ limit, provides assurance that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1A.1 (continued)

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required within 24 hours after achieving equilibrium conditions (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.2.1A.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_{Q(Z)}$ limits. Because power distribution measurements are taken either at or near equilibrium conditions, the variations in power distribution resulting from normal operational maneuvers are not typically 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, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_{Q(Z)}$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power anddirectly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations are normally not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and

b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1A.2 (continued)

measurement in these regions. However, it is permissi~~a~~ble to exclude a smaller region from the evaluation. This is desirable if, for example, the limiting elevation is in the upper or lower 15% of the core based on cycle-specific supporting analyses.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. When $F_Q^C(Z)$ is measured, an evaluation of the expression below is required to account for any increase to $F_Q(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right]$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the appropriate factor specified in the COLR, or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. (The 25% extension allowed by SR 3.0.2 applies to this frequency.) These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power at any other time, and verifying the inferred results for 100% RTP meet the 100% RTP $F_Q(Z)$ limit, provides assurance that the $F_Q(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, within 24 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

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BASES (Continued)

REFERENCES

1. 10 CFR 50.46, 1974.

2. FSAR, Section 15.4.8.

3. 10 CFR 50, Appendix A, GDC 26.

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

5. WCAP-12472-P-A, “BEACON Core Monitoring and Operations Support System,” August 1994.

6. WCAP-12472-P-A, Addendum 1-A

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1B Heat Flux Hot Channel Factor [$F_Q(Z)$] Framatome COLR Methods

BASES

BACKGROUND

The purpose of the limits on the values of $F_{Q(Z)}$ is to limit the local (i.e., pellet) peak power density. The value of $F_{Q(Z)}$ varies along the axial height (Z) of the core.

$F_{Q(Z)}$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_{Q(Z)}$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3B, "Axial Flux Difference (AFD)," and LCO 3.2.4, "Quadrant Tilt Power Ratio (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_{Q(Z)}$ varies with fuel loading patterns, bank insertion, fuel burnup, and changes in axial power distribution.

$F_{Q(Z)}$ is not directly measurable but is inferred from a power distribution measurement obtained with the movable incore detector system or from an OPERABLE power distribution monitoring system. The results of the three-dimensional power distribution measurement are analyzed to derive a measured value for $F_{Q(Z)}$. Thesemeasurements are generally taken with the core at or near equilibrium conditions.

However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_{Q(Z)}$ that are present during nonequilibrium situations, such as load following. To account for these possible variations, the steady state value of $F_{Q(Z)}$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate

(continued)

BASES

BACKGROUND (continued)

LCOs, including the limits on AFD, QPTR, and control and shutdown bank insertion.

LCO 3.2.1B works in conjunction with LCO 3.2.2B and LCO 3.2.3B to define the power distribution limits when Framatome analysis methods govern COLR development.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the criteria specified in Chapter 15.4.8 of the FSAR must not be violated; and

d. The control and shutdown rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the LOCA peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} K(Z) \qquad \text{for } P>0.5$$

(continued)

BASES

LCO (continued)

$$F_Q(Z) \leq \frac{CFQ}{0.5} K(Z) \quad \text{for } P \leq 0.5$$

where: CFQ = F_Q^{RTP} is the $F_Q(Z)$ limit at RTP provided in the COLR,
K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The actual values of CFQ and K(Z) are given in the COLR.

$F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^V(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^V(Z)$ must meet the preceding limits on $F_Q(Z)$.

An $F_Q^C(Z)$ evaluation requires obtaining a power distribution measurement in MODE 1. From the power distribution measurement results the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z) \; U_{FQ}$$

where U_{FQ} is a factor that accounts for fuel manufacturing tolerances and measurement uncertainty.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the power distribution measurement was taken.

The expression for $F_Q^V(Z)$ is:

$$F_Q^V(Z) = F_Q^C(Z) \; V(Z)$$

where V(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. V (Z) is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

(continued)

BASES

LCO (continued)

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required. If the power distribution measurements are performed at a power level less than 100% RTP, then the $F_Q^C(Z)$ and $F_Q^V(Z)$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F_Q^{RTP} assures compliance with the LCO at all power levels.

Violating the LCO limits for $F_Q(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

LCO 3.2.1B works in conjunction with LCO 3.2.2B and LCO 3.2.3B to define the power distribution limits when Framatome analysis methods govern COLR development.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

The Applicability is modified by a Note that states that this LCO is not applicable when Westinghouse COLR methods govern COLR development. When the Westinghouse analysis methods are used to develop the COLR limits, LCO 3.2.1A is the governing TS. The LCO 3.2.1B limits on $F_Q(Z)$ are the governing requirements when Framatome analysis methods govern the development of the COLR limits.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor which accounts for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$.

(continued)

BASES

The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

(continued)

BASES

ACTIONS

A.1 (continued)

Calculate the percent $F_Q^C(Z)$ exceeds its limit by the following expression:

$$\left[\frac{\left\{\text{max over z of}\left(\frac{F^C_Q(Z)}{\frac{CFQ}{P}\times K(Z)}\right)\right\}}{} - 1\right] \times 100 \text{ for } P \geq 0.5$$

$$\left[\frac{\left\{\text{max over z of}\left(\frac{F^C_Q(Z)}{\frac{CFQ}{0.5}\times K(Z)}\right)\right\}}{} - 1\right] \times 100 \text{ for } P < 0.5$$

A reduction of the Power Range Neutron Flux – High trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordancewith Required Action A.1.

The maximum allowable Power Range Neutron Flux - High trip, setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux - High trip setpoints.

A.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance withRequired Action A.1.

(continued)

BASES

ACTIONS

A.3 (continued)

The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum Overpower ΔT trip setpoints.

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1B.1 and SR 3.2.1B.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

Inherent in this action is identification of the cause of the out of limit condition, and the correction of the cause, to the extent necessary to allow safe operation at the higher power level. The allowable power level is determined by evaluating $F_Q^C(Z)$ and $F_Q^V(Z)$ for the higher power level. Both SR 3.2.1B.1 and SR 3.2.1B.2 must be satisfied prior to increasing power above the higher allowable power level or restoration of any reduced Reactor Trip System Setpoints.

B.1.1

If it is found that the maximum calculated value of $F_{Q(Z)}$ that can occur during normal maneuvers, $F_Q^V(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing both the positive and negative AFD limits by the %AFD/%F_Q specified in the COLR for each 1% the current $F_Q^V(Z)$ exceeds the limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

(continued)

BASES

ACTIONS

B.1.1 (continued)

Calculate the percent $F_Q^V(z)$ exceeds its limit by the following expression:

$$\left[\frac{\left\{\text{max over z of}\left(\frac{F_Q^C(Z) \times V(Z)_Q}{CFQ}\right)\right\}}{\left(\frac{CFQ}{P} \times K(Z)\right)} - 1\right] \times 100 \text{ for } P \geq 0.5$$

$$\left[\frac{\left\{\text{max over z of}\left(\frac{F_Q^C(Z) \times V(Z)_Q}{CFQ}\right)\right\}}{\left(\frac{CFQ}{0.5} \times K(Z)\right)} - 1\right] \times 100 \text{ for } P < 0.5$$

B.1.2

Verification that $F_Q^V(Z)$ has been restored to within its limit, by performing SR 3.2.1B.1 and SR 3.2.1B.2 prior to increasing THERMAL POWER above the limit imposed by Required Action B.1.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

Inherent in this action is identification of the cause of the out of limit condition, and the correction of the cause, to the extent necessary to allow safe operation at the higher power level. The allowable power level is determined by evaluating $F_Q^C(Z)$ and $F_Q^V(Z)$ for the higher power level. Both SR 3.2.1B.1 and SR 3.2.1B.2 must be satisfied prior to increasing power above the higher allowable power level or restoration of any reduced Reactor Trip System Setpoints.

B.2.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^V(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^V(Z)$ is $F_Q^C(Z)$ multiplied by a cycle dependent function that accounts for power distribution transients encountered during normal operation. V(Z) is included in the COLR.

(continued)

BASES

B.2.1 (continued)

The Completion Time of 4 hours provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

The maximum allowable power level initially determined by Required Action B.1.1 may be affected by subsequent determinations of $F_Q^V(Z)$ and would require power reductions within 4 hours of the $F_Q^V(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^V(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

B.2.2

A reduction of the Power Range Neutron Flux – High trip setpoints by ≥ 1% for each 1% by which $F_Q^V(V)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordancewith Required Action A.1

The maximum allowable Power Range Neutron Flux - High trip, setpoints initially determined by Required Action B.2.2 may be affected by subsequent determinations of $F_Q^V(Z)$ and would require Power Range Neutron Flux - High trip setpoint reductions within 72 hours of the $F_Q^V(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux - High trip setpoints. Decreases in $F_Q^V(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux - High trip setpoints.

B.2.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^V(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance withRequired Action B.2.1.

(continued)

BASES

B.2.4

Verification that $F_Q^V(Z)$ has been restored to within its limit, by performing SR 3.2.1B.1 and SR 3.2.1B.2 prior to increasing THERMAL POWER above the limit imposed by Required Action B.2.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

Inherent in this action is identification of the cause of the out of limit condition, and the correction of the cause, to the extent necessary to allow safe operation at the higher power level. The allowable power level is determined by evaluating $F_Q^C(Z)$ and $F_Q^V(Z)$ for the higher power level. Both SR 3.2.1B.1 and SR 3.2.1B.2 must be satisfied prior to increasing power above the higher allowable power level or restoration of any reduced Reactor Trip System Setpoints.

C.1

If Required Actions A.1 through A.4 or B.1.1 and B.1.2 or B.2.1 through B.2.4 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1B.1 and SR 3.2.1B.2 are modified by a Note. The Note applies during the first power ascension at the beginning of each operating cycle, i.e., following a refueling outage. The Note allows for power ascension if the surveillances are not current. 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 $F_Q^C(Z)$ and $F_Q^V(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^V(Z)$ could not have previously been

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

measured in a reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^V(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^V(Z)$ following a power increase of more than 10%, ensures that they are verified within 24 hours from when equilibrium conditions at RTP (or any other power level for extended operation) are achieved. Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainty allowances associated with the measurement are valid. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for a duration allowed by the Surveillance Frequency Control Program without verification of $F_Q^C(Z)$ and $F_Q^V(Z)$. The Frequency condition is not intended to require verification of these parameters 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 $F_Q(Z)$ was last measured.

SR 3.2.1B.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from core power distribution measurement results and $F_Q^C(Z) = F_Q^M(Z)U_{FQ}$ (Ref. 4). The value of U_{FQ} is determined using the formulation provided in the COLR. $F_Q^C(Z)$ is then compared to its specified limits.

The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP $F_Q(Z)$ limit, provides assurance that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1B.1 (continued)

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required within 24 hours after achieving equilibrium conditions (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.2.1B.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because power distribution measurements are taken either at or near equilibrium conditions, the variations in power distribution resulting from normal operational maneuvers are not typically 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, calculated as a function of core elevation, Z, is called V(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by V(Z) covers the maximum peaking variation in power distribution that can occur during the interval between flux maps.

The limit with which $F_Q^V(Z)$ is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The V(Z) data, which can be burnup window dependent, is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^V(Z)$ evaluations are normally not applicable for the following axial core regions,measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and

b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1B.2 (continued)

measurement in these regions. However, it is permissible to exclude a smaller region (defined in the COLR) from the evaluation. This is desirable if, for example, the limiting elevation is in the upper or lower 15% of the core based on cycle-specific supporting analyses.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power at any other time, and verifying the inferred results for 100% RTP meet the 100% RTP $F_{Q(Z)}$ limit, provides assurance that the $F_{Q(Z)}$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_{Q(Z)}$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, within 24 hours after achieving equilibrium conditions to ensure that $F_{Q(Z)}$ is within its limit at higher power levels.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

REFERENCES

1. 10 CFR 50.46, 1974.

2. FSAR, Section 15.4.8.

3. 10 CFR 50, Appendix A, GDC 26.

4. ANP-10297P-A, Revision 0, The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results, February 2013.

5. ANP-10297P-A, Revision 0, Supplement 1PA, Revision 1, The ARCADIA Reactor Analysis System for PWRs Methodology Description and Benchmarking Results, December 2020.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2A Nuclear Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$ Westinghouse COLR Methods

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 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.

$F^N_{\Delta H}$ 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, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. $F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution measurement obtained with either the movable incore detector system or from an OPERABLE power distribution monitoring system (PDMS) (Reference 4). Specifically, the results of the three dimensional power distribution measurement are analyzed to determine $F^N_{\Delta H}$. This factor is calculated at a frequency controlled by the Surveillance Frequency Control Program. However, during power operation, the global power distributionis monitored by LCO 3.2.3A, "Axial Flux Difference (AFD)," and LCO 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," which address directly and continuously measured process variables. Compliance with these LCOs, along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

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. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

LCO 3.2.2A works in conjunction with LCO 3.2.1A and LCO 3.2.3A to define the power distribution limits when Westinghouse analysis methods govern COLR development.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

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

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 average fuel pellet enthalpy at the hot spot in irradiated fuel must not exceed 200 cal/gm (Ref. 1); and

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

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency.The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion applicable to a specific DNBR correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB condition.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ in the analyses. Likewise,

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all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_{Q(Z)}$) 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 3.2.3A, “Axial Flux Difference (AFD),” LCO 3.2.4, “Quadrant Power Tilt Ratio (QPTR),” LCO 3.1.4 “Rod Group Alignment Limits,” LCO 3.1.5 “Shutdown Bank Insertion Limits,” LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_{Q(Z)}$)."

$F^N_{\Delta H}$ and $F_{Q(Z)}$ 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 I events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removalcapability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$ described by the equation contained in the COLR, is the design radial peaking factor used in the unit safetyanalyses.

A power multiplication factor in this equation includes an additional allowance for higher radial peaking from reduced thermal feedback and greater control bank insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by a cycle-dependent factor, $PF_{\Delta H}$, specified in the COLR for each 1% RTP reduction in THERMAL POWER.

(continued)

BASES

LCO (continued)

If the power distribution measurements are performed at a power level less than 100% RTP, then the $F^N_{\Delta H}$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with $F^{RTP}_{\Delta H}$ assures compliance with the LCO at all power levels.

APPLICABILITY

The $F^N_{\Delta H}$ 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.

The Applicability is modified by a Note that states that this LCO is not applicable when Framatome COLR methods govern COLR development. When the Framatome analysis methods are used to develop the COLR limits, LCO 3.2.2B is the governing TS. The LCO 3.2.2A limits on $F^N_{\Delta H}$ are the governing requirements when Westinghouse analysis methods govern the development of the COLR limits.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ 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 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition foran extended period of time. The restoration of the peaking factor to withinits limits by power reduction or control rod movement does not restore compliance with the LCO. Thus, even though actions are taken to satisfy Required Action A.1.1, Condition A cannot be exited until a valid surveillance demonstrates compliance with the LCO.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Even if Required Action A.1.1 is completed within the 4 hour time period, Required Action A.2 requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2A.1.

(continued)

BASES

Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP, however THERMAL POWER does not have to be reduced to comply with these

(continued)

BASES

ACTIONS

A.1.1 (continued)

requirements. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux - High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing power to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steadystate operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required powerlevel from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints; however, for extended operations at the reduced power level, the reduced trip setpoints are required to protect against transients involving positive reactivity excursions. This is a sensitive operation that may inadvertently actuate the Reactor Trip System.

A.2

Once actions have been taken to restore $F^N_{\Delta H}$ to within its limits per Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, a power distribution measurement (SR 3.2.2A.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The CompletionTime of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of

(continued)

BASES

ACTIONS

A.2 (continued)

having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain an incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the$F^N_{\Delta H}$ 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. SR 3.2.2A.1 must be satisfied prior to increasing power above the allowable power level or restoration of any reduced Reactor Trip System setpoints. When$F^N_{\Delta H}$ is measured at reduced power levels, the allowable power level is determined by evaluating $F^N_{\Delta H}$ for higher power levels.

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.1.1 through A.3 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 3.2.2A.1

SR 3.2.2A.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving Mode 1). The note allows for power ascensions if the surveillances are not current. 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. Equilibrium conditions are achieved when the core is sufficiently stable

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2A.1 (continued)

such that the uncertainty allowances associated with the measurement are valid.

The value of $F^N_{\Delta H}$ is determined by either using the movable incore detector system to obtain a flux distribution map or from the power distribution information provided by an OPERABLE PDMS. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distribution map. The measured value of $F^N_{\Delta H}$ must have the appropriate uncertainty included (4% for a flux distribution map and $U_{\Delta H}$ as defined in Reference 4 for a PDMS surveillance) before comparison to the limit. The value of $U_{\Delta H}$ is determined using the formulain the COLR.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle. Performing this Surveillance in Mode 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and verifying the inferred results for 100% RTP meet the 100% RTP $F^N_{\Delta H}$ limit, provides assurance that $F^N_{\Delta H}$ limit will be met when RTP is achieved, because peaking factors generally decrease as power level is increased.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 15.4.8.

2. 10 CFR 50, Appendix A, GDC 26.

3. 10 CFR 50.46.

4. WCAP-12472-P-A, “BEACON Core Monitoring and Operations Support System,” August 1994.

5. WCAP-12472-P-A, Addendum 1-A

B 3.2.2B Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$ – Framatome COLR Methods

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 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.

$F^N_{\Delta H}$ 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, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. $F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution measurement obtained with the movable incore detector system or from an OPERABLE power distribution monitoring system. Specifically, the results of the three dimensional power distribution measurement are analyzed to determine $F^N_{\Delta H}$. This factor is calculated at a frequency controlled by the Surveillance Frequency Control Program. However, during power operation, the global power distributionis monitored by LCO 3.2.3B, "Axial Flux Difference (AFD)," and LCO 3.2.4,"Quadrant Power Tilt Ratio (QPTR)," which address directly and continuously measured process variables. Compliance with these LCOs,along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuousbasis.

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. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

LCO 3.2.2B works in conjunction with LCO 3.2.1B and LCO 3.2.3B to define the power distribution limits when Framatome analysis methods govern COLR development.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

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

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 criteria specified in Chapter 15.4.8 of the FSAR must not be violated;and

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

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion applicable to a specific DNBR correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB condition.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin (continued)

with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LOCA safety analysis models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_{Q(Z)}$) 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 3.2.3B, “Axial Flux Difference (AFD),” LCO 3.2.4, “Quadrant Power Tilt Ratio (QPTR),” LCO 3.1.4 “Rod Group Alignment Limits,” LCO 3.1.5 “Shutdown Bank Insertion Limits,” LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_{Q(Z)}$)."

$F^N_{\Delta H}$ and $F_{Q(Z)}$ 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 I events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional allowance for higher radial peaking from reduced thermal feedback and greater control bank insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase by a cycle-dependent factor, $PF_{\Delta H}$, specified in the COLR for each 1% RTP reduction in THERMAL POWER.

(continued)

BASES

LCO (continued)

If the power distribution measurements are performed at a power level less than 100% RTP, then the $F^N_{\Delta H}$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with $F^{RTP}_{\Delta H}$ assures compliance with the LCO at all power levels.

APPLICABILITY

The $F^N_{\Delta H}$ 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.

The Applicability is modified by a Note that states that this LCO is not applicable when Westinghouse COLR methods govern COLR development. When the Framatome analysis methods are used to develop the COLR limits, LCO 3.2.2B is the governing TS. The LCO 3.2.2A limits on $F^N_{\Delta H}$ are the governing requirements when Westinghouse analysis methods govern the development of the COLR limits.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ 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 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition foran extended period of time. The restoration of the peaking factor to within its limits by power reduction or control rod movement does not restore compliance with the LCO. Thus, even though actions are taken to satisfy Required Action A.1.1, Condition A cannot be exited until a valid surveillance demonstrates compliance with the LCO.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Even if Required Action A.1.1 is completed within the 4 hour time period, Required Action A.2 requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2B.1.

(continued)

BASES

ACTIONS

A.1.1 (continued)

Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be performed prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP, however THERMAL POWER does not have to be reduced to comply with these requirements. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux - High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing power to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steadystate operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required powerlevel from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints; however, for extended operations at the reduced power level, the reduced trip setpoints are required to protect against transients involving positive reactivity excursions. This is a sensitive operation that may inadvertently actuate the Reactor Trip System.

A.2

Once actions have been taken to restore $F^N_{\Delta H}$ to within its limits per Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, a power distribution measurement (SR 3.2.2B.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the4 hours allowed by either Action A.1.1 or Action A.1.2.1. The CompletionTime of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of (continued)

BASES

ACTIONS

A.2 (continued)

having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain an incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ 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. SR 3.2.2.1 must be satisfied prior to increasingpower above the allowable power level or restoration of any reduced Reactor Trip System setpoints. When $F^N_{\Delta H}$ is measured at reduced power levels, the allowable power level is determined by evaluating $F^N_{\Delta H}$for higher power levels.

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.1.1 through A.3 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 3.2.2B.1

This Surveillance has been modified by a Note that may require that more frequent surveillance be performed. When $F^N_{\Delta H}$ is measured, an evaluation of the trend of the $F^N_{\Delta H}$ is required to account for any increase to $F^N_{\Delta H}$ that may occur and cause the $F^N_{\Delta H}$ limit to be exceeded before the next required verification.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2B.1 (continued)

If the two most recent verifications show an increase in the $F^N_{\Delta H}$, it is required to meet the $F^V_Q(Z)$ limit with the last $F^V_Q(Z)$ increased by the appropriate factor specified in the COLR, or to evaluate $F^V_Q(Z)$ more frequently, each 7 EFPD. (The 25% extension allowed by SR 3.0.2 applies to this frequency.) These alternative requirements prevent $F^V_Q(Z)$ from exceeding its limit for any significant period of time without detection.

BASES

The value of $F^N_{\Delta H}$ is determined by either using the movable incore detector system to obtain a flux distribution map or from the power distribution monitoring system. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distribution map. The measured value of $F^N_{\Delta H}$ must have the appropriate uncertainty included (4% for a flux distribution map and $U_{\Delta H}$ as defined in the COLR) before comparison to the limit. The value of $U_{\Delta H}$ is determined using the formula in the COLR.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle. Performing this Surveillance in Mode 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and verifying the inferred results for 100% RTP meet the 100% RTP $F^N_{\Delta H}$ limit, provides assurance that $F^N_{\Delta H}$ limit will be met when RTP is achieved, because peaking factors generally decrease as power level isincreased.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 15.4.8.

2. 10 CFR 50, Appendix A, GDC 26.

3. 10 CFR 50.46.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3A AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology) Westinghouse COLR Methods

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

RAOC is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidates the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for two or more OPERABLE excore channels is outside its specified limits.

RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. Axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_{Q(Z)}$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition II, III, or IV events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the LOCA. The most important Condition III event is the loss of flow accident. The most important Condition II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition II accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks. Automatic rod control is available for insertion only.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as $\%\Delta$ flux or $\%\Delta I$.

The AFD limits are provided in the COLR. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

(continued)

BASES

LCO (continued)

Violating this LCO on the AFD could produce unacceptable consequences if a Condition II, III, or IV event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

The Applicability is modified by a Note that states that this LCO is not applicable when Framatome COLR methods govern COLR development. When the Framatome analysis methods are used to develop the COLR limits, LCO 3.2.3B is the governing TS. The LCO 3.2.3A limits on AFD are the governing requirements when Westinghouse analysis methods govern the development of the COLR limits.

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses.

A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3A.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

2. WCAP-10216-P-A, “Relaxation of Constant Axial Offset Control and FQ Surveillance Technical Specification,” February 1994.

3. FSAR, Chapter 7.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3B AXIAL FLUX DIFFERENCE (AFD) Framatome COLR Methods

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

PDC-A is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidates the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for two or more OPERABLE excore channels is outside its specified limits.

PDC-A calculations assume a wide range of xenon distributions and thenconfirm that the resulting power distributions satisfy the requirements of the accident analyses.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The PDC-A methodology generates a set of axial shapes through load follow maneuvers that are performed at various times in the cycle about a target AFD. The target AFD is the AFD at the time in cycle being analyzed. These axial power shapes are acceptable for LOCA and loss of flow accident, and for the initial condition of anticipated transients.

Since the PDC-A AFD barn is a floating barn about a target AFD, SR 3.2.3B.2 is performed. The AFD barn is established in the COLR where the barn is defined as the deviation about a target AFD. The V(Z) penalty is determined based on this floating barn. Therefore, when an $F_Q(Z)$ measurement is made, a target AFD must be established to ensure consistency between the operating barn and the V(Z) penalty function. (Ref. 1)

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition II, III, or IV events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the LOCA. The most important Condition III event is the loss of flow accident. The most important Condition II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition II accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks. Automatic rod control is available for insertion only.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 2). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR.

(continued)

BASES

LCO (continued)

Violating this LCO on the AFD could produce unacceptable consequences if a Condition II, III, or IV event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using PDC-A methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

The Applicability is modified by a Note that states that this LCO is not applicable when Westinghouse COLR methods govern COLR development. When the Framatome analysis methods are used to develop the COLR limits, LCO 3.2.3B is the governing TS. The LCO 3.2.3A limits on AFD are the governing requirements when Westinghouse analysis methods govern the development of the COLR limits.

ACTIONS

A.1

With AFD exceeding its limit, the unit is allowed 30 minutes to restore AFD within limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring AFD within its limit. When the AFD limit is exceeded, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 30 minutes provides an acceptable time to restore AFD to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3B.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.3B.2

This Surveillance determines the target AFD that corresponds with the measurement of $F_Q(Z)$ during performance of SR 3.2.1B.2. To best correlate with the conditions assumed in the analyses, it is desired that this SR be performed during steady state, equilibrium conditions with control rods at their normal positions at the highest power level as close to RTP as practical.

SR 3.2.3B.2 is modified by a Note. This Note states that the initial target AFD after each refueling may be determined from design predictions and, between measurement, the target AFD may be updated by adding the most recently measured AFD value and the change in the predicted AFD value since that measurement. The 'AFD limits' will then be updated by imposing the resulting new 'target AFD' value on the current 'target AFD band', which was expressed in deviation from the 'target AFD'.

Measurement of the target flux difference is accomplished by taking a flux map, or use of a power distribution measurement system, when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux profile determination provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

The Surveillance must be performed once within 31 EFPD after each refueling outage. Thereafter, the Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. CALLAWAY SER For [This] Amendment [xxx]

2. FSAR, Chapter 7.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3A, "AXIAL FLUX DIFFERENCE (AFD) Westinghouse COLR Methods," or LCO 3.2.3B, "AXIAL FLUX DIFFERNECE (AFD) Framatome COLR Methods," LCO 3.2.4, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains withinthe bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot in irradiated fuel must not exceed 200 cal/gm (Ref. 2) for Westinghouse fuel, or violate the criteria specified in Chapter 15.4.8 of the FSAR for Framatome fuel; and

d. The control and shutdown rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and bank insertion and rod group alignment are established to preclude core power distributions that exceed the safety analyses limits. (continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, above which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt, or reduce power, as necessary. Note that a power reduction may cause a change in the tilted condition.

The maximum allowable THERMAL POWER level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require a THERMAL POWER reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable THERMAL POWER level. Decreases in

(continued)

BASES

ACTIONS

A.1 (continued)

QPTR would allow raising the maximum allowable THERMAL POWER level and increasing THERMAL POWER up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR may still exceed its limits. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F_{\Delta H}^{N}$ and $F_Q(Z)$, as approximated by $F_C(Z)$ and $F^W(Z)$ (for Westinghouse methods of analysis) and $F^V(Z)$ (for Framatome methods of analysis) are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^{N}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators ofpower distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support obtaining a power distribution measurement. Power distribution information can be obtained either by using the movable incore detectors or from an OPERABLE power distribution monitoring system ~~(PDMS) (Reference 5)~~. A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per RequiredAction A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform power distribution measurement. If these peaking factors are not within their limits, the Required Actions associated with these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^{N}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

This Required Action is modified by a Note that directs the performer to the applicable set of SRs associated with the governing COLR development method.

(continued)

BASES

ACTIONS (continued)

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptionsin the safety analyses.

A.5

If the QPTR remains above the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limit prior to increasing THERMAL POWER to above the limit of Required Action A.1. This is done to detect any subsequent significant changes in QPTR. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00.

Required Action A.5 is modified by two Notes. Note 1 states that excore detectors are not normalized to restore QPTR to within limit until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limit, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing a power distribution measurement to verify peaking factors per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

(continued)

BASES

ACTIONS (continued)

A.6

Once the excore detectors are normalized to restore QPTR to within limit (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$ ~~,~~(for Westinghouse methods) or $F_Q^V(Z)$ (for Framatome methods), and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. Equilibrium conditions at RTP areachieved when the core is sufficiently stable at the intended operating conditions to support obtaining a power distribution measurement. As an added precaution, if the core does not reach equilibrium conditions at RTP within 24 hours, but power is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by ~~a~~two Notes. Note 1 ~~that~~ states that the peaking factor surveillances must be completed when the excore detectors have been normalized to restore QPTR to within limit (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limit. Note 2 directs the performer to perform the applicable set of SRs associated with the governing COLR development method.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

For those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the inputs from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors may be used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. If one of the symmetric thimbles is not available, then other pairs (triples) of symmetric thimbles can be monitored to gain information about the quadrant with the out-of-service thimble, provided the reference case is set up with the same thimble groupings (Ref. 4). Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore tilt result may be compared against previous tilt values either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent power distribution measurement data.

REFERENCES

1. 10 CFR 50.46.

2. FSAR, Section 15.4.8.

3. 10 CFR 50, Appendix A, GDC 26.

4. Westinghouse Recommendations on Monitoring QPTR with One Power Range Channel Out of Service, (Proprietary). Only applicable to the Westinghouse COLR methods.

5. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994. Only applicable to the Westinghouse COLR methods.

6. WCAP-12472-P-A, Addendum 1-A. Only applicable to the Westinghouse COLR methods.