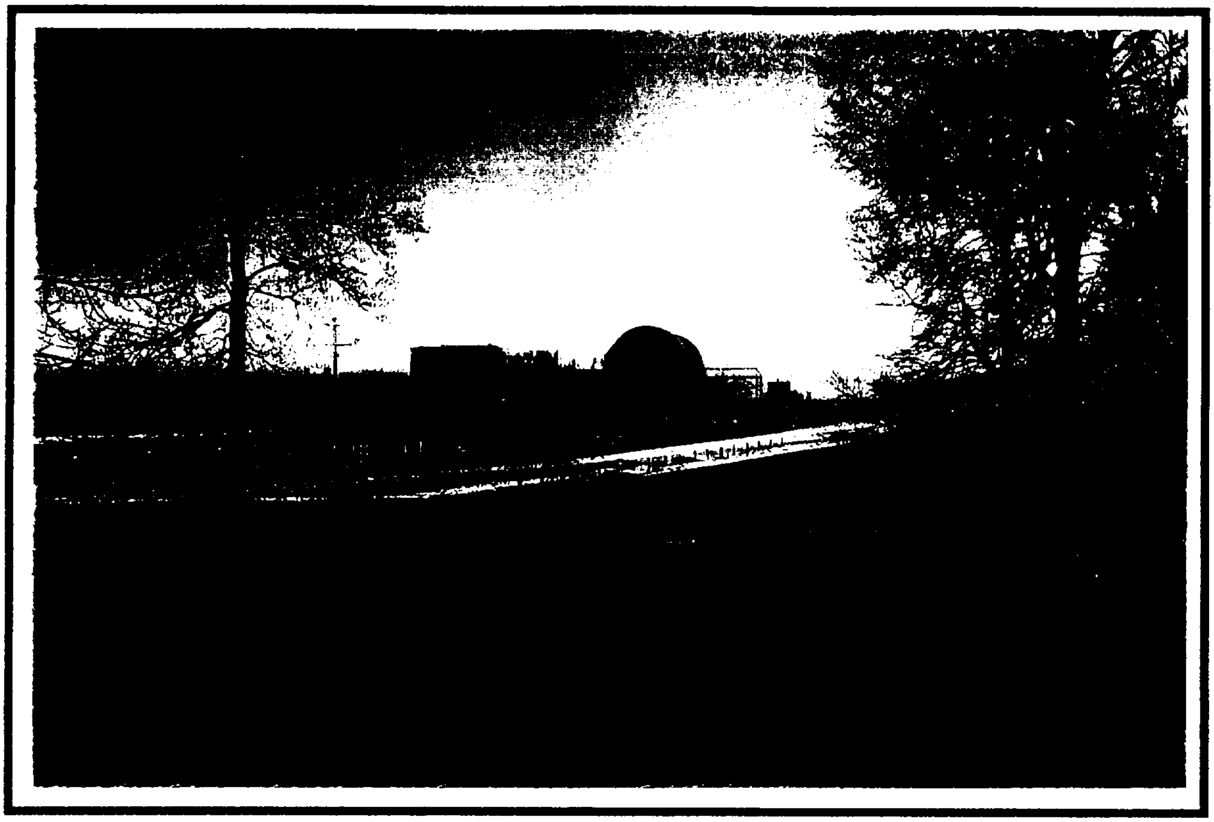
NORTH ANNA POWER STATION

Section 3.2
Power Distribution Limits



VOLUME 8

Improved Technical Specifications

Dominion

NORTH ANNA POWER STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.2 - POWER DISTRIBUTION LIMITS

SECTION 3.2 - POWER DISTRIBUTION LIMITS

IMPROVED TECHNICAL SPECIFICATIONS

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_Q^M(Z)$ not within limit.	A.1 Reduce AFD limits ≥ 1% for each 1% $F_Q^M(Z)$ exceeds limit.	15 minutes after each $F_Q^M(Z)$ determination
	<u>OR</u>	
	A.2.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^M(Z)$ exceeds limit.	15 minutes after each $F_Q^M(Z)$ determination
	<u>AND</u>	
	A.2.2 Reduce Power Range Neutron Flux–High trip setpoints ≥ 1% for each 1% $F_Q^M(Z)$ exceeds limit.	72 hours after each $F_Q^M(Z)$ determination
	<u>AND</u>	
	A.2.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^M(Z)$ exceeds limit.	72 hours after each $F_Q^M(Z)$ determination
	<u>AND</u>	
	A.2.4 Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.2.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

------------------- NOTE -------------------

During power escalation, THERMAL POWER may be increased until a power level for extended operation has been achieved, at which a power distribution map is obtained.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	------------------NOTE------------------ If $F_Q^M(Z)$ measurements indicate $$\text{maximum over } z \left[\frac{F_Q^M(Z)}{K(Z)}\right]$$ has increased since the previous evaluation of $F_Q^M(Z)$: a. Increase $F_Q^M(Z)$ by the appropriate factor and verify $F_Q^M(Z)$ is still within limits; or b. Repeat SR 3.2.1.1 once per 7 EFPD until two successive flux maps indicate $$\text{maximum over } z \left[\frac{F_Q^M(Z)}{K(Z)}\right]$$ has not increased. --	
	Verify $F_Q^M(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F_Q^M(Z)$ was last verified AND 31 EFPD thereafter

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3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE--------- Required Actions A.3 and A.4 must be completed whenever Condition A is entered. --------------------- $F^N_{\Delta H}$ not within limit.	A.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	<u>AND</u>		
	A.2	Reduce Power Range Neutron Flux–High trip setpoints to ≤ 55% RTP.	72 hours
	<u>AND</u>		
	A.3	Perform SR 3.2.2.1.	24 hours
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 --------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. --------------------- Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP <u>AND</u> Prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

-------------- NOTE --------------
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days

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3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	AND		
	A.2	Determine QTPR.	Once per 12 hours after achieving equilibrium Conditions from a THERMAL POWER reduction per Required Action A.1
	AND		
	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours AND Once per 7 days thereafter
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		
	A.5	--------NOTES--------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed. ---------------------	
		Normalize excore detectors to restore QPTR to within limits.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.6 --------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. ---------------------	
	Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after achieving equilibrium Conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------------------NOTES------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --	
	Verify QPTR is within limit by calculation.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.2	-------------------NOTE------------------- Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. --	
	Verify QPTR is within limit using the movable incore detectors.	12 hours

SECTION 3.2 - POWER DISTRIBUTION LIMITS

IMPROVED TECHNICAL SPECIFICATIONS BASES

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

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 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with 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, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$, $F_Q^M(Z)$. However, because this value represents a steady state condition, it does not encompass the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations, such as load changes.

To account for these possible variations, the steady state limit for $F_Q(Z)$ 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 LCOs, including the limits on AFD, QPTR, and control rod insertion.

BASES

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 energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control 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 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 Measured Heat Flux Hot Channel Factor, $F_Q^M(Z)$, shall be limited by the following relationships, as described in Reference 4:

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

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

(continued)

BASES

LCO (continued)

where: CFQ 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,

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

P is the fraction of RATED THERMAL POWER defined as

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

The actual values of CFQ, K(Z), and N(Z) are given in the COLR; however, CFQ is normally approximately 2, K(Z) is a function that looks like the one provided in Figure B 3.2.1-1, and N(Z) is a value greater than 1.0.

An $F_Q^M(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value of $F_Q(Z)$. Then, the measured $F_Q^M(Z)$ is increased by 1.03 which is a factor that accounts for fuel manufacturing tolerances and 1.05 which accounts for flux map measurement uncertainty (Ref. 5).

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.

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.

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

BASES

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.

ACTIONS

A.1

If $F_Q^M(Z)$ exceeds its specified limits, reducing the AFD limit by ≥ 1% for each 1% by which $F_Q^M(Z)$ exceeds its limit within the allowed Completion Time of 15 minutes, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded. The maximum AFD limits initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^M(Z)$ and would require AFD reductions with 15 minutes of the $F_Q^M(Z)$ determination, if necessary.

A.2.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^M(Z)$ exceeds its limit, maintains an acceptable absolute power density. The percent that $F_Q^M(Z)$ exceeds the limit can be determined from:

$$\left\{\text{maximum over } z \left(\frac{F_Q^M(Z)}{\frac{CFQ\ K(Z)}{P\ \ N(Z)}}\right) - 1.0\right\} \times 100 \quad \text{for } P > 0.5$$

$$\left\{\text{maximum over } z \left(\frac{F_Q^M(Z)}{\frac{CFQ\ K(Z)}{0.5\ N(Z)}}\right) - 1.0\right\} \times 100 \quad \text{for } P \le 0.5$$

$F_Q^M(Z)$ is the measured $F_Q(Z)$ multiplied by factors accounting 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 unit to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.2.1 may be affected by subsequent determinations of $F_Q^M(Z)$ and would require power reductions

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BASES

ACTIONS

A.2.1 (continued)

within 15 minutes of the $F_Q^M(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^M(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2.2

A reduction of the Power Range Neutron Flux–High trip setpoints by ≥ 1% for each 1% by which $F_Q^M(Z)$ 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.2.1. The maximum allowable Power Range Neutron Flux–High trip setpoints initially determined by Required Action A.2.2 may be affected by subsequent determinations of $F_Q^M(Z)$ and would require Power Range Neutron Flux–High trip setpoint reductions within 72 hours of the $F_Q^M(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux–High trip setpoints. Decreases in $F_Q^M(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux–High trip setpoints.

A.2.3

Reduction in the Overpower ΔT trip setpoints (value of K_4) by ≥ 1% (in ΔT span) for each 1% by which $F_Q^M(Z)$ 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.2.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.2.3 may be affected by subsequent determinations of $F_Q^M(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^M(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^M(Z)$ would allow increasing the maximum Overpower ΔT trip setpoints.

BASES

ACTIONS (continued)

A.2.4

Verification that $F^M_Q(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.2.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If Required Actions A.1, A.2.1, A.2.2, A.2.3, or A.2.4 are not met within their associated Completion Times, the unit must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the unit 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 unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 is modified by a Note. It states that THERMAL POWER may be increased until a power level for extended operation has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F^M_Q(Z)$ is within its specified limit after a power rise of more than 10% RTP over the THERMAL POWER at which it was last verified to be within specified limits. In the absence of this Frequency condition, it is possible to increase power to RTP and operate for 31 days without verification of $F^M_Q(Z)$. 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 was last measured.

SR 3.2.1.1

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

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 N(Z).

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

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^M(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^M(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^M(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the unit is operated in accordance with the Technical Specifications (TS).

Flux map data are taken for multiple core elevations. $F_Q^M(Z)$ evaluations are 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 measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. An evaluation of the expression below is required to account

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q^M(Z)$ limit to be exceeded before the next required $F_Q^M(Z)$ evaluation.

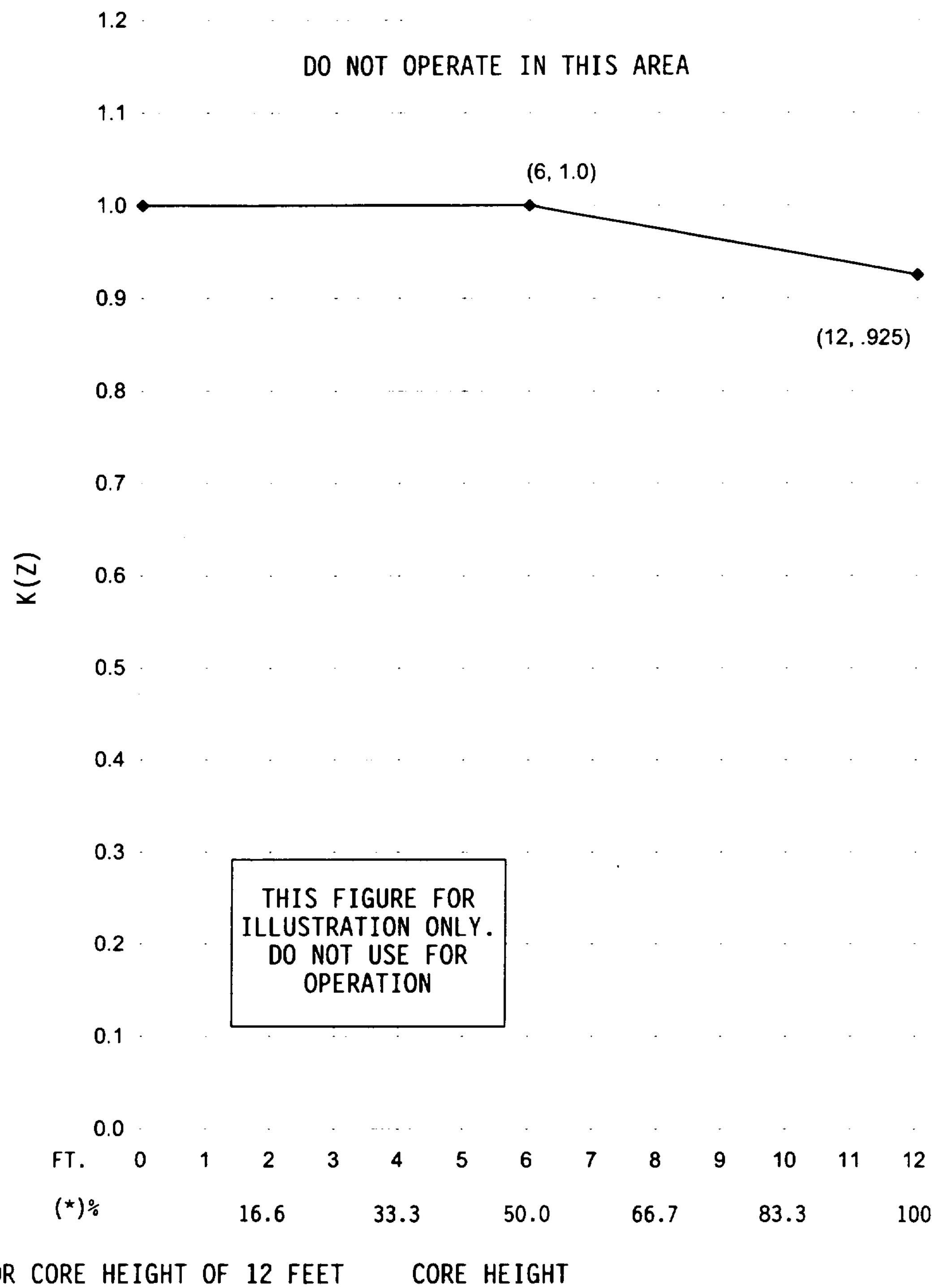
If the two most recent $F_Q^M(Z)$ evaluations show an increase in the expression

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

it is required to meet the $F_Q^M(Z)$ limit with the last $F_Q^M(Z)$ increased by the appropriate factor, or to evaluate $F_Q^M(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit without detection.

REFERENCES

1. 10 CFR 50.46, 1974.
2. VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient."
3. UFSAR, Section 3.1.22.
4. Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications, VEP-NE-1-A, March 1986.
5. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.



* FOR CORE HEIGHT OF 12 FEET

Figure B 3.2.1-1 (page 1 of 1)
K(Z)–Normalized $F_Q(Z)$ as a Function of Core Height

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

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. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

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

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to a value greater than the design limits. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

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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.

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 energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

For transients that may be DNB limited, the Reactor Coolant System flow, temperature, and pressure, and $F^N_{\Delta H}$ are the 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 a value which provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

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 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 ($F_Q(\ )$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," and LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits."

$F^N_{\Delta H}$ 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 1 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 identifies the coolant flow channel with the maximum enthalpy rise. This channel has 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 margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels.

BASES

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 design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have sufficient margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1 and A.2

Condition A is modified by a Note that requires that Required Actions A.3 and A.4 must be completed whenever Condition A is entered. Thus, because even if $F^N_{\Delta H}$ is restored to within limits, Required Action A.3 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power is reduced below 50% RTP, Required Action A.4 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. In addition, Required Action A.3 is performed if power ascension is delayed past 24 hours.

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 and reduce the Power Range Neutron Flux–High to ≤ 55% RTP in accordance with Required Action A.2. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state 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 provides an acceptable time to reach the required power level from full power operation without allowing the unit to remain in an unacceptable condition for an extended period of time.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.2 recognizes that, once power is reduced, the safety analysis assumptions are

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.3

Once the power level has been reduced to < 50% RTP per Required Action A.1, an incore flux map (SR 3.2.2.1) 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 Action A.1. The Completion Time 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 having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.4

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 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.

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

B.1

When Required Actions A.1 through A.4 cannot be completed within their required Completion Times, the unit must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the unit 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 unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The $F^N_{\Delta H}$ limit contains an allowance of 1.04 to account for measurement uncertainty.

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.

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

REFERENCES

1. VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient."

2. UFSAR, Section 3.1.22.

3. 10 CFR 50.46.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

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.

Relaxed Power Distribution Control (RPDC) 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 invalidate 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 immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

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 RPDC methodology (Ref. 1) 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 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident. The most important Condition 2 events are uncontrolled rod withdrawal, excessive heat removal, and boration or dilution accidents. Condition 2 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 or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

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 $\%\Delta$ flux or $\%\Delta I$.

(continued)

BASES

LCO (continued)

The AFD limits are provided in the COLR. Figure B 3.2.3-1 shows typical RPDC AFD limits. The AFD limits for RPDC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

The LCO is modified by a Note which states that AFD shall be considered outside its limit when two or more OPERABLE excore channels indicate AFD to be outside its limit.

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

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 unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

BASES

REFERENCES

1. K.L. Basehore et al., "Virginia Power Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," VEP-NE-1-A, March 1986.

2. UFSAR, Chapter 7.

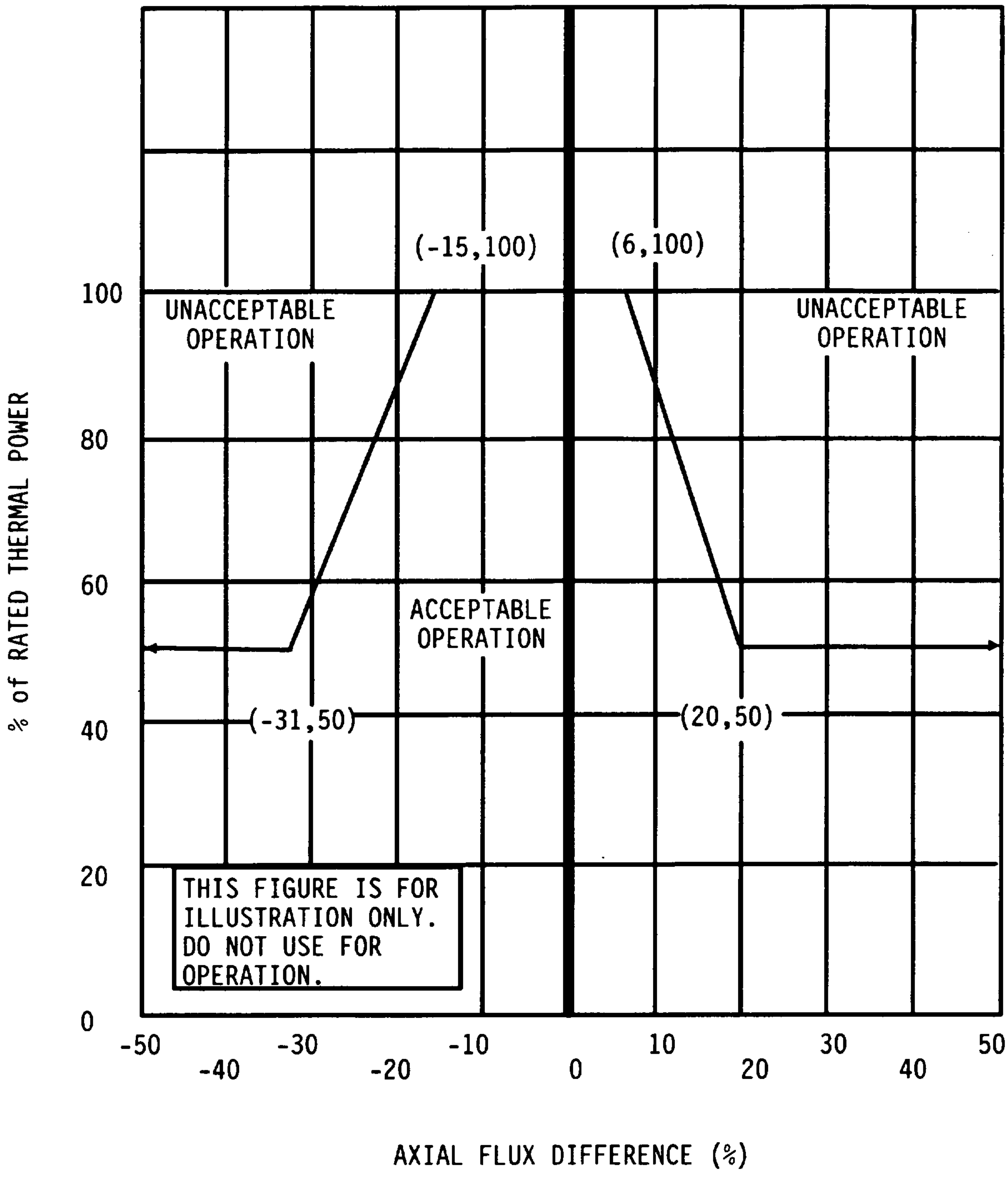

Figure B 3.2.3-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

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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 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod 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 within the 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 energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 2); and

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

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, at 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

(continued)

BASES

ACTIONS

A.1 (continued)

to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to the revised limit.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ 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 of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the unit and perform a flux map. If these peaking factors are not within their limits, the Required Actions of 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 $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution.

(continued)

BASES

ACTIONS

A.3 (continued)

Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

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 to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded 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 limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR is not restored to within limits 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, the Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents

(continued)

BASES

ACTIONS

A.5 (continued)

exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (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 $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing 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 Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (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 limits and the core returned to power.

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 unit systems.

BASES

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.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

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. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

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 result may be compared against previous flux maps 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 flux map data.

REFERENCES

1. 10 CFR 50.46.
2. VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient."
3. UFSAR, Section 3.1.22.

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SECTION 3.2 - POWER DISTRIBUTION LIMITS

IMPROVED STANDARD TECHNICAL SPECIFICATIONS

MARKUP AND JUSTIFICATION FOR DEVIATIONS

CTS

3.2.2

3.2 POWER DISTRIBUTION LIMITS

3.2.1B Heat Flux Hot Channel Factor ($F_Q(Z)$) ~~(F_Q Methodology)~~ (3)

LCO 3.2.1B $F_Q(Z)$, as approximated by $F_Q^C(Z)$ M ~~and $F_Q^W(Z)$~~, shall be within the limits specified in the COLR. (1)

APPLICABILITY: MODE 1.

ACTIONS

TSTF-241

after each $F_Q^M(Z)$ determination

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q^C(Z)$ M not within limit. Insert Required Action A.1 from page 3.2-5	A.1 (2.)	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ M exceeds limit.	15 minutes (1)
	AND		
	A.2 (2.)	Reduce Power Range Neutron Flux—High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ M exceeds limit.	~~8~~ 72 hours TSTF-95 (1)
	AND		
	A.3 (2.)	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ M exceeds limit.	72 hours (1)
	AND		
	A.4 (2.)	Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1 (2.) (1)

Action a

Action a

Action a

Action b

(continued)

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ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
4.2.2.2.f.2.a	~~B. $F^W_Q(Z)$ not within limits.~~	~~B.1~~ A.1	Reduce AFD limits ≥ 1% for each 1% $F^M_Q(Z)$ exceeds limit. M	~~2 hours~~ 15 minutes after each $F^M_Q(Z)$ determination.	TSTF-99 1
new	B ~~C~~. Required Action and associated Completion Time not met.	B ~~C~~.1	Be in MODE 2.	6 hours	1

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SURVEILLANCE REQUIREMENTS

Note to SR 4.2.2.2.d.1

------------------------------------NOTE------------------------------------
During power escalation ~~at the beginning of each cycle~~, THERMAL POWER may be increased until ~~an equilibrium~~ power level for extended operation has been achieved, at which a power distribution map is obtained. (1)

	SURVEILLANCE	FREQUENCY
New	SR 3.2.1.1 Verify ~~$F_Q(Z)$~~ $F_Q^M(Z)$ is within limit. Insert SR 3.2.1.2 Note	Once after each refueling prior to THERMAL POWER exceeding 75% RTP
4.2.2.2.d.1		AND Once within ~~[12]~~ hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which ~~$F_Q(Z)$~~ $F_Q^M(Z)$ was last verified (2) (1)
4.2.2.2.d.2		AND 31 EFPD thereafter

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
4.2.2.2.e	SR ~~3.2.1.2~~	------------------NOTE------------------ If $F_Q^M(Z)$ ~~is within limits and~~ measurements indicate $$\text{maximum over } z \left[\frac{F_Q^M(Z)}{K(Z)} \right]$$ has increased since the previous evaluation of $F_Q^M(Z)$: a. Increase $F_Q^M(Z)$ by ~~a~~ the appropriate factor ~~of [1.02]~~ and ~~to~~ verify $F_Q^M(Z)$ is still within limits; or b. Repeat SR 3.2.1.~~2~~1 once per 7 EFPD until two successive flux maps indicate $$\text{maximum over } z \left[\frac{F_Q^M(Z)}{K(Z)} \right]$$ has not increased. --	TSTF-97
4.2.2.2.e.1	Move to SR 3.2.1.1		TSTF-98 (2) (1)
4.2.2.2.e new	~~Verify $F_Q^W(Z)$ is within limit.~~	~~Once after each refueling prior to THERMAL POWER exceeding 75% RTP~~ ~~AND~~ (continued)	(1)

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once within [12] hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^W_Q(Z)$ was last verified AND 31 EFPD thereafter

①

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1. ITS 3.2.1B, FQ(z) (FQ Methodology) is revised to reflect the Relaxed Power Distribution Control (RPDC) methodology and associated FQ Surveillance Technical Specifications approved by the NRC in VEP-NE-1-A, March, 1986, and incorporated in the current Technical Specifications. The most significant differences are that the RPDC methodology does not utilize the $F_Q^W(z)$ and $F_Q^C(z)$ terms used in the NUREG. Only one F_Q value is measured, represented by $F_Q^M(Z)$, which is compared to the transient $F_Q(Z)$ limit (i.e., the FQ limit is decreased by the cycle specific multiplier function, called N(z), which accounts for power distribution transients during normal operation). Unlike the NUREG terminology, $F_Q^M(Z)$ includes measurement uncertainties. Because only one value is measured, only one Surveillance and one Condition are required, rather than two under the Westinghouse methodology. Some approved generic changes applicable to ITS 3.2.1B are not incorporated because they do not apply to the approved methodology. These are TSTF-98, 290, and 338.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. ITS 3.2.1A, $F_Q(Z)$ (F_{xy} Methodology), is deleted. ITS 3.2.1B, $F_Q(Z)$ (F_Q Methodology) is used as the model for the North Anna ITS.

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3.23

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Actions A.2 3 and A.8 4 must be completed whenever Condition A is entered. -------------------- $F^N_{\Delta H}$ not within limit.	~~A.1.1 Restore $F^N_{\Delta H}$ to within limit.~~	~~4 hours~~
	~~OR~~	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
	A.1.2.2 Reduce Power Range Neutron Flux—High trip setpoints to ≤ 55% RTP.	72 ~~8~~ hours
	AND	
	A.2 3 Perform SR 3.2.2.1.	24 hours
	AND	
		(continued)

Action a

Action a

Action b

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.~~3~~ 4 --------NOTE-------- THERMAL POWER does not have to be reduced to comply with this Required Action. --------------------	TSTF-240
Action C	Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP __AND__ Prior to THERMAL POWER exceeding 75% RTP __AND__ 24 hours after THERMAL POWER reaching ≥ 95% RTP
Action 6 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

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SURVEILLANCE REQUIREMENTS

4.2.3.1

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP **AND** 31 EFPD thereafter

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None

3F

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC) Methodology)~~

3F

3.2.1

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

4.2.1.2

-----NOTE-----
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

Action a

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

4.2.1.1

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify AFD within limits for each OPERABLE excore channel.	7 days ~~AND~~ ~~Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable~~

TSTF-110

Rev. 0

1. ITS 3.2.3A, AFD (CAOC Methodology), is deleted. ITS 3.2.3B, AFD (RAOC Methodology) is used as the model for the North Anna ITS.

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action a.1.b (U1) Action a.2.b (U2)	A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00. **AND**	2 hours after each QPTR determination	TSTF-241
new	Determine QPTR.	A.2 ~~Perform SR 3.2.4.1 and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.~~ **AND**	Once per 12 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1	TSTF-109 TSTF-241
new		A.3 Perform SR 3.2.1.1 and SR 3.2.2.1. **AND**	24 hours **AND** Once per 7 days thereafter	
new		A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition. **AND**	Prior to increasing THERMAL POWER above the limit of Required Action A.1	
			(continued)	

CTS

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
new	A. (continued)	A.5	--------NOTES-------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed. --------------------------		TSTF-241 TSTF-241
			~~Calibrate excore detectors to show zero QPTR.~~ Normalize excore detectors to restore QPTR to within limits	Prior to increasing THERMAL POWER above the limit of Required Action A.1	TSTF-241
		<u>AND</u>			
new		A.6	--------NOTE-------- Perform Required Action A.6 only after Required Action A.5 is completed. --------------------------		
			Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after ~~reaching~~ achieving equilibrium conditions at RTP not to exceed ~~OR~~ ~~Within~~ 48 hours after increasing THERMAL POWER above the limit of Required Action A.1	TSTF 241
new	B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours	

CTS

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY	
QPTR Definition 4.2.4.2	SR 3.2.4.1	-------------------NOTES------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR.		TSTF-241
new		2. SR 3.2.4.2 may be performed in lieu of this Surveillance ~~if adequate Power Range Neutron Flux channel inputs are not OPERABLE~~. ---		TSTF-109
4.2.4.1		Verify QPTR is within limit by calculation.	7 days ~~AND~~ ~~Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable~~	TSTF-110
new	SR 3.2.4.2	Not --------------------NOTE-------------------- until 12 hours after ~~Only~~ required to be performed ~~if~~ input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ > 75% RTP. ---		TSTF-109 TSTF-241
4.2.4.2		Verify QPTR is within limit using the movable incore detectors.	~~Once within~~ 12 hours ~~AND~~ ~~12 hours thereafter~~	TSTF-109 TSTF-109

None

SECTION 3.2 - POWER DISTRIBUTION LIMITS

IMPROVED STANDARD TECHNICAL SPECIFICATIONS BASES

MARKUP AND JUSTIFICATION FOR DEVIATIONS

 (6)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1B Heat Flux Hot Channel Factor ($F_Q(Z)$) ~~(F_Q Methodology)~~ (6)

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 3.2.3, "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 3.1.5, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis. (7) TSTF-136 (6)

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$, $F^M_Q(Z)$. However, because this value represents a steady state condition, it does not include encompass the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations, such as load following changes. (1) (1) (1)

To account for these possible variations, the steady state value of limit for $F_Q(Z)$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions. (1)

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

(continued)

Rev. 0

6

BASES

BACKGROUND (continued)

the appropriate LCOs, including the limits on AFD, QPTR, and control rod 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;

Unirradiated fuel is limited to 225cal/gm and irradiated fuel is limited to 200cal/gm

c. During an ejected rod accident, the energy deposition to ~~the fuel must not exceed 280 cal/gm~~ (Ref. 2); and 4

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control 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 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 ~~the NRC Policy Statement~~. 2

10 CFR 50.36(c)(2)(ii)

(continued)

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6

BASES (continued)

, as described in Reference 4

LCO

Measured

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

$$F_Q^M(Z) \le \cancel{\frac{CFQ}{P}K(Z)} \quad \frac{CFQ\ K(Z)}{P\ N(Z)} \quad \text{for } P > 0.5$$

1

P is the fraction of RATED THERMAL POWER defined as

$$F_Q^M(Z) \le \cancel{\frac{CFQ}{0.5}K(Z)} \quad \frac{CFQ\ K(Z)}{0.5\ N(Z)} \quad \text{for } P \le 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

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

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR. ~~and~~

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

~~For this facility,~~ the actual values of CFQ ~~and K(Z)~~ , K(Z), and N(Z) are given in the COLR; however, CFQ is normally ~~a number on the order of [2.32]~~ approximately 2 ~~and~~ K(Z) is a function that looks like the one provided in Figure B 3.2.1-1 and N(Z) is a value greater than 1.0 3 1 1 1

~~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 $F_Q(Z)$.~~ 1

The measured $F_Q^M(Z)$ is increased by

An $F_Q^M(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value (~~$F_Q^M(Z)$~~) of $F_Q(Z)$. Then, 1 1

~~$F_Q^C(Z) = F_Q^M(Z)$ [1.0815]~~ which 1

1.03

~~where [1.0815]~~ 1.05 which accounts for is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty (Ref. 5). 1

~~$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.~~ 1

(continued)

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6

BASES

LCO (continued)

~~The expression for $F^W_Q(Z)$ is:~~

~~$F^W_Q(Z) = F^C_Q(Z)\ W(Z)$~~

~~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.~~ 1

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.

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.

Violating the LCO limits for $F_Q(Z)$ produces 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.

ACTIONS

Insert from page B 3.2-15

A.~~1~~ 2

M the measured $F_Q(Z)$

M Reducing THERMAL POWER by $\geq$ 1% RTP for each 1% by which $F^M_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q(Z)$ is $F^M_Q(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F^M_Q(Z)$ 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~~ Unit to remain in an unacceptable condition for an extended period of time. Insert

1

4

TSTF-241

(continued)

The percent that $F^M_Q(Z)$ exceeds the limit can be determined from:

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F^M_Q(Z)}{\frac{CFQ}{P}\ K(Z) \over N(Z)} \right) - 1.0 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F^M_Q(Z)}{\frac{CFQ}{0.5}\ K(Z) \over N(Z)} \right) - 1.0 \right\} \times 100 \text{ for } P \leq 0.5$$

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INSERT

The maximum allowable power level initially determined by Required Action A.2.1 may be affected by subsequent determinations of $F_Q^M(Z)$ and would require power reductions within 15 minutes of the $F_Q^M(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^M(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

BASES

ACTIONS (continued)

A.2 (2.)

A reduction of the Power Range Neutron (M) Flux—High trip setpoints by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of ~~8~~ 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. Insert 1

(1) TSTF-95 TSTF-241

A.3 (2.)

Reduction in the (M) Overpower ΔT trip setpoints (value of K_4) by ≥ 1% (in ΔT span) for each 1% by which $F_Q(Z)$ 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. (2.) Insert 2

(1)

TSTF-241

A.4 (2)

Verification that $F_Q(Z)$ (M) has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 (2), ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

(1)

Move to page B 3.2-14

~~B.1~~ A.1

If ~~it is found that~~ the maximum ~~calculated~~ value of ~~$F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$,~~ $F_Q^M(Z)$ exceeds its specified limits ~~, there exists a potential for $F_Q(Z)$ to become excessively high if a normal operational transient occurs.~~ Reducing the AFD by ≥ 1% for each 1% by which (M) $F_Q(Z)$ exceeds its limit within the allowed Completion Time of ~~2 hours~~ 15 minutes, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded. The maximum AFD limits initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^M(Z)$ and would require AFD reductions within 15 minutes of the $F_Q^M(Z)$ determination, if necessary.

TSTF-99 TSTF-99 (1)

(continued)

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INSERT 1

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

INSERT 2

The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.2.3 may be affected by subsequent determinations of $F_Q^M(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^M(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^M(Z)$ would allow increasing the maximum Overpower ΔT trip setpoints.

BASES

ACTIONS (continued)

~~A~~.1 B

A.1, A.2.1, A.2.2, A.2.3, or A.2.4

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~~ MODE or condition in which the LCO requirements are not applicable. This is done by placing the ~~plant~~ unit in at least MODE 2 within 6 hours. 1 4

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~~ unit systems. 4

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 ~~and SR 3.2.1.2 are~~ is modified by a Note. ~~The Note applies during the first power ascension after a refueling.~~ It states that THERMAL POWER may be increased until ~~an equilibrium~~ for extended operation power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F^M_Q(Z)$ ~~and $F^W_Q(Z)$ are~~ is within ~~their~~ its specified limit after a power rise of more than 10% RTP over the THERMAL POWER at which ~~they were~~ it was last verified to be within specified limits. ~~Because $F^C_Q(Z)$ and $F^W_Q(Z)$ could not have previously been measured in this 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 $F^C_Q(Z)$ and $F^W_Q(Z)$ 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 $F^C_Q(Z)$ and $F^W_Q(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved.~~ In the absence of ~~these~~ this Frequency ~~conditions~~ condition, it is possible to increase power to RTP and operate for 31 days without verification of $F^M_Q(Z)$ ~~and $F^W_Q(Z)$~~. 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 was last measured. 1

(continued)

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(6)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.1

~~Verification that $F^C_Q(Z)$ is within its specified limits involves increasing $F^M_Q(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F^C_Q(Z)$. Specifically, $F^M_Q(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F^C_Q(Z) = F^M_Q(Z)$ [1.0815] (Ref. 4). $F^C_Q(Z)$ is then compared to its specified limits.~~ (1)

The limit with which $F^C_Q(Z)$ (M) is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR. and N(Z) (1)

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F^C_Q(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been (M) increased by ≥ 10% RTP since the last determination of $F^C_Q(Z)$, another evaluation of this factor is required ~~[12]~~ hours after achieving equilibrium (M) conditions at this higher power level (to ensure that $F^C_Q(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits). (1) (3) (1)

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the ~~plant is operated in~~ unit accordance with the Technical Specifications (TS). (9)

~~SR 3.2.1.2~~ (1)

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called ~~W~~(Z). (N) ~~Multiplying~~ (1) (1)

(continued)

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(6)

BASES

SURVEILLANCE REQUIREMENTS

~~the measured total peaking factor, $F^C_Q(Z)$, by W(Z) gives the~~

~~SR 3.2.1.2 (continued)~~ (1)

~~maximum $F_Q(Z)$ calculated to occur in normal operation, $F^W_Q(Z)$.~~

~~The limit with which $F^W_Q(Z)$ is compared varies inversely with power and directly with the function K(Z) provided in the COLR.~~ (1)

~~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. $F^C_Q(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height: multiple (4)

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 measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F^W_Q(Z)$ is evaluated ~~and found to be within its limit~~, an evaluation of the expression below is required to account for any increase to $F^M_Q(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation. TSTF-97 (1) (M) (M) (1)

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression (M) (1)

(1)

$$\text{maximum over z} \left[\frac{F^C_Q(Z)}{K(Z)} \right],$$

it is required to meet the $F_Q(Z)$ limit with the last $F^C_Q(Z)$ increased by a factor ~~of [1.02]~~, or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements (M) (M) (M) (1) (3) (1) the appropriate TSTF-98

(continued)

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BASES

prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection. 1

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is 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, [12] hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels. 1

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

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

REFERENCES

1. 10 CFR 50.46, 1974.
2. Regulatory Guide 1.77, Rev. 0, May 1974. 4
3. 10 CFR 50, Appendix A, GDC 26. UFSAR Section 3.1.22. 5
5 4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988. 1

4. Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications, VEP-NE-1-A, March 1986. 1

VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient."

(continued)

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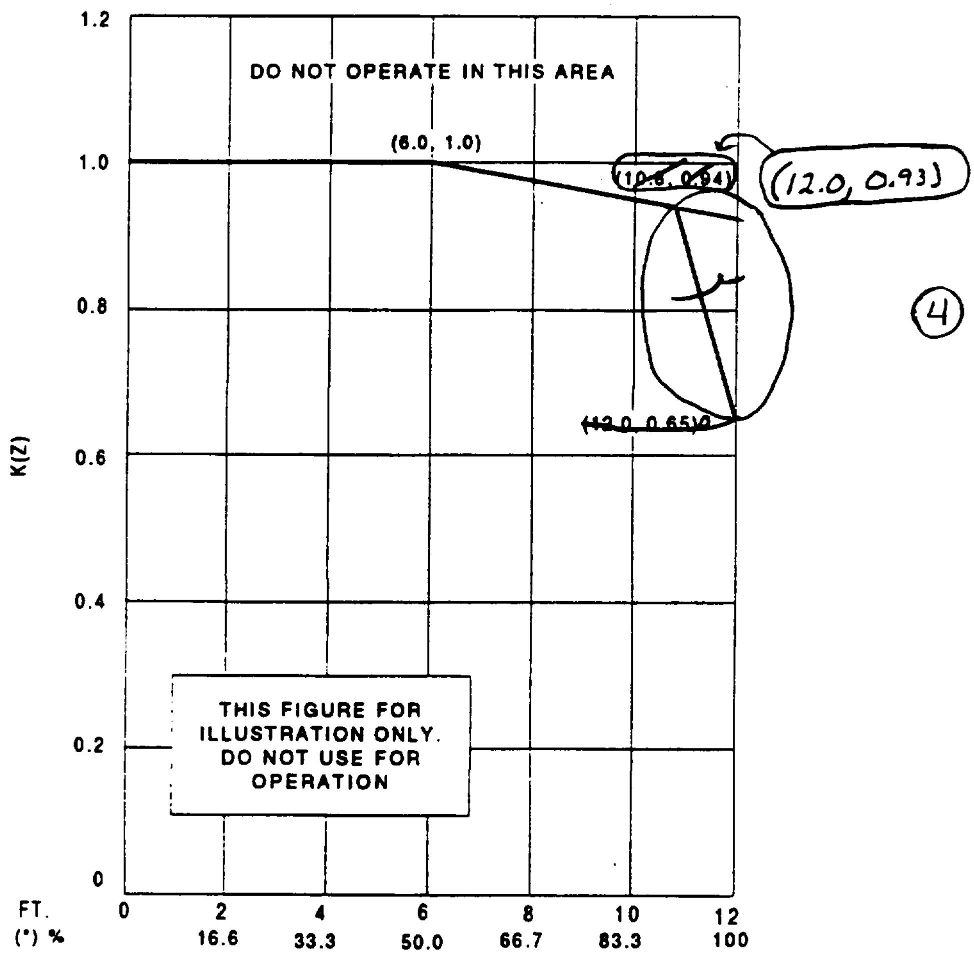
(6)



*For core height of 12 feet

(6)

Figure B 3.2.1~~B~~-1 (page 1 of 1)
K(Z) - Normalized Fo(Z) as a Function of Core Height

Rev.0

1. The Bases are revised to reflect the Relaxed Power Distribution Control (RPDC) methodology and changes made to the ITS to implement that methodology.

2. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference 10 CFR 50.36.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. North Anna Units 1 and 2 were designed and constructed on the basis of the proposed General Design Criteria, published in 1966. Since February 20, 1971, when the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50, were published, the Company attempted to comply with the intent of the newer criteria to the extent practical, recognizing previous design commitments. The NRC's Safety Evaluation Report for North Anna Units 1 and 2 reviewed the plant against 10 CFR Part 50, Appendix A and concluded that the facility design conforms to the intent of the newer criteria. The North Anna UFSAR contains discussions comparing the design of the plant to the 10 CFR 50, Appendix A, General Design Criteria. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

6. ITS 3.2.1A, $F_Q(Z)$ (F_{xy} Methodology), Bases are deleted. ITS 3.2.1B, $F_Q(Z)$ (F_Q Methodology) is used as the model for the North Anna ITS.

7. Editorial correction of the Bases.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

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. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

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

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio ~~to [1.3] using the [W3] CHF correlation.~~ to a value greater than the design limits. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

7

(continued)

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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.

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;

Unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm

c. During an ejected rod accident, the energy deposition to ~~the fuel must not exceed 280 cal/gm~~ (Ref. ~~1~~); and 1 2 1

Insert

d. ~~Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.~~ 3

temperature, and pressure

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 of [1.3] using the [WRB-1] CHF correlation~~. ~~This~~ value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB. 8 7

to a which

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

(continued)

Rev. 0

INSERT

The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, 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.

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). ①

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," ~~and~~ LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)." TSTF-136 ⑧

and LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits."

$F^N_{\Delta H}$ 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 1 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 ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). ④

LCO

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

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the ~~least heat removal capability and thus the~~ highest probability for a DNB. 3⑧

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 margin for higher radial peaking from reduced

(continued)

Rev. 0

BASES

LCO (continued)

thermal feedback and greater control rod insertion at low power levels. ~~The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.~~ (5)

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. ~~Specifically,~~ ~~the~~ design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have ~~significant~~ sufficient margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes. (2)

ACTIONS

~~A.1.1~~ A.1 and A.2 TSTF-240

~~With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its 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 for an extended period of time.~~ TSTF-240

Condition A is (4) ~~modified by a~~ Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, ~~if power is not reduced because this Required Action is completed within the 4 hour time period,~~ even if $F^N_{\Delta H}$ is restored to within limits, Required Action A.2 nevertheless requires another measurement and calculation (3) of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1. (4) TSTF-240

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding TSTF-240

(continued)

Rev. 0

BASES A.1 and A.2

ACTIONS A.1.1 (continued)

75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.~~2~~ 3 is performed if power ascension is delayed past 24 hours. TSTF-240

~~A.1.2.1 and A.1.2.2~~ TSTF-240

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. TSTF-240 Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state 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 power level from full power operation without allowing the ~~plant~~ unit 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.~~ TSTF-240 2 TSTF-240

The allowed Completion Time of ~~8~~ 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. This is a sensitive operation that may inadvertently trip the Reactor Protection System. TSTF-95 TSTF-240

A.~~2~~ 3 TSTF-240

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) 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 Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which TSTF-240 TSTF-240

(continued)

Rev. 0

BASES

ACTIONS

A.2 3 (continued) TSTF-240

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

A.3 4 TSTF-240

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 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.

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 2 through A.3 4 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems. TSTF-240 unit 2

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The ~~measured value of~~ $F^N_{\Delta H}$ ~~must be multiplied by~~ limit contains an allowance of 1.04 to account for 6

(continued)

Rev. 0

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

measurement uncertainty ~~before making comparisons to the $F^N_{\Delta H}$ limit~~. (6)

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.

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

REFERENCES

1. ~~Regulatory Guide 1.77, Rev. [0], May 1974.~~ (2)
2. ~~10 CFR 50, Appendix A, GDC 26.~~ (3)
3. 10 CFR 50.46.

UFSAR Section 3.1.22.

VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient."

Rev 0

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2 BASES, $F^N_{\Delta H}$

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The description of GDC 26 is revised. The existing description is not consistent with GDC 26. A correct discussion from the ITS 3.2.1 Bases is substituted. North Anna Units 1 and 2 were designed and constructed on the basis of the proposed General Design Criteria, published in 1966. Since February 20, 1971, when the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50, were published, the Company attempted to comply with the intent of the newer criteria to the extent practical, recognizing previous design commitments. The NRC's Safety Evaluation Report for North Anna Units 1 and 2 reviewed the plant against 10 CFR Part 50, Appendix A and concluded that the facility design conforms to the intent of the newer criteria. The North Anna UFSAR contains discussions comparing the design of the plant to the 10 CFR 50, Appendix A, General Design Criteria. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

4. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference 10 CFR 50.36.

5. ITS LCO 3.2.2 Bases state, "The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in Thermal Power." This sentence is removed. The LCO Bases state, "$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR." Part of the relationship specified in the COLR describes how the $F^N_{\Delta H}$ limit changes as a function of power. Describing part of the $F^N_{\Delta H}$ limit relationship in the Bases is inconsistent and does not provide any value without the rest of the relationship contained in the COLR. Therefore, the sentence is removed.

6. The Bases are revised to reflect the North Anna $F^N_{\Delta H}$ limit. The Bases state that the measured value of $F^N_{\Delta H}$ must be increased by 1.04 to account for measurement uncertainty. At North Anna, the $F^N_{\Delta H}$ limit includes 1.04 adjustment for measurement uncertainty. Therefore, adjusting the measured value is not necessary.

7. The Bases are revised to reflect the North Anna DNB limits and correlation. The North Anna safety analyses utilize different DNB limits for various analyses, so a specific value is not provided in the Bases. Also, the correlation used is subject to change and it is an analytical detail that does not add to the understanding of the $F^N_{\Delta H}$ specification. Therefore, this information is not specified in the Bases.

8. The Bases are changed to present correct and complete information.

③

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3B AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC) Methodology)~~

③

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.

Relaxed Power Distribution Control (RPDC)

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 invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity. ①

(Insert from Page B 3.2-41)

TSTF-110

~~Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.~~

①

~~The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.~~

(continued)

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3

BASES (continued)

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.

RPDC

The RAOC methodology (Ref. 2 10) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional 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. 1 5

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 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident. The most important Condition 2 events are uncontrolled bank rod withdrawal and , excessive heat removal and boration or dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints. 1

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

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 or automatic motion of control banks. The automatic motion

(continued)

Rev. 0

The LCO is modified by a Note which states that AFD shall be considered outside its limit when two or more OPERABLE excore channels indicate AFD to be outside its limit.

③

BASES

LCO (continued)

of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

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. ⑤

The AFD limits are provided in the COLR. Figure B 3.2.3B-1 shows typical RPDC AFD limits. The AFD limits for RPDC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution. ③ ①

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits. ⑥

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 RPDC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES. ①

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

(continued)

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3

BASES

ACTIONS

A.1 (continued)

30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging ~~plant~~ unit systems.

5

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

Move to Page 3.2-38

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 immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

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This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits~~, and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE,~~ the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

TSTF-110

REFERENCES

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

1 ~~2~~. ~~R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10217(NP), June 1983.~~ 1

K.L. Basehore et al., "Virginia Power Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications, VEP-NE-1-A, March 1986.

2 ~~3~~. U FSAR, Chapter ~~[15]~~ 7. 1 4 5

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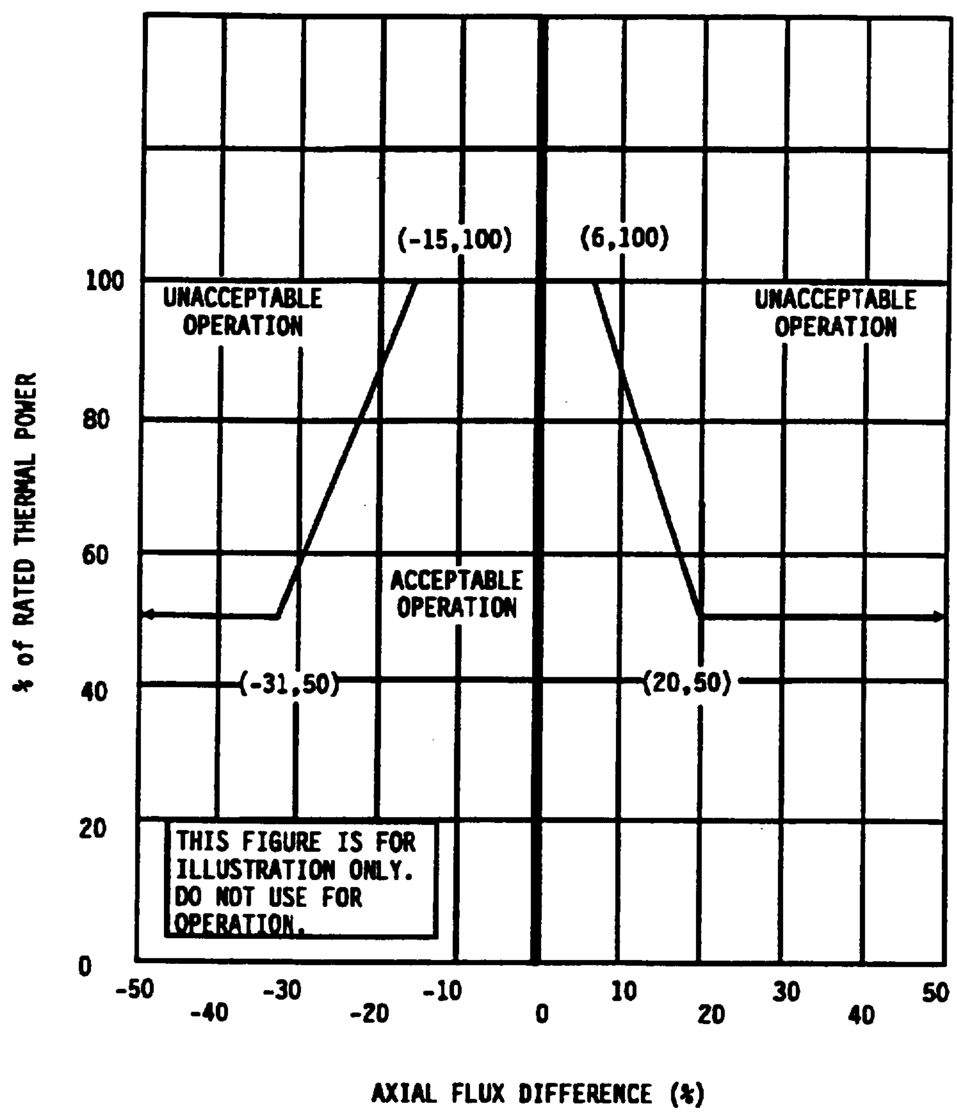

Figure B 3.2.3B-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

3

Rev. 0

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3 BASES, AFD

1. The Bases of ITS 3.2.3B, AFD (RAOC Methodology) is revised to reflect the Company Relaxed Power Distribution Control (RPDC) methodology and associated FQ Surveillance Technical Specifications approved by the NRC in VEP-NE-1-A, March, 1986. Terminology, descriptions of analytical methods, and references are changed as needed.

2. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference 10 CFR 50.36.

3. ITS ITS 3.2.3A, AFD (CAOC Methodology), Bases are deleted. ITS ITS 3.2.3B, AFD (RAOC Methodology) is used as the model for the North Anna ITS.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

6. The Bases are modified to describe the LCO Note in accordance with the ITS Writer's Guide.

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 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and (6) LCO 3.1.~~7~~6, "Control Rod 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 within the bounds used in the safety analyses. TSTF-136

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;

Unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm

c. During an ejected rod accident, the energy deposition to ~~the fuel must not exceed 280 cal/gm~~ (Ref. 2); and (1)

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control 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

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

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 ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) ②

LCO

The QPTR limit of 1.02, at 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 ③

(continued)

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BASES

ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

Insert

TSTF-241

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. 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.

TSTF-241

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of 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 $F^N_{\Delta H}$ 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.

after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1

Unit

TSTF-241

①

Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping.

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

(continued)

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INSERT

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to the revised limit.

BASES

ACTIONS

A.4 (continued)

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 to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

normalized to restore QPTR to within limits

Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00.

If the QPTR has exceeded 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 ~~recalibrated to show a zero QPTR~~ 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. Two Notes.

TSTF 241

Note 1

(R) Required Action A.5 is modified by ~~a Note that~~ states that the QPTR is not ~~zeroed out~~ 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). ~~This Note is~~ intended to prevent any ambiguity about the required sequence of actions.

(3)

Insert

These Notes are

A.6

Restored to within limits

Once the flux tilt is ~~zeroed out~~ (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 ~~at RTP~~ is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of ~~reaching RTP~~. As an added precaution, if the

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reaching equilibrium conditions at RTP

(continued)

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INSERT

Note 2 states that if Required Action A.5 is performed, the Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6.

BASES

ACTIONS

A.6 (continued)

Equilibrium conditions at

core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours ~~of the time when the ascent to power was begun~~. 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. TSTF-241

after increasing power above the limit of Required Action A.1

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been ~~calibrated to show zero tilt~~ (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 ~~calibrated to show zero tilt~~ and the core returned to power. TSTF-241 TSTF-241

Normalized to restore QPTR to within limits

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. 3 1

unit

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 ~~if more than one input from Power Range Neutron Flux channels are inoperable~~. TSTF-241 TSTF-109

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

takes into account other information and alarms available to the operator in the control room.

within its limits. The Frequency of 7 days ~~when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.~~ TSTF-110

~~When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because~~ 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. TSTF-110

SR 3.2.4.2

Until 12 hours after

not

This Surveillance is ~~modified by a Note, which states that it is~~ required ~~only when~~ the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ~~≥~~ 75% RTP. TSTF-109

⑦ TSTF-241

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. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are 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 ~~for three and four loop cores~~. ③

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

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

incore monitoring of

TSTF-241

core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

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 result may be compared against previous flux maps 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 flux map data.

REFERENCES

1. 10 CFR 50.46.

~~2. Regulatory Guide 1.77, Rev [0], May 1974.~~ ①

~~3. 10 CFR 50, Appendix A, GDC 26.~~ ①

3. UFSAR Section 3.1.22.

2. VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient."

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1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference 10 CFR 50.36.

3. Editorial changes made for consistency with the ITS Writer's Guide.

SECTION 3.2 - POWER DISTRIBUTION LIMITS

CURRENT TECHNICAL SPECIFICATIONS

MARKUP AND DISCUSSION OF CHANGES

UNIT 1

A.1

ITS 3.2.1

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

6-7-91

ITS

LIMITING CONDITION FOR OPERATION

, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR.

LCO 3.2.1

3.2.2 $F_Q(Z)$ ~~shall be limited by the following relationships:~~

~~$F_Q(Z) \le (\frac{CFQ}{P})$ [K(Z)] for P > 0.5~~

~~$F_Q(Z) \le (\frac{CFQ}{0.5})$ [K(Z)] for P ≤ 0.5~~

~~where CFQ = the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT,~~

~~$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and~~

~~K(Z) = the normalized F_Q limit as a function of core height specified in the CORE OPERATING LIMITS REPORT.~~

A.2

APPLICABILITY: MODE 1.

A.5

ACTION: M

After each $F_Q^M(Z)$ determination,

With $F_Q(Z)$ exceeding its limit:

A.3

72

Action A.2.1
Action A.2.2
Action A.2.3

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoint (value of K_4) has been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit.

L.1

LA.1

Action A.2.4

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

L.2

Insert Proposed Action B

M.1

(A.1)

~~4.2.2.1 The provisions of Specification 4.0.4 are not applicable.~~ A.4

M 4.2.2.2 $F_Q^M(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by: A.3

a. ~~Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~ LA.2

b. ~~Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.~~ LA.3

c. ~~Satisfying the following relationship:~~

$$\cancel{F_Q^M(z) \leq \frac{CFQ \times K(z)}{P \times N(z)} \text{ for } P > 0.5}$$

$$\cancel{F_Q^M(z) \leq \frac{CFQ \times K(z)}{N(z) \times 0.5} \text{ for } P \leq 0.5}$$

~~where $F_Q^M(z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, and N(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.7.~~ LA.4

SR 3.2.1.1 d. Measuring $F_Q^M(z)$ according to the following schedule:

SR 3.2.1.1 1st Frequency 1. Upon achieving equilibrium conditions after exceeding the THERMAL POWER at which $F_Q(Z)$ was last determined by 10% or more of RATED THERMAL POWER*, or

SR 3.2.1.1 2nd Frequency 2. At least once per 31 effective full power days, whichever occurs first.

or

3. Once after each refueling prior to THERMAL POWER exceeding 75% RTP M.3

SR 3.2.1.1 Note e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

SR Note *During power escalation, the power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

A.1

ITS

SURVEILLANCE REQUIREMENTS (Continued) an appropriate factor 6-7-91

SR 3.2.1.1 Note

1. $F_Q^M(z)$ shall be increased by ~~2%~~ over that specified in 4.2.2.2.c, or

LA.7

SR 3.2.1.1 Note

2. $F_Q^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

~~f. With the relationships specified in 4.2.2.2.c above not being satisfied:~~

~~1. Calculate the percent $F_Q(Z)$ exceeds its limit by subtracting one from the measurement/limit ratio and multiplying by 100:~~

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{CFQ \times K(z)}{P \times N(z)}} \right) - 1 \right\} \times 100 \text{ for } P \geq 0.5$$

LA.5

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{CFQ \times K(z)}{0.5 \times N(z)}} \right) - 1 \right\} \times 100 \text{ for } P < 0.5$$

2. Either of the following actions shall be taken:

Action A.1

a. Power operation may continue provided the AFD limits of Specification 3.2.1 are reduced 1% AFD for each percent $F_Q(Z)$ exceeded its limits, ~~or~~ within 15 minutes

M.2

b. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated above.

~~g. The limits specified in 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:~~

~~1. Lower core region 0 to 15 percent inclusive.~~

~~2. Upper core region 85 to 100 percent inclusive.~~

LA.6

~~4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.~~

LA.3

UNIT 2

(A.1)

ITS 3.2.1

POWER DISTRIBUTION LIMITS

6-7-91

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

ITS

LIMITING CONDITION FOR OPERATION

, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR.

LCO 3.2.1

3.2.2 $F_Q(Z)$ ~~shall be limited by the following relationships:~~

~~$F_Q(Z) \le \left(\frac{CFQ}{P}\right)$ [K(Z)] for P > 0.5~~

~~$F_Q(Z) \le \left(\frac{CFQ}{0.5}\right)$ [K(Z)] for P ≤ 0.5~~

~~where CFQ = the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT,~~

~~$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and~~

~~K(Z) = the normalized F_Q limit as a function of core height specified in the CORE OPERATING LIMITS REPORT.~~

(A.2)

APPLICABILITY: MODE 1.

ACTION:

After each $F_Q^M(Z)$ determination,

(A.5)

With $F_Q(Z)$ exceeding its limit:

(M)

(A.3)

Action A.2.1

Action A.2.2

Action A.2.3

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 (72) hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints ~~(value of K_4)~~ have been reduced at least 1% ~~(in ΔT span)~~ for each 1% $F_Q(Z)$ exceeds the limit.

(L.1)

(LA.1)

Action A.2.4

b. ~~Identify and~~ correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

(L.2)

Insert Proposed Action B

(M.1)

page 1 of 3

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(A.1)

ITS

SURVEILLANCE REQUIREMENTS

~~4.2.2.1 The provisions of Specification 4.0.4 are not applicable.~~

(M) 4.2.2.2 $F_Q^V(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by: (A.4) (A.3)

~~a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~ (LA.2)

~~b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.~~ (LA.3)

~~c. Satisfying the following relationship:~~

$$\cancel{F_Q^M(z) \le \frac{CFQ \times K(z)}{P \times N(z)} \text{ for } P > 0.5}$$

$$\cancel{F_Q^M(z) \le \frac{CFQ \times K(z)}{N(z) \times 0.5} \text{ for } P \le 0.5}$$

~~where $F_Q^M(z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, and N(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.7.~~ (LA.4)

SR 3.2.1.1 d. Measuring $F_Q^M(z)$ according to the following schedule:

SR 3.2.1.1 1st Frequency 1. Upon achieving equilibrium conditions after exceeding the THERMAL POWER at which $F_Q(Z)$ was last determined by 10% or more of RATED THERMAL POWER*, or

SR 3.2.1.1 2nd Frequency 2. At least once per 31 effective full power days, whichever occurs first.

Or
3. Once after each refueling prior to THERMAL POWER exceeding 75% RTP. (M.3)

e. With measurements indicating

SR 3.2.1.1 Note

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)}\right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

SR NOTE *During power escalation, the power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

ITS 3.2.1

A.1

ITS

SURVEILLANCE REQUIREMENTS (Continued)

an appropriate factor

SR 3.2.1.1 Note 1. $F_Q^M(z)$ shall be increased by ~~2%~~ over that specified in 4.2.2.2.c, or LA.7

SR 3.2.1.1 2. $F_Q^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$ is not increasing.

~~f. With the relationships specified in 4.2.2.2.c above not being satisfied:~~

~~1. Calculate the percent $F_Q(Z)$ exceeds its limit by subtracting one from the measurement/limit ratio and multiplying by 100:~~

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{CFQ \times K(z)}{P \times N(z)}} \right) - 1 \right\} \times 100 \text{ for } P \geq 0.5$$

LA.5

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{CFQ \times K(z)}{0.5 \times N(z)}} \right) - 1 \right\} \times 100 \text{ for } P < 0.5$$

2. Either of the following actions shall be taken:

Action A.1 a. Power operation may continue provided the AFD limits of Specification 3.2.1 are reduced 1% AFD for each percent $F_Q(Z)$ exceeded its limit, within 15 minutes or M.2

b. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated above.

~~g. The limits specified in 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:~~ LA.6

~~1. Lower core region 0 to 15 percent inclusive.~~

~~2. Upper core region 85 to 100 percent inclusive.~~

~~4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.~~ LA.3

ADMINISTRATIVE CHANGES

A.1 In the conversion of the North Anna Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 1, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.2.2 provides the limit for $F_Q(Z)$. The LCO provides two equations, which give the $F_Q(Z)$ limit for power > 50% RTP and power ≤ 50% RTP. ITS 3.2.1 does not contain these equations.

This change is acceptable because the technical requirements have not changed. CTS Surveillance 4.2.2.2 states "$F_Q^M(Z)$ shall be evaluated to determining if $F_Q(Z)$ is within limit by: . . . c. Satisfying the following relationship" and provides two equations. These equations for $F_Q^M(Z)$ are always more limiting than the equations presented in the LCO. Under CTS 4.0.1 and ITS SR 3.0.1, failure to meet the SR results in failure to meet the LCO. Therefore, the equations presented in the LCO are never limiting. In the ITS, the equations presented in the CTS Surveillance 4.2.2.2 are used to establish the LCO limit. This change is designated as administrative because it eliminates information from the CTS that is not used.

A.3 CTS 3.2.2 provides a limit for $F_Q(Z)$. The Actions for CTS 3.2.2 apply when $F_Q(Z)$ exceeds its limit. ITS 3.2.1 states, "$F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limit specified in the COLR." The ITS Condition is, "$F_Q^M(Z)$ not within limit." ITS SR 3.2.1.1 requires verification that $F_Q^M(Z)$ is within its limit. This changes the CTS by stating the limited value as $F_Q^M(Z)$ instead of $F_Q(Z)$.

This change is acceptable because the requirements have not changed. CTS SR 4.2.2.2, which is used to determine if $F_Q(Z)$ is within its limit, is written in terms of the measured $F_Q(Z)$, given as $F_Q^M(Z)$. The value used to determine if $F_Q(Z)$ is within its limit in CTS SR 4.2.2.2.c is $F_Q^M(Z)$. Therefore, the ITS use of $F_Q^M(Z)$ is consistent with the CTS limits. This change is designated as administrative because it does not result in a technical change to the specifications.

A.4 CTS 4.2.2.1 states, "The provisions of Specification 4.0.4 are not applicable." The ITS does not include this statement.

The purpose of a Specification 4.0.4 exception is to allow the plant to enter the MODE of applicability without performing the required Surveillances. This change is acceptable because the CTS Specification 4.0.4 exception is not used. CTS 4.2.2.2 is modified by a Note which states, "During power escalation, the power level may be increased until a

power level for extended operation has been achieved and a power distribution map obtained." Therefore, the CTS Surveillance Note provides the allowance to enter MODE 1 and increase power without performing the Surveillance. This serves the same purpose as the Specification 4.0.4 exception. The ITS does not need the exception because ITS Surveillance 3.2.1.1 contains the same Note as the CTS Surveillance. This change is designated as administrative because it eliminates a CTS provision which is not used.

A.5 ITS 3.2.1, Action A.2.1, A.2.2, and A.2.3 state that the Required Actions must be taken "after each $F_Q^M(Z)$ determination." CTS 3.2.2, Action a, does not explicitly state this requirement.

This change is acceptable because it does not result in a technical change to the specifications. The CTS is understood to apply after each measurement of $F_Q^M(Z)$. This change is designates as administrative because it does not result in a technical change to the CTS.

<u>MORE RESTRICTIVE CHANGES</u>

M.1 CTS 3.2.2 does not contain an Action to follow if the provided Actions or Completion Times are not followed. Therefore, CTS 3.0.3 would be entered which would require the plant to be in MODE 2 within 7 hours. ITS 3.2.1, Action B, states that when the Required Action and associated Completion Time is not met, the plant must be in MODE 2 within 6 hours. This changes the CTS by providing 6 hours vice 7 hours to be in MODE 2.

This change is acceptable because, based on operating experience, 6 hours is a reasonable time to be in MODE 2 from full power operation in an orderly manner and without challenging plant systems. This change is designated as more restrictive because the ITS allows less time to be in MODE 2 than does the CTS.

M.2 CTS 3.2.2, Action f.2.a, states that power operation may continue with $F_Q^M(Z)$ outside its limit provided the AFD limits are reduced 1% for each percent $F_Q(Z)$ exceeded its limit. ITS 3.2.1, Action A.1 requires the AFD limits to be reduced $\geq$ 1% for each 1% $F_Q^M(Z)$ exceeds its limit within 15 minutes. This changes the CTS by providing a Completion Time for an action which does not have a Completion Time in the CTS.

This change is acceptable because it is appropriate to provide a time to complete the reduction of the AFD limits. CTS Action 3.2.2.a requires a reduction in power is $F_Q^M(Z)$ exceeds its limit. This action to reduce the AFD limits is equivalent, in that both are mitigating actions for the limit violation. Therefore, equivalent Completion Times are appropriate. This change is designed as more restrictive because it applies a completion time when none existed in the CTS.

M.3 CTS 4.2.2.2.d requires $F_Q^M(Z)$ to be measured upon achieving equilibrium conditions after exceeding the THERMAL POWER at which $F_Q(Z)$ was last determined by 10% or more of RATED THERMAL POWER or at lease once per 31 EFPD. ITS SR 3.2.1.1 contains the same requirements, but also requires $F_Q^M(Z)$ to be verified to be within its limit once after each refueling prior to THERMAL POWER exceeding 75% RTP. This changes the CTS by adding a new Surveillance Frequency.

This change is acceptable because it provides an appropriate verification of $F_Q^M(Z)$ prior to proceeding to full power. Without this requirement, it would be possible to progress to full power without verifying $F_Q^M(Z)$. This change will not result in a change in operation because $F^N_{\Delta H}$ is required to be measured after each refueling prior to THERMAL POWER exceeding 75% RTP, and $F_Q^M(Z)$ is measured each time $F^N_{\Delta H}$ is measured. This change is designed as more restrictive because it applies a Frequency which did not exist in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 3.2.2, Action a, states than when $F_Q(Z)$ is exceeding its limit, POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoint (value of K_4) has been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit. ITS 3.2.1, Required Action A.2.3 states, "Reduce Overpower ΔT trip setpoints $\geq$ 1% for each 1% $F_Q^M(Z)$ exceeds limit." This changes the CTS by eliminating the parenthetical phrases, "(value of K_4)" and "(in ΔT span)" and placing the information in the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to lower the Overpower ΔT setpoint. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.2.2.2.a states that $F_Q(Z)$ shall be evaluated to determine if F_Q

is within its limit by using the moveable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER. The ITS does not contain a similar statement and this information appears in the ITS Bases. This changes the CTS by moving information to the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The information is a statement of fact, not a requirement. The moveable incore detector system is used to measure core power distribution, including $F_Q(Z)$. The $F_Q(Z)$ specification is applicable in MODE 1, which is THERMAL POWER ≥ 5% RTP, so measuring $F_Q(Z)$ at less than 5% RTP is not required. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.3 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.2.2.2.b states that the measured $F_Q(Z)$ must be increased by 3% to account for manufacturing tolerances and further increased by 5% for measurement uncertainties. CTS 4.2.2.3 states that when $F_Q(Z)$ is measured for reasons other than meeting the requirements of Surveillance 4.2.2.2, the measured $F_Q(Z)$ must be increased by 3% to account for manufacturing tolerances and further increased by 5% for measurement uncertainties. The ITS does not contain this requirement. This information is contained in the ITS Bases. This changes the CTS by moving information to the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS requires that the measured $F_Q(Z)$, labeled $F_Q^M(Z)$, be used to verify the limit is met. The ITS Bases state that $F_Q^M(Z)$ is based on increasing the measured $F_Q(Z)$ by 1.03 for manufacturing tolerances and by 1.05 for measurement uncertainties. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.4 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.2.2.2.c states that the measured $F_Q(Z)$ must meet a relationship provided in the Surveillance. The values for the principle components of the relationship, CFQ, K(Z), and N(Z), are specified in the COLR. ITS LCO 3.2.1 requires

that $F_Q(Z)$ meet this same relationship by stating, "$F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR." The equation for the relationship is located in the ITS Bases. This changes the CTS by moving information to the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS requires that the measured $F_Q(Z)$ be within the limit. The equation for calculating the limit is located in the Bases with the values specified in the COLR, but the requirement to meet the limit is unchanged. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.5 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.2.2.2.f states that with $F_Q^M(Z)$ not within limit, power operation may continue provided the AFD are reduced 1% AFD for each percent $F_Q(Z)$ exceeded its limits or by complying with the requirements of the specification for $F_Q(Z)$ exceeding its limit by the same percentage. CTS 4.2.2.2 also provides an equation for determining the percent by which $F_Q(Z)$ exceeds its limit. ITS 3.2.1 contains the same requirements described for the CTS, but does not contain an equation for determining the percentage by which $F_Q(Z)$ exceeds the limit. This equation is relocated to the ITS Bases. This changes the CTS by moving information to the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS requires determination of the percentage by which $F_Q(Z)$ exceeds its limit. This is a simple mathematical relationship and does not need to be placed in the specification. However, as a operator aid the equation is placed in the Bases. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.6 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.2.2.2.g states that the $F_Q(Z)$ limits are not applicable in the lower core region 0 to 15 percent inclusive, and the upper core region 85 to 100 percent inclusive. ITS 3.2.1 does not contain this information. This information is located in the ITS Bases. This changes the CTS by moving information to the Bases.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS requires that the $F_Q(Z)$ be determined as a function of core height. The Bases describes how the limit is calculated and, in this case, over what core heights the limit is applicable. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.7 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.2.1, Action e.1, states that $F_Q^M(Z)$ shall be increased by 2% over the measured amount when $F_Q^M(Z)$ / K(Z) (maximum over Z) is increasing. ITS SR 3.2.1.1 Note states that $F_Q^M(Z)$ shall be increased by an appropriate factor when $F_Q^M(Z)$ / K(Z) (maximum over Z) is increasing. This changes the CTS by relocating the amount by which $F_Q^M(Z)$ must be increased to the COLR.

The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits From the Technical Specifications, that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. The 2% penalty is based on the assumption that $F_Q^M(Z)$ will not increase more than 2% in the 31 day period between $F_Q^M(Z)$ measurements. However, cores at some Westinghouse PWRs have experienced increases of 6% in a 31 day period. The appropriate penalty factor will be calculated using NRC approved methodologies and provided in the COLR. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, Core Operating Limits Report. ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.2, Action a, states the Power Range Neutron Flux - High Trip setpoints must be reduced 1% for each 1% $F_Q^M(Z)$ exceeds its limit within 4 hours. ITS 3.2.1, Action A.2.2, requires the Power Range Neutron Flux - High trip setpoints be reduced ≥ 1% for each 1% $F_Q^M(Z)$ exceeds its limit within 72 hours. This changes the CTS by extending the Completion Time from 4 hours to 72 hours.

The purpose of CTS 3.2.2, Action a, is to reduce the Power Range Neutron Flux - High Trip Setpoints when $F_Q^M(Z)$ exceeds its limit to prevent inadvertently exceeding the maximum power level. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. Following a significant power reduction at least 24 hours is required to reestablish steady state xenon concentration and power distribution prior to taking a flux map and approximately 8 to 12 hours is required to take and analyze a flux map. If it is determined that $F_Q^M(Z)$ is still not within its limit, reducing the Power Range Neutron Flux - High Trip Setpoints takes approximately 2 hours per channel, with additional time required to preparation and channel restoration. Furthermore, setpoint changes should only be required for extended operation in this condition because of the risk of a plant trip during the adjustment. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.2.2, Action b, states that when $F_Q^M(Z)$ exceeds its limit, identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced power limit. ITS 3.2.1, Action A.2.4, requires verification that $F_Q^M(Z)$ is within its limit prior to increasing THERMAL POWER above the reduced power limit. This changes the CTS by eliminating the requirement to identify the cause of the out of limit condition prior to increasing power above the reduced power limit.

The purpose of CTS 3.2.2, Action b, is to ensure $F_Q^M(Z)$ is within its limit prior to increasing THERMAL POWER above the reduced power limit. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Identifying the cause of the out of limit condition is not required to restore compliance with the LCO. Identifying the cause of the condition is a function of the corrective action program required by 10 CFR 50, Appendix B. This change is designated as less

restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

UNIT 1

A.1

ITS 3.2.2

POWER DISTRIBUTION LIMITS

6-7-91

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

ITS

LIMITING CONDITION FOR OPERATION

LCO 3.2.2

3.2.3 $F^N_{\Delta H}$ shall be ~~limited by the following relationship:~~ within the limits specified in the COLR

LA.1

~~$F^N_{\Delta H} \leq$ CFDH [1 + PFDH (1-P)]~~

~~where CFDH = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT,~~

~~$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$~~

~~PFDH = The Power Factor Multiplier for $F^N_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT, and~~

~~$F^N_{\Delta H}$ = measured value of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.~~

LA.1

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit: Insert Proposed Condition A Note

M.1

Action A.1
Action A.2

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within ~~2~~ 4 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within ~~the next 4~~ 72 hours,

L.1

Action A.3
Action B

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next ~~2~~ 6 hours, and

L.2

c. ~~Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core~~ A.2

Action A.4

mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

Perform SR 3.2.2.1

Insert Proposed A.4 Note

A.3

A.2

Rev. 0

A.1

ITS 3.2.2

ITS

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit ~~by using the movable incore detectors to obtain a power distribution map~~: A.4

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. ~~The provisions of Specification 4.0.4 are not applicable.~~ A.5

UNIT 2

A.1

ITS 3.2.2

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

ITS

LIMITING CONDITION FOR OPERATION

LCO 3.2.2 3.2.3 $F^N_{\Delta H}$ shall be ~~limited by the following relationship:~~ within the limits specified in the COLR. LA.1

~~$F^N_{\Delta H} \leq$ CFDH [1 + PFDH (1-P)]~~

~~where CFDH = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT,~~

~~$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$~~ LA.1

~~PFDH = The Power Factor Multiplier for $F^N_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT, and~~

~~$F^N_{\Delta H}$ = measured value of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.~~

APPLICABILITY: MODE 1

ACTION:

Insert Proposed Condition A Note M.1

With $F^N_{\Delta H}$ exceeding its limit:

Action A.1
Action A.2
a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within ~~2~~ 4 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within ~~the next 4~~ 72 hours, L.1

Action A.3
Action B
b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next ~~2~~ 6 hours, and L.2

Action A.4
c. ~~Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit~~ at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to A.2

Perform SR 3.2.2.1

Insert Proposed A.4 Note A.3

A.2

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A.1

8-25-86

ITS

POWER DISTRIBUTION LIMITS

ACTION Continued

Action A.4 exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map: A.4

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. ~~The provisions of Specification 4.0.4 are not applicable.~~ A.5

ADMINISTRATIVE CHANGES

A.1 In the conversion of the North Anna Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 1, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.2.3, Action c states that with $F^N_{\Delta H}$ exceeding its limit, identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit. ITS 3.2.2, Action A, states that SR 3.2.2.1 shall be performed. SR 3.2.2.1 requires measurement of $F^N_{\Delta H}$. This changes the CTS by eliminating the statement that the cause of the out of limit condition must be identified and corrected prior to increasing power and the statement that $F^N_{\Delta H}$ must be demonstrated through incore mapping.

This change is acceptable because the requirements have not changed. Stating that the cause of the $F^N_{\Delta H}$ limit violation must be identified and corrected prior to increasing power (i.e., exiting the Action which required power reduction) is unnecessary. Restoration of compliance with the LCO is always an option and allows exiting the Action per ITS 3.0.2. Therefore, it does not have to be stated. Stating that $F^N_{\Delta H}$ must be measured with the incore mapping system is unnecessary, as $F^N_{\Delta H}$ can only be measured with the incore mapping system. Therefore, stating that $F^N_{\Delta H}$ must be measured (by invoking SR 3.2.2.1) means that the incore mapping system must be used. This change is designated as administrative because it does not result in technical changes to the specifications.

A.3 CTS 3.2.3, Action c, states that with $F^N_{\Delta H}$ exceeding its limit, $F^N_{\Delta H}$ must be measured prior to exceeding 50% RTP, 75% RTP, and within 24 hours of exceeding 95% RTP. ITS 3.2.2, Action A.4, contains the same requirements. ITS 3.2.2, Action A.4, is modified by a Note which states, "THERMAL POWER does not have to be reduced to comply with this Required Action." This modifies the CTS by adding a Note stating that THERMAL POWER does not have to be reduced to comply with the Action.

This change is acceptable because the requirements have not changed. The Note is included in the ITS to make clear that THERMAL POWER does not have to be reduced to perform the Action. For example, if $F^N_{\Delta H}$ exceeded its limit and power was reduced to 60% RTP before $F^N_{\Delta H}$ is demonstrated to be within its limit, under the Note THERMAL POWER does not have to be reduced to less than 50% RTP for a $F^N_{\Delta H}$ measurement. $F^N_{\Delta H}$ must be measured prior to exceeding 75% RTP and within 24 hours of exceeding

95% RTP. The Condition A is needed because the ITS contains a Note on ITS 3.2.3, Condition A, which states, "Required Actions A.3 and A.4 must be completed whenever Condition A is entered." The Condition A Note does not exist in the CTS and could be construed as requiring THERMAL POWER to be reduced to comply with Action A.4. The Condition A Note is described in DOC M.1. As a result, the Action A.4 Note makes the ITS and CTS actions consistent. This change is designated as administrative because it does not result in technical changes to the specifications.

A.4 CTS 4.2.3.1 states that $F^N_{\Delta H}$ shall be determined to be within its limit by using the moveable incore detectors to obtain a power distribution map. ITS SR 3.2.2.1 states that $F^N_{\Delta H}$ shall be verified to be within the limits specified in the COLR. This changes the CTS by eliminating the statement that $F^N_{\Delta H}$ must be determined by using the moveable incore detector system to obtain a power distribution map.

This change is acceptable because the requirements have not changed. Stating that $F^N_{\Delta H}$ must be measured by using the incore mapping system to obtain a power distribution map is unnecessary, as $F^N_{\Delta H}$ can only be measured with the incore mapping system to create a power distribution map. Therefore, eliminating a statement of the method that must be used to measure $F^N_{\Delta H}$ does not change the specifications. This change is designated as administrative because it does not result in technical changes to the specifications.

A.5 CTS 4.2.3.1.c states, "The provisions of Specification 4.0.4 are not applicable." The ITS does not include this statement.

The purpose of a Specification 4.0.4 exception is to allow the plant to enter the MODE of applicability without performing the required Surveillances. This change is acceptable because the CTS Specification 4.0.4 exception is not required in the ITS. CTS 4.2.3.1 is required to be performed prior to operation above 75% RTP after each fuel loading and once per 31 EFPD. Without the SR 4.0.4 exception, MODE 1 could not be entered without a measurement because the "once per 31 EFPD" Frequency would be violated under SR 4.0.4 because Surveillances must be met prior to entering the MODE of applicability. However, under the ITS, the Frequency "Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD <u>thereafter</u>," means that the 31 EFPD Frequency does not apply until after the 75% RTP measurement is performed. Therefore, the applicability of the SR is changed and MODE 1 can be entered without the SR being met. This change is designated as administrative because it does not result in technical changes to the specifications.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.2.3, Action c, states that with $F^N_{\Delta H}$ exceeding its limit, subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through incore mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, and within 24 hours after attaining 95% or greater RATED THERMAL POWER. However, under CTS 3.0.2, these measurements do not have to be completed if compliance with the LCO is reestablished. ITS 3.2.2 Condition A contains a Note which states, "Required Actions A.3 and A.4 must be completed whenever Condition A is entered." ITS Required Actions A.3 and A.4 require performance of a $F^N_{\Delta H}$ measurement every 24 hours and prior to exceeding 50% RTP, 75% RTP, and within 24 hours after THERMAL POWER $\geq$ 95% RTP. This changes the CTS by requiring the $F^N_{\Delta H}$ measurements to be made even if $F^N_{\Delta H}$ is restored to within its limit.

This change is acceptable because it establishes appropriate compensatory measurements for violation of the $F^N_{\Delta H}$ limit. As power is reduced under ITS Action A.1, the margin to the $F^N_{\Delta H}$ limit increases. Therefore, compliance with the LCO could be reestablished during the power reduction. Verifying that the limit is met as power is increased ensures that the limit continues to be met and does not remain unmeasured for 31 EFPD. This change is designated as more restrictive because it imposes requirements in addition to those in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.2.3 states that $F^N_{\Delta H}$ shall be limited by an equation, which is contained in the LCO. All of the parameters in the CTS equation are specified in the CORE OPERATING LIMITS REPORT (COLR). ITS LCO 3.2.2 states, "$F^N_{\Delta H}$ shall be within the limits specified in the COLR." This changes the CTS by relocating the equation to the COLR.

The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits From the Technical Specifications, that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still

retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. The ITS requires that $F^N_{\Delta H}$ be within the limit provided in the COLR. All of the parameters for the $F^N_{\Delta H}$ limit are located in the COLR. Moving the equation itself to the COLR does not change the requirement that the $F^N_{\Delta H}$ limit be met. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, Core Operating Limits Report. ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.3, Action a states that when $F^N_{\Delta H}$ exceeds its limit, reduce THERMAL POWER to less than 50% RTP within 2 hours and reduce the Power Range Neutron Flux - High trip setpoints to less than 55% of RTP within the next 4 hours. ITS 3.2.2, Actions A.1 and A.2 state than with $F^N_{\Delta H}$ not within this limit, reduce THERMAL POWER to ≤ 50% RTP within 4 hours and reduce the Power Range Neutron Flux - High trip setpoints to ≤ 55% RTP within 72 hours. This changes the CTS by allowing a 4 hour Completion Time to reduce power to ≤ 50% RTP and 72 hours to reduce the trip setpoint.

The purpose of CTS 3.2.3, Action a, is to reduce power, and, therefore, increase the margin to the $F^N_{\Delta H}$ limit, and to lower the trip setpoints to avoid inappropriately increasing power and violating the $F^N_{\Delta H}$ limit. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. The revised Completion Times allow reactor power to be reduced in a controlled manner without challenging operators, technicians, or plant systems. Following a significant power reduction at least 24 hours is required to reestablish steady state xenon concentration and power distribution prior to taking a flux map and approximately 8 to 12 hours is required to take and analyze a flux map. If it is determined that $F^N_{\Delta H}$ is still not within its limit, reducing the Power Range Neutron Flux - High Trip Setpoints takes approximately 2 hours per channel, with additional time required to preparation and channel restoration. Furthermore, setpoint changes should only be required for extended operation in this condition because of the risk of a plant trip during the adjustment. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L.2 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.3, Action b states that when $F^N_{\Delta H}$ exceeds its limit, demonstrate through incore mapping that $F^N_{\Delta H}$ is within its limit or reduce THERMAL POWER to less than 5% within the next 2 hours. ITS 3.2.2, Action B states that with the Required Action and associated Completion Time not met, be in MODE 2 within 6 hours. This changes the CTS by allowing a 6 hour Completion Time to reduce power to < 5% RTP.

The purpose of CTS 3.2.3, Action b, and ITS 3.2.2, Action B, is to reduce power when compliance with the $F^N_{\Delta H}$ limits cannot be obtained to a MODE in which the LCO is not applicable. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. The revised Completion Times allow reactor power to be reduced in a controlled manner without challenging operators, technicians, or plant systems. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

UNIT 1

ITS 3.2.3

A.1

ITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.3 3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER

ACTION:

Action A a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the CORE OPERATING LIMITS REPORT,

1. ~~Either restore the indicated AFD to within the limits within 15 minutes, or~~ A.2

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes ~~and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.~~ L.1

b. ~~THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the CORE OPERATING LIMITS REPORT.~~ A.3

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(A.1) ITS 3.2.3

ITS

LIMITING CONDITION FOR OPERATION (Continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days ~~when the AFD Monitor Alarm is OPERABLE, and~~

~~2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.~~

~~b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.~~

(L.2)

LCO Note

4.2.1.2 The indicated AFD shall be considered outside of its limit when at least 2 OPERABLE excore channels are indicating the AFD to be outside of the limits specified in the CORE OPERATING LIMITS REPORT.

UNIT 2

A.1

AXIAL FLUX DIFFERENCE (AFD)

ITS

LIMITING CONDITION FOR OPERATION

LCO 3.2.3

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the CORE OPERATING LIMITS REPORT,

1. ~~Either restore the indicated AFD to within the limits within 15 minutes, or~~ A.2

Action A

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes ~~and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.~~ L.1

b. ~~THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the CORE OPERATING LIMITS REPORT.~~ A.3

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

SR 3.2.3.1

1. At least once per 7 days ~~when the AFD Monitor Alarm is OPERABLE, and~~

2. ~~At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.~~

b. ~~Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.~~ L.2

ITS 3.2.3

A.1

ITS

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

SURVEILLANCE REQUIREMENTS (Continued)

LCO NOTE

4.2.1.2 The indicated AFD shall be considered outside of its limit when at least two OPERABLE excore channels are indicating the AFD to be outside of the limits specified in the CORE OPERATING LIMITS REPORT.

ADMINISTRATIVE CHANGES

A.1 In the conversion of the North Anna Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 1, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.2.1, Action a, states that with AFD outside its limit, restore the indicated AFD to within its limit within 15 minutes or reduce THERMAL POWER to less than 50% RTP within 30 minutes. ITS 3.2.3, Condition A, states that with AFD not within limits, reduce THERMAL POWER to less than 50% within 30 minutes. This changes the CTS by eliminating the action to restore AFD within its limit within 15 minutes.

This change is acceptable because the technical requirements have not changed. If AFD is not restored to within its limit within 15 minutes, no CTS Actions apply except to reduce power to less than 50% RTP within 30 minutes. Therefore, the action to restore AFD to within its limit within 15 minutes contains no requirement to take action. Both the CTS and the ITS require power to be reduced to less than 50% RTP within 30 minutes if AFD is not restored to within its limit. This change is designated as administrative because it does not result in a technical change to the specifications.

A.3 CTS 3.2.1, Action b, states, "THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the CORE OPERATING LIMITS REPORT." ITS 3.2.3 does not contain a similar requirement. This changes the CTS by eliminating a prohibition in the CTS.

This change is acceptable because the requirements have not changed. CTS 3.0.4 and ITS LCO 3.0.4 prohibit entering the MODE of applicability of a specification unless the requirements of the LCO are met. CTS 3.2.1 and ITS 3.2.3 are applicable in MODE 1 with THERMAL POWER ≥ 50%. Therefore, the Use and Application rules in the CTS and the ITS prohibit exceeding 50% of RATED THERMAL POWER without the LCO requirements met. CTS 3.2.1, Action b, is duplicative of CTS 3.0.4 and ITS LCO 3.0.4 and its elimination does not make a technical change to the specifications. This change is designated as administrative because it does not result in a technical change to the specifications.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.2.1, Action a, states that when AFD is not within its limit, reduce THERMAL POWER to less than 50% within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to ≤ 55% of RTP within the next 4 hours. ITS 3.2.3, Action A.1, requires THERMAL POWER to be reduced to less than 50% within 30 minutes when AFD is outside of its limit. This changes the CTS by eliminating the requirement to reduce the High Flux Trip Setpoint to ≤ 55% within 4 hours.

The purpose of CTS 3.2.1, Action a, is to reduce THERMAL POWER to the point at which the LCO is not longer applicable if AFD is not restored within its limit. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. With THERMAL POWER outside the applicability of the Specification, further actions are not required to ensure that the assumptions of the safety analysis are met. Increases in THERMAL POWER are governed by ITS LCO 3.0.4 which requires the LCO to be met prior to entering a MODE or other specified condition in which the LCO applies. Therefore, power increases are prohibited without the risk of changing Reactor Protection System setpoints during operation. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.2.1.1 requires the indicated AFD for each excore channel to be determined to be within its limits once per 7 days when the AFD Monitor is OPERABLE, and at least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status, and once per hour for the first 24 hours and once per 30 minutes thereafter when the AFD Monitor Alarm is inoperable. ITS SR 3.2.3.1 requires AFD to be verified within its limits for each

OPERABLE excore channel every 7 days. This changes the CTS by eliminating all AFD Surveillance Frequencies based on the OPERABILITY of the AFD Monitor.

The purpose of ITS 3.2.3 is to ensure that AFD is within its limit. This change is acceptable because the remaining Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the Frequency of monitoring AFD when the AFD Monitor is inoperable is unnecessary as inoperability of the alarm does not increase the probability that AFD is outside its limit. The AFD Monitor is for indication only. Its use is not credited in any safety analysis. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

UNIT 1

A.1

12-26-79

POWER DISTRIBUTION LIMITS

QUADRANT POWER TILT RATIO

ITS

LIMITING CONDITION FOR OPERATION

LCO 3.2.4

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER* — A.2

ACTION:

Condition A

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but ≤ 1.09: — L.1

1. Within 2 hours:

a) ~~Either reduce the QUADRANT POWER TILT RATIO to within its limit, or~~ — A.3

Action A.1

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 ~~and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.~~ — L.3

2. ~~Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.~~ — L.4

3. ~~Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~ — L.4

b. ~~With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:~~

1. ~~Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.~~ — L.1

2. ~~Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or~~

~~*See Special Test Exception 3.10.2.~~ — A.2

new — Insert Proposed Actions A.2, A.3, A.4, A.5, and A.6 — M.1

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ITS

POWER DISTRIBUTION Insert Proposed Action B (L.4)

LIMITING CONDITION FOR OPERATION (Continued)

~~reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.~~ (L.1)

~~3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~

~~c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:~~ (L.1)

~~1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.~~

~~2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified as 95% or greater RATED THERMAL POWER.~~

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

~~a.~~ Calculating the ratio at least once per 7 days ~~when the alarm is OPERABLE.~~

~~b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.~~ (L.2)

SR 3.2.4.2 4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the moveable incore detectors to confirm ~~that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map,~~ is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours. (LA.1)

Insert Proposed SR 3.2.4.1 Note 2 (L.5)

(A.1)

I TS 3.2.7

ITS

11-22-91

1.0 DEFINITIONS (Continued)

QUADRANT POWER TILT RATIO

See ITS Section 1.1

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

SR 3.2.4.1, Note 1

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be FULLY WITHDRAWN.

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SLAVE RELAY TEST

1.30 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

See ITS Section 1.1

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UNIT 2

(A.1)

8-21-80

ITS

POWER DISTRIBUTION LIMITS

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

LCO 3.2.4 3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02. (A.2)

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*

ACTION:

Condition A a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09: (L.1)

1. ~~Calculate the QUADRANT POWER TILT RATIO at least once per hour until:~~ (A.3)

(a) ~~Either the QUADRANT POWER TILT RATIO is reduced to within its limit, or~~

(b) ~~THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.~~ (L.4)

2. Within 2 hours:

a) ~~Either reduce the QUADRANT POWER TILT RATIO to within its limit, or~~ (A.3)

Action A.1 b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 ~~and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.~~ (L.3)

3. ~~Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2~~ (L.4)

~~*See Special Test Exception 3.10.2.~~ (A.2)

A.1

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ITS

hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL power within the next 4 hours.

4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

L.4

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until:

   (a) Either the QUADRANT POWER TILT RATIO is reduced to within its limit, or

   (b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

L.1

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

Insert Proposed Actions A.2, A.3, A.4, A.5, and A.6

M.1

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(A.1)

I TS 3.2.4

Insert Proposed Action B

(L.4)

I TS

POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION (Continued)

~~c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:~~

~~1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until:~~

~~(a) Either the QUADRANT POWER TILT RATIO is reduced to within its limit, or~~

~~(b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.~~

(L.1)

~~2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

~~3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.~~

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days ~~when the alarm is OPERABLE.~~

~~b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.~~

(L.2)

SR 3.2.4.2

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the moveable incore detectors to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

(LA.1)

Insert Proposed SR 3.2.4.1, Note 2

(L.5)

Rev. 0

(A.1)

ITS 3.2.4

ITS

1.0 DEFINITIONS (Continued)

See ITS Section 1.1

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

SR 3.2.4.1, Note 1

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be FULLY WITHDRAWN.

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SLAVE RELAY TEST

1.30 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

See ITS Section 1.1

ADMINISTRATIVE CHANGES

A.1 In the conversion of the North Anna Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 1, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 The Applicability of CTS 3.2.4 is modified by a footnote, designated "*", stating, "See Special Test Exception 3.10.2." ITS 3.2.4 Applicability does not contain the footnote or a reference to the Special Test Exception.

The purpose of the footnote reference is to alert the reader that a Special Test Exception exists which may modify the Applicability of the specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative because it does not result in technical changes to the specifications.

A.3 CTS 3.2.4, Action a.1.a (Unit 1) states that with QPTR > 1.02, within 2 hours reduce the QPTR to within its limit. CTS 3.2.4, Action a.1(a) and 2.a state that with QPTR > 1.02, calculate QPTR at least once per hour until QPTR is within its limit and within 2 hours reduce QPTR to within its limit. ITS 3.2.4 does not contain a Required Action stating QPTR must be calculated at least once per hour and QPTR must be reduced to within its limit.

This change is acceptable because the technical requirements have not changed. Restoration of compliance with the LCO is always an available Required Action and it is the convention in the ITS to not state such "restore" options explicitly unless it is the only action or is required for clarity. Monitoring a parameter that is outside its limit in order to determine if it has been restored to within its limit is a necessary action which must occur whether or not it is explicitly required by the TS. This change is designated as administrative because it does not result in technical changes to the specifications.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.2.4, Action a.1.b) (Unit 1) and Action a.2(b) (Unit 2) requires THERMAL POWER to be reduced at least 3% for every 1% QPTR exceeds 1.0 and allows a maximum of 24 hours of operation above 50% RTP with QPTR greater than the limit. ITS 3.2.4, Condition A, also requires THERMAL POWER to be reduced at least 3% for every 1% QPTR exceeds 1.0, but the ITS allows indefinite power operation above 50% RTP provided that QPTR is determined within 12 hours of achieving equilibrium conditions after the power reduction, $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within limit within

24 hours and every 7 days thereafter, and the safety analyses are reevaluated to confirm the results are still valid for the duration of operation under this condition prior to increasing power. If the reevaluation of the safety analyses confirms that the results remain valid, the ITS allows the excore detectors to be normalized to restore QPTR within limit provided that $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within limits within 24 to 48 hours after achieving equilibrium condition at RTP. This changes the CTS by requiring $F_Q(Z)$ and $F^N_{\Delta H}$ be verified, the safety analyses be reevaluated, and the excore detectors be normalized to restore QPTR to within the limits. The change eliminating the requirement to reduce power to less than 50% RTP is discussed in DOC L.4.

This change is acceptable because it provides an appropriate set of remedial actions for the indicated condition. With QPTR outside of its limit, the core power distribution is not necessarily unacceptable. QPTR is used as a readily available indicator of stable power distribution, but is not itself a critical core power distribution parameter. Should QPTR exceed its limit, verification that the critical core power distribution parameters, $F_Q(Z)$ and $F^N_{\Delta H}$, are within limit and verification that the safety analyses remain valid is sufficient to allow continued power operation. This change is designated as more restrictive because it adds additional actions to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS Surveillance 4.2.4.2 states that the QPTR shall be determined to be within limit when above 75 % RTP with one Power Range Channel inoperable by using the movable incore detector to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QPTR at least once per 12 hours. ITS SR 3.2.4.2 states, "Verify QPTR is within limit using the movable incore detectors." ITS SR 3.2.4.2 is modified by a Note which states, "Not required to be performed until 12 hours after input from one or more Power Range neutron Flux channels are inoperable with THERMAL POWER > 75% RTP." This changes the CTS by relocating the details of how the movable incore detector system is used to determine QPTR by moving the phrase "the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map" to the Bases of the Surveillance.

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to use the movable incore detector system

to determine QPTR. The specifics of how the movable incore detector system is used to gather the information and how the data is analyzed is a detail of Surveillance performance that is not required to be in the Technical Specifications. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in ITS Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4 states that the QPTR shall not exceed 1.02. CTS Action a provides actions for QPTR > 1.02 and ≤ 1.09 and CTS 3.2.4 actions b and c provide actions for QPTR > 1.09. CTS action b applies when QPTR > 1.09 due to misalignment of a RCCA and requires a power reduction of 3% RTP for every 1% QPTR exceeds 1.0 within 30 minutes and reduce power to < 50% RTP within 2 hours if QPTR is not restored to within limits. CTS action c applies when QPTR > 1.09 for any other reason and requires reducing power to < 50% RTP within 2 hours. ITS LCO 3.2.4 states that QPTR shall be ≤ 1.02. ITS 3.2.4 contains actions for QPTR > 1.02, but does not contain additional actions for QPTR > 1.09. This changes the CTS by eliminating additional actions for QPTR > 1.09.

The purpose of CTS 3.2.4 is provide appropriate compensatory measures for a Quadrant Power Tilt Ratio greater than that assumed in the accident analyses. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to restore compliance with the LCO. The Required Actions are consistent with safe operation under the specified Condition, considering other indications available to the operator, a reasonable time for restoring compliance with the LCO, and the low probability of a DBA occurring during the restoration period. The ITS Required Actions provided for a QPTR > 1.02 are also sufficient to address a QPTR > 1.09. Under the ITS, a QPTR of 1.09 would require THERMAL POWER to be reduced ≤ 73% RTP. This will provide sufficient thermal margin to account for the radial power distribution. In addition, the ITS requires $F_Q(Z)$ and $F^N_{\Delta H}$ to be verified to be within their limits within 24 hours and the safety analyses to be reevaluated to confirm the results remain valid for continued operation prior to increasing power. If these Actions are not completed, the ITS requires THERMAL POWER to be reduced ≤ 50% RTP within 4 hours. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.2.4.1 requires the QPTR to be verified to be within limit every 7 days with the QPTR alarm is OPERABLE and every

12 hours with the QPTR alarm is inoperable. ITS SR 3.2.4.1 requires verification that the QPTR is within limit every 7 days. This changes the CTS by eliminating the requirement to verify QPTR more frequently when the QPTR alarm is inoperable.

The purpose of CTS Surveillance 4.2.4.1 is to periodically verify that QPTR is within limit. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the frequency of QPTR verification when the QPTR alarm is inoperable is unnecessary as inoperability of the alarm does not increase the probability that the QPTR is outside its limit. The QPTR alarm is for indication only. Its use is not credited in any safety analysis. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.3 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4, Action a.1.b) (Unit 1) and Action a.2.(b) (Unit 2), states that when QPTR is not within its limit, reduce THERMAL POWER by at least 3% RTP for every 1% of indicated QPTR in excess of 1.0 and reduce the Power Range Neutron Flux - High Trip setpoints within the next 4 hours. ITS 3.2.4, Action A.1, requires THERMAL POWER to be reduced > 3% RTP for each 1% QPTR > 1.00. This changes the CTS by eliminating the requirement to reduce the High Flux Trip Setpoint.

The purpose of CTS 3.2.4, Action a, is to reduce THERMAL POWER to increase the margin to the core power distribution limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. With THERMAL POWER reduced by 3% for each 1% QPTR > 1.00, further actions are not required to ensure that THERMAL POWER is not increased. Power increases are administratively prohibited by the Technical Specifications without the risk of changing Reactor Protection System setpoints during operation. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.4 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4, Action a.2 (Unit 1) states that with QPTR ≥ 1.02 and ≤ 1.09, verify that QPTR is within its limit within 24 hours or reduce THERMAL POWER to less than 50% RTP within the next 2 hours and reduce the Power Range Neutron Flux - High Trip setpoints to ≤ 55% RTP within the next 4 hours. CTS 3.2.4, Action a.1(a) and a.3 (Unit 2) states that with QPTR ≥ 1.02 and ≤ 1.09, calculate QPTR at least once per hour until THERMAL POWER is reduced to less than 50% of RTP and verify that QPTR is within its limit within 24 hours or reduce THERMAL POWER to less than 50% RTP within the next 2 hours and reduce the Power

Range Neutron Flux - High Trip setpoints to ≤ 55% RTP within the next 4 hours. CTS 3.2.4, Action a.3 (Unit 1) and a.4 (Unit 2) state that the cause of the out of limit QPTR must be identified and corrected prior to increasing THERMAL POWER and subsequent operation above 50% RTP can proceed provided that the QPTR is verified to be within its limit at least once per hours for 12 hours or until verified acceptable at 95% or greater RTP. ITS 3.2.4, Action B, states that with the Required Actions and Associated Completion Times of Condition A not met, reduce THERMAL POWER to ≤ 50% RTP within 4 hours. This changes the CTS by eliminating requirements to be ≤ 50% RTP within a specified time of exceeding the LCO and substituting compensatory measures in Condition A, which if not met, result in a reduction in power.

The purpose of the CTS actions is to lower reactor power to less than 50% when QPTR is not within its limit and cannot be restored to within its limit within a reasonable period. In addition, the Power Range Neutron Flux - High Trip setpoints are reduced to ≤ 55% to ensure that reactor power is not inadvertently increased without QPTR within its limit. This action is taken because with QPTR not within limit, the core power distribution is not within the analyzed assumptions and critical core parameters, such as $F_Q(Z)$ and $F^N_{\Delta H}$ may not be within their limits. A QPTR not within limit may not be an unacceptable condition if the critical core parameters, such as $F_Q(Z)$ and $F^N_{\Delta H}$, are within their limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features or restore out of limit parameters. The Required Actions are consistent with safe operation under the specified Condition, considering the status of the redundant indications, the capacity and capability of remaining features, a reasonable time for repairs or restoration of required features, and the low probability of a DBA occurring during the repair period. The ITS requires measurement of $F_Q(Z)$ and $F^N_{\Delta H}$ within 24 hours and every 7 days thereafter to verify that those parameters are within limit. In addition, the ITS requires the safety analyses to be reevaluated to ensure that the results remain valid. Assuming that these actions are successful, the ITS allows indefinite operation with QPTR out of its limit and allows the excore nuclear detectors to be normalized to eliminate the indicated QPTR. This ensures that the core is operated within the safety analysis. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.5 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS Surveillance 4.2.4.1 states that QPTR shall be determined to be within the limit by calculating the ratio at least once per 7 days. ITS SR 3.2.4.1, Note 2, states that SR 3.2.4.2, which requires verification of QPTR using the movable incore detectors, may be performed in lieu of SR 3.2.4.1. This changes the CTS by allowing the movable incore detectors to be used to determine QPTR instead of the excore detectors.

The purpose of CTS Surveillance 4.2.4.1 is to periodically verify that QPTR is within limit. This change is acceptable because it has been determined that the relaxed

Surveillance Requirement acceptance criteria are sufficient for verification that the parameters meet the LCO. The movable incore detector system provides a more accurate indication of QPTR than the excore detectors. In fact, the movable incore detector system is used to calibrate the excore detectors. Therefore, allowing the use of the movable incore detector system or the excore detectors is appropriate. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

SECTION 3.2 - POWER DISTRIBUTION LIMITS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

GENERIC NSHCs

10 CFR 50.92 EVALUATION FOR ADMINISTRATIVE CHANGES

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications with no change in intent. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording that does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1431. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR MORE RESTRICTIVE CHANGES

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements that currently do not exist.

These changes include additional commitments that decrease allowed outage times, increase the frequency of surveillances, impose additional surveillances, increase the scope of specifications to include additional plant equipment, increase the applicability of specifications, or provide additional actions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more restrictive requirements either has no effect on or increases the margin of plant safety. As provided in the discussion of change, each change in this category is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR RELOCATED SPECIFICATIONS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relocating existing Technical Specification LCOs to licensee controlled documents.

The the Company has evaluated the current Technical Specifications using the criteria set forth in 10 CFR 50.36. Specifications identified by this evaluation that did not meet the retention requirements specified in the regulation are not included in the Improved Technical Specifications (ITS) submittal. These specifications have been relocated from the current Technical Specifications to the Technical Requirements Manual.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria of 10 CFR 50.36 (c)(2)(ii) for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the North Anna Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to the Technical Requirements Manual, which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR.50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change will not reduce a margin of safety because it has no significant effect on any safety analyses assumptions, as indicated by the fact that the requirements do not meet the 10 CFR 50.36 criteria for retention. In addition, the relocated requirements are moved without change and any future changes to these requirements will be evaluated per 10 CFR 50.59.

NRC prior review and approval of changes to these relocated requirements, in accordance with 10 CFR 50.92, will no longer be required. This review and approval does not provide a specific margin of safety which can be evaluated. However, since the proposed change is consistent with the Westinghouse Standard Technical Specifications, NUREG-1431 issued by the NRC, revising the Technical Specifications to reflect the approved level of detail gives assurance that this relocation does not result in a significant reduction in the margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES - REMOVED DETAIL

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve moving details out of the Technical Specifications and into the Technical Specifications Bases, the UFSAR, the TRM or other documents under regulatory control such as the Quality Assurance Program Topical Report. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1431 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates certain details from the Technical Specifications to other documents under regulatory control. The Bases, UFSAR, and Technical Requirement Manual will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the Technical Specifications. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e). Other documents are subject to controls imposed by Technical Specifications or regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not reduce a margin of safety because it has no effect on any safety analysis assumptions. In addition, the details to be moved from the Technical Specifications to other documents are not being changed. Since any future changes to these details will be evaluated under the applicable regulatory change control mechanism,

no significant reduction in a margin of safety will be allowed. A significant reduction in the margin of safety is not associated with the elimination of the 10 CFR 50.92 requirement for NRC review and approval of future changes to the relocated details. The proposed change is consistent with the Westinghouse Standard Technical Specifications, NUREG-1431, issued by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail, which indicates that there is no significant reduction in the margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 1 RELAXATION OF LCO REQUIREMENTS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by the elimination of specific items from the LCO or Tables referenced in the LCO, or the addition of exceptions to the LCO.

These changes reflect the ISTS approach to provide LCO requirements that specify the protective conditions that are required to meet safety analysis assumptions for required features. These conditions replace the lists of specific devices used in the CTS to describe the requirements needed to meet the safety analysis assumptions. The ITS also includes LCO Notes which allow exceptions to the LCO for the performance of testing or other operational needs. The ITS provides the protection required by the safety analysis and provides flexibility for meeting the conditions without adversely affecting operations since equivalent features are required to be OPERABLE. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides less restrictive LCO requirements for operation of the facility. These less restrictive LCO requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the current safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the change is consistent with the assumptions in the current safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The imposition of less restrictive LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the current safety analyses and licensing basis requirements are maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 2
RELAXATION OF APPLICABILITY

The North Anna Nuclear Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the applicability of current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by reducing the conditions under which the LCO requirements must be met.

Reactor operating conditions are used in CTS to define when the LCO features are required to be OPERABLE. CTS Applicabilities can be specific defined terms of reactor conditions or more general such as, "all MODES" or "any operating MODE." Generalized applicability conditions are not contained in ITS, therefore the ITS eliminates CTS requirements such as "all MODES" or "any operating MODE," replacing them with ITS defined MODES or applicable conditions that are consistent with the application of the plant safety analysis assumptions for operability of the required features.

CTS requirements may also be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the TS may be satisfied by exiting the applicability which takes the plant out of the conditions that require the safety system to be OPERABLE.

This change provides the protection required by the safety analysis and provides flexibility for meeting limits by restricting the application of the limits to the conditions assumed in the safety analyses. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. The change is generally made to conform with NUREG-1431 and has been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes the conditions under which the LCO requirements for operation of the facility must be met. These less restrictive applicability requirements for the LCOs do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure that process variables, structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. Therefore, this change

does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the requirements are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The relaxed applicability of LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the LCO requirements are applied in the MODES and specified conditions assumed in the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 3 RELAXATION OF COMPLETION TIME

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the Completion Times for Required Actions in the current Technical Specifications (CTS).

Upon discovery of a failure to meet an LCO, the ITS specifies times for completing Required Actions of the associated TS Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken within specified Completion Times (referred to as Allowed Outage Times (AOTs) in the CTS). These times define limits during which operation in a degraded condition is permitted. Adopting Completion Times from the ITS is acceptable because the Completion Times take into account the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. In addition, the ITS provides consistent Completion Times for similar conditions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes the Completion Time for a Required Action. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Completion Time does not significantly increase the probability of any accident previously evaluated. The consequences of an analyzed accident during the relaxed Completion Time are the same as the consequences during the existing AOT. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the method governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The relaxed Completion Time for a Required Action does not involve a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the allowed Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 4
RELAXATION OF REQUIRED ACTION

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the Required Actions in the current Technical Specifications (CTS).

Upon discovery of a failure to meet an LCO, the ITS specifies Required Actions to complete for the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken in response to the degraded conditions. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from the ISTS is acceptable because the Required Actions take into account the operability status of redundant systems of required features, the capacity and capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes Required Actions. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Required Actions do not significantly increase the probability of any accident previously evaluated. The Required Actions in the ITS have been developed to provide appropriate remedial actions to be taken in response to the degraded condition considering the operability status of the redundant systems of required features, and the capacity and capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The relaxed Required Actions do not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to minimize the risk of continued operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 5
DELETION OF SURVEILLANCE REQUIREMENT

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve deletion of Surveillance Requirements in the current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. The ITS eliminates unnecessary CTS Surveillance Requirements that do not contribute to verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change deletes Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining Surveillance Requirements are consistent with industry practice and are considered to be sufficient to prevent the removal of the subject Surveillances from creating a new or different type of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The deleted Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the deleted Surveillance Requirements are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 6 RELAXATION OF SURVEILLANCE REQUIREMENT ACCEPTANCE CRITERIA

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the relaxation of Surveillance Requirements acceptance criteria in the current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. The ITS eliminates or relaxes the Surveillance Requirement acceptance criteria that do not contribute to verification that the equipment used to meet the LCO can perform its required functions. For example, the ITS allows some Surveillance Requirements to verify Operability under actual or test conditions. Adopting the ITS allowance for "actual" conditions is acceptable because required features cannot distinguish between an "actual" signal or a "test" signal. Also included are changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements which, when combined, include Operability verification of all TS required components for the features specified in the CTS. Adopting this format preference in the ISTS is acceptable because Surveillance Requirements that remain include testing of all previous features required to be verified OPERABLE. Changes which provide exceptions to Surveillance Requirements to provide for variations which do not affect the results of the test are also included in this category. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes the acceptance criteria of Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The relaxed acceptance criteria for Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxed Surveillance Requirement acceptance criteria have been evaluated to ensure that they are sufficient to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner that gives confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 7 RELAXATION OF SURVEILLANCE FREQUENCY

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the relaxation of Surveillance Frequencies in the current Technical Specifications (CTS).

CTS and ITS Surveillance Frequencies specify time interval requirements for performing surveillance testing. Increasing the time interval between Surveillance tests in the ITS results in decreased equipment unavailability due to testing which also increases equipment availability. In general, the ITS contain test frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the ITS is acceptable based on similar design, like-component testing for the system application and the availability of other Technical Specification requirements which provide regular checks to ensure limits are met. Relaxation of Surveillance Frequency can also include the addition of Surveillance Notes which allow testing to be delayed until appropriate unit conditions for the test are established, or exempt testing in certain MODES or specified conditions in which the testing can not be performed.

Reduced testing can result in a safety enhancement because the unavailability due to testing is reduced and; in turn, reliability of the affected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the frequency is acceptable from a reliability standpoint. Surveillance Frequency changes to incorporate alternate train testing have been shown to be acceptable where other qualitative or quantitative test requirements are required which are established predictors of system performance. Surveillance Frequency extensions can be based on NRC-approved topical reports. The NRC staff has accepted topical report analyses that bound the plant-specific design and component reliability assumptions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relaxes Surveillance Frequencies. The relaxed Surveillance Frequencies have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment which could initiate an accident previously evaluated will continue to operate as expected and the probability of the initiation of any accident previously evaluated will not be significantly increased. The equipment being

tested is still required to be Operable and capable of performing any accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The relaxed Surveillance Frequencies do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 8 DELETION OF REPORTING REQUIREMENTS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the deletion of requirements in the current Technical Specifications (CTS) to send reports to the NRC.

The CTS includes requirements to submit reports to the NRC under certain circumstances. However, the ITS eliminates these requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The ITS changes to reporting requirements are acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change has no effect on the safe operation of the plant. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change deletes reporting requirements. Sending reports to the NRC is not an initiator to any accident previously evaluated. Consequently, the probability of any accident previously evaluated is not significantly increased. Sending reports to the NRC has no effect on the ability of equipment to mitigate an accident previously evaluated. As a result, the consequences of any accident previously evaluated is not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The deletion of reporting requirements does not result in a significant reduction in the margin of safety. The ITS eliminates the requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The change to reporting requirements does not affect the margin of safety because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT
SECTION 3.2 - POWER DISTRIBUTION LIMITS

This proposed Technical Specification change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed Technical Specification change meets the criteria for categorical exclusion.

10 CFR 51.22(c)(9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements,

(i) proposed change involves No Significant Hazards Considerations (refer to the Determination of No Significant Hazards Considerations section of this Technical Specification Change Request);

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths; and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental affect statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed change of this request.

SECTION 3.2 - POWER DISTRIBUTION LIMITS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

SPECIFIC NSHCs

There are no specific NSHC discussions for this Section.