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Nuclear Design Methodology Report For Core Operating Limits of Westinghouse Reactors Non-Proprietary

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Revision History

Revision	Description
DPC-NE-2011NP, Original Issue	Originally submitted to the NRC for approval in April 1988. An additional submittal was made to the NRC supplying reponses to a request for additional information.
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DPC-NE-2011NP, Revision 1	Submitted to the NRC for approval in August 2001. This revsion updates the report for completeness (1) to indicate the use of methods approved by the NRC subsequent to the implementation of the original issue and (2) to expand the description of Fq monitoring to include the centerline fuel melt criterion. This revison also reflects changes in the descriptions of the xenon transient conditions. Finally, various editorial changes are made, including updating the Table of Contents and revising the page numbering format. Changes associated with this revision are denoted by revision bars, except for format changes.
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1. INTRODUCTION

1.1. Purpose

This report describes the methodology for performing a maneuvering analysis for four-loop, 193 fuel assembly Westinghouse reactors, such as McGuire and Catawba Nuclear Stations. Duke Power Company has developed this methodology as an alternative to the existing Relaxed Axial Offset Control (RAOC) Methodology (1). This maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of Duke Power Company's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower ΔT (OPΔT) and overtemperature ΔT (OTΔT) reactor protection system (RPS) trip functions.

Specifically, these limits are the axial flux difference (AFD) - power level operating space, the rod insertion limits and the f(ΔI) function of either the OPΔT or the OTΔT trip functions of the RPS.

These limits are monitored via Technical Specifications.

1.2. Summary of the Methods

The operating limits define the AFD - power level space and rod insertion limits which provide assurance that the peak local power in the core is not greater than that assumed in the analysis of design basis accidents or transients (loss of coolant accident (LOCA) or loss of flow accident(LOFA)). Operating the reactor within the allowed AFD - power level window and rod insertion limits satisfies the power peaking assumptions of the LOCA and LOFA analyses.

The RPS limits, among other functions, provide protection against fuel failure due to fuel melting (CFM) or departure from nucleate boiling (DNB) during anticipated transients. The relevant limits are set such that the RPS will trip the reactor before fuel damage occurs.

The maneuvering analysis uses a three dimensional nodal reactor model to calculate a set of power distributions at several points in core life. These power distributions are based on a set of abnormal xenon distributions to insure predicted power distributions are conservative with respect to those expected to occur. In the EPRI-NODE-P (NODE) model, the three dimensional nodal power distribution is augmented by pin to assembly factors for the maximum pin power in each assembly. These pin to assembly factors are derived from a two dimensional fine mesh (pin by pin) model of the core. In the SIMULATE-3P (SIMULATE) model, the three dimensional local peak pin power distributions are explicitly calculated. Appropriate uncertainty factors are applied to the calculated power distributions which are then evaluated against the various thermal limits. The operating limits and the f(ΔI) function of either the OPΔT or the OTΔT RPS trip functions are then set to exclude the power distributions that exceed the respective thermal limits. Figures 1A and 1B show representative flow charts of the data as it goes through a NODE and a SIMULATE based maneuvering analysis.

1.3. Applicability of the Method

The maneuvering analysis presented in this report applies to Westinghouse four loop, 193 assembly reactors. This method is intended to be used to set or validate the AFD - power level operating limits, the control rod insertion limits, and the RPS trip limits.

A system of computer programs is used to implement this method. A description of the computer programs currently in use is contained in Appendix A. This list includes both the major design codes approved by the NRC (4, 15) and minor codes that are used for post-processing data.

1.4. Definition of Terms

AFD

Axial Flux Difference is the percent power in the top of the core minus the percent power in the bottom of the core.

Radial Local Factors

A Radial Local Factor (RADLOC) is the peak rod power in an assembly divided by the average rod power in the same assembly.

F_Q

F_Q is the local heat flux on a fuel rod surface divided by the core average fuel rod heat flux.

$F_{\Delta H}$

$F_{\Delta H}$ is the integral of linear power along a particular fuel rod divided by the average integral of all of the fuel rods.

QPTR

Quadrant Power Tilt Ratio is the normalized radial power distribution in each quadrant of the core as measured by excore nuclear detectors.

MATP

Maximum Allowed Total Peak values derived from core thermal-hydraulic analysis.

MARP

Maximum Allowed Radial Peak values derived from MATP values by dividing the MATP by the axial peak.

Figure 1A
Flow of Data Through a Maneuvering Analysis - EPRI-NODE-P

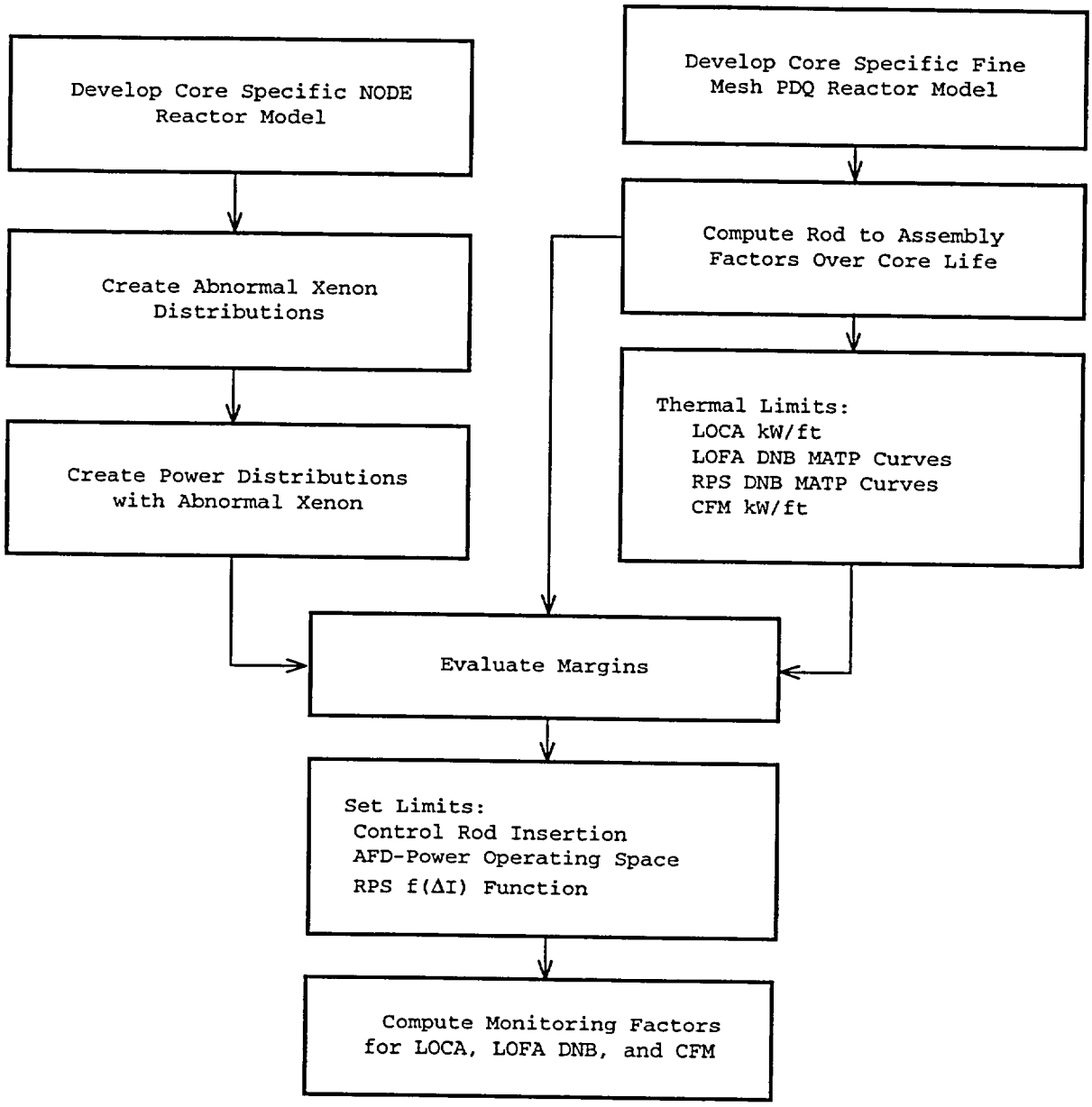
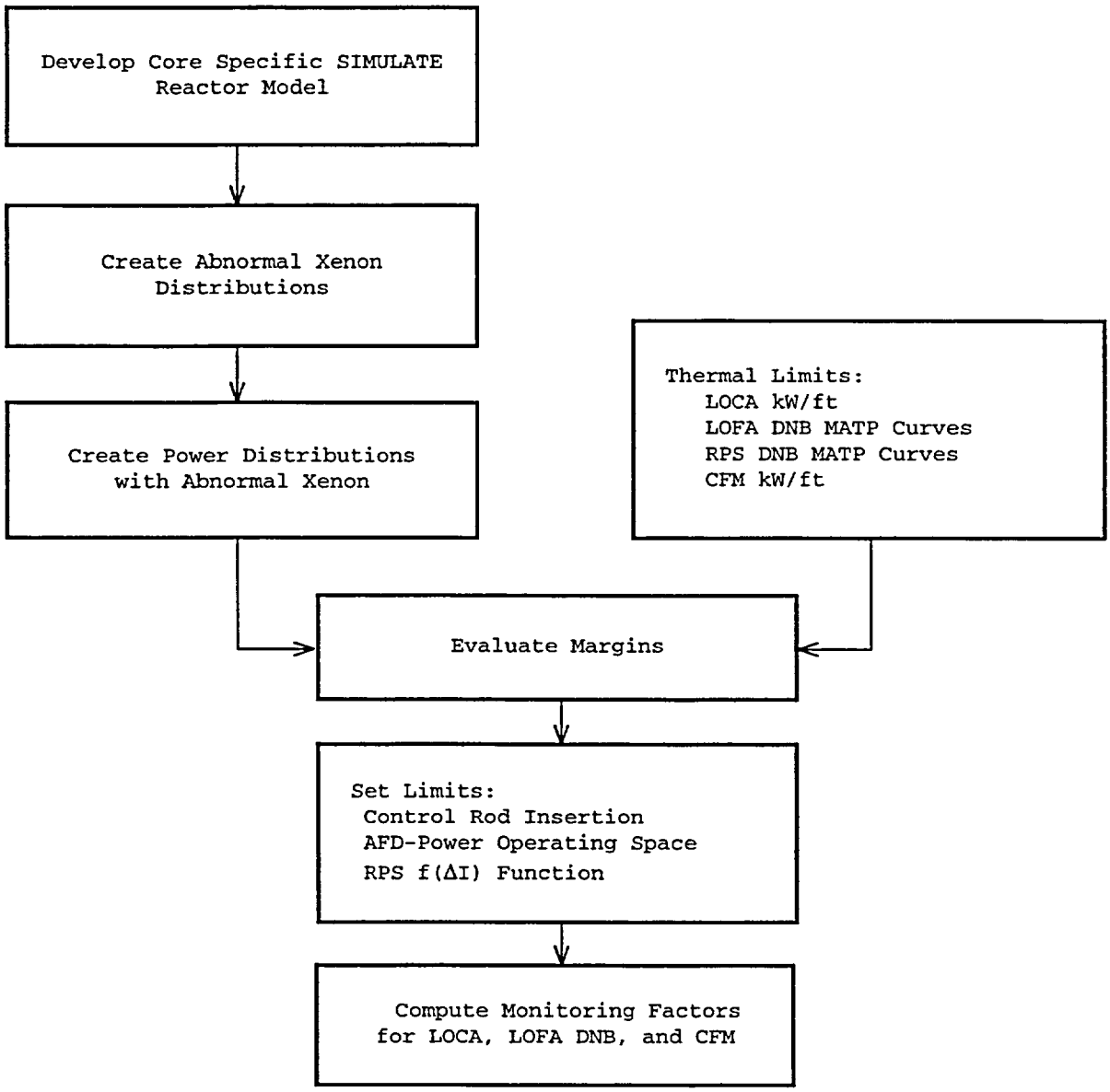


Figure 1B
Flow of Data Through a Maneuvering Analysis - SIMULATE-3P



2. GENERATION OF POWER DISTRIBUTIONS

2.1. Description of the Models Used

The three dimensional nodal power and xenon distributions are generated by a DPC version of EPRI-NODE-P (NODE). NODE has an explicit xenon and iodine model that allows power and time dependent xenon transients. NODE has a closed channel thermal hydraulic feedback model to generate fuel and moderator temperature distributions that are used in the neutronics model. The neutronics model accounts for fuel and moderator temperature, coolant flow, soluble boron concentration, lumped burnable absorbers, control rods, fuel burnup, and xenon and iodine distributions. The NODE model was approved by the NRC for use in reload design in Reference 5.

The radial local factors are extracted from a quarter core, one pin per mesh PDQ07 model of the core. PDQ calculations are run in two dimensions (X-Y) with a two dimensional thermal hydraulic feedback model. The PDQ model was approved for use in reload core design in Reference 5.

SIMULATE-3P (SIMULATE) can be used to generate three dimensional local peak pin power distributions. The SIMULATE model was approved for use in reload core design analyses in Reference 15.

2.2. Times in Core Life

The maneuvering analysis is typically performed at three times in core life:

[

]

2.3. Generation of Abnormal Xenon Distributions

The abnormal xenon distributions are generated with a set of limiting xenon transients at each point in core life that is to be analyzed. [

]

Table 1 shows the initial and transient conditions of the reactor for each of the transients. [

]

To add to the conservatism, these transients are modeled conservatively in several respects: [

] Because of these factors, the xenon transients in the reactor model will be more severe than could be reasonably expected to occur.

Each of the xenon transients start with xenon in equilibrium with the core at the initial conditions. The initial conditions are different for each transient. [

]

The control rod positions for the xenon transients were chosen to be at or near the expected rod insertion limits. The final control rod insertion limits may be different from the positions used in the xenon transients and the analysis will still be valid. This is because the xenon transients are so severe that the maneuvering analysis results are not sensitive to the control rod motions that drive the xenon transients.

The xenon transients proceed until [ ] Depending on the transient power level, this usually takes about [ ] hours. Figures 2 through 5 show graphs of AFD, xenon offset, xenon concentration, and soluble boron concentration plotted against time for a typical set of beginning of cycle xenon transients.

2.4. Generation of Power Distributions

Using the abnormal xenon distributions from the xenon transients, three dimensional power distributions are generated so that the operating and the RPS limits can be determined. As shown on Table 2, power distributions are generated with

] The operating limits are pre-conditions that would prevent exceeding the peak local power in the core assumed in the loss of coolant accident (LOCA) analysis or the loss of flow accident (LOFA, or a primary coolant pump trip) analysis. Because this is the normal operating mode of the reactor, control rod motion will be constrained by the power dependent rod insertion limits. [

] Power distributions for the operating limits are generated with these abnormal xenon distributions with the reactor at nominal conditions.

The RPS limits protect the fuel against damage from DNB or fuel melting even if the reactor should go through any one of several anticipated transients: [

]

The limit of the control rod motion for [

]

During an [

]

The abnormal xenon distributions from the xenon transients are chosen so that [

] Table 2 shows the reactor conditions and range of control rod positions. Criticality in the reactor model is maintained by instantaneous changes in soluble boron concentrations.

2.5. Generation of Radial Local Factors

The radial local factor is the ratio of the maximum rod power in an assembly to the average rod power of the assembly. Radial local factors are assembly and burnup dependent.

In the NODE methodology, the radial local factors are extracted from a core specific fine mesh PDQ model that has been depleted over the life of the cycle. The assembly average burnup, used as the independent variable to interpolate the radial local factors, is also extracted from the PDQ model. The PDQ model has two neutron energy groups and one spatial mesh point per fuel pin. Cross sections are taken from the EPRI-CELL (6) system and the CASMO (7) system. The PDQ model is described more fully in Reference 4.

SIMULATE (15) directly calculates local peak pin power distributions.

Table 1

Typical Reactor Conditions During Xenon Transients

	Initial Conditions		Transient Conditions	
Transient Name	% Power	Control Rods	% Power	Control Rods

Table 2

Typical Power Levels and Control Rod Bank Positions
for Generating Power Distributions

Figure 2
Sample Xenon Transient at Beginning of Life
AFD vs Transient Time

Figure 3
Sample Xenon Transient at Beginning of Life
Xenon Concentration vs Transient Time

Figure 4
Sample Xenon Transient at Beginning of Life
Xenon Offset vs Transient Time

Figure 5
Sample Xenon Transient at Beginning of Life
Soluble Boron Concentration vs Transient Time

3. UNCERTAINTY FACTORS

3.1. Power Distribution

The power peaks calculated in the Maneuvering Analysis are adjusted to account for calculation uncertainty and other applicable factors that may affect the power peaking in the core.

References 4, 5, and 15 present calculation peaking uncertainties based on the benchmarking analysis of measured to predicted power distribution. The peaking uncertainty factor is calculated as described below.

$$\text{Peaking Uncertainty Factor} = 1+\text{BIAS}+\sqrt{(UC^2+Ux1^2+Ux2^2+...)}$$

Where:

Peaking Uncertainty Factor - Defined as UCT, UCR, UCA in this report

UC - Calculation Uncertainty

For the Pin Total Peak (F_Q), UCT: $UC^2 = UT^2 + URL^2$

For the Pin Radial Peak ($F_{\Delta H}$), UCR: $UC^2 = UR^2 + URL^2$

For the Assembly Axial Peak (F_Z), UCA: $UC^2 = UA^2$

UT - Total Peaking Uncertainty

URL - Assembly Radial Local (or Pin) Power Peaking Uncertainty

UR - Assembly Radial Power Peaking Uncertainty

UA - Assembly Axial Power Peaking Uncertainty

Uxi - Additional Uncertainties, e.g. engineering HCF, rod bow, etc.

BIAS - Calculation Bias

When additional, independent, peaking augmentation factors (shown as Uxi above) such as the engineering hot channel factor and/or rod bow factor are required, the corresponding uncertainty values are statistically combined with the pin and assembly power calculation uncertainty values to obtain the total uncertainty factor. The application of specific parameters is discussed in Section 4.

3.2. Quadrant Tilt

The excore detector system is used to monitor gross changes in the core power distribution. The primary purpose of the excore detectors with respect to quadrant power tilts is to detect changes in tilt from the previous calibration. Since the Technical Specifications (2, 3) allow reactor operations with excore quadrant power tilts up to 2%, the relationship between excore quadrant power tilt and a penalty to apply to the thermal limits calculations had to be determined.

This relationship was determined by evaluating various tilt causing mechanisms for several reactor cores. This analysis was performed with full core NODE models. The results showed that a [ ] power peaking penalty is required to account for the allowed 2% excore quadrant power tilt. This penalty will be applied as TILT to the LOCA, DNB and centerline fuel melt margin calculations in Section 4.

4. LCO AND RPS LIMITS

4.1. General Methodology

The power distributions are divided into two categories for the thermal limits calculations. The operating limits use power distributions that were calculated with nominal inlet temperature, with control rod positions that bound expected insertion limits, and with power less than or equal to 100% power. Control rod positions will bound insertion limits in order to set the insertion limits. The RPS limits use power distributions with the power level up to and including 118% power, no administrative restriction on the control rod insertions and either nominal or low inlet temperature.

The margin to the various limits is calculated in the following fashion:

MARGIN % = (ALLOWED PEAK - CALCULATED PEAK)*100 / ALLOWED PEAK

The calculated peak is obtained directly from SIMULATE or is a synthesis of the three dimensional nodal power distribution from NODE and the radial local factors from the fine mesh two dimensional PDQ calculations. Depending on the limit type, this equation may be in terms of a peaking factor or a linear heat rate. Either the calculated peak or the allowed peak would contain sufficient factors to account for the various uncertainties and tolerances. AFD and control rod insertion limits for each limit type are set to exclude all power distributions with negative margins of the same limit type.

4.2. LOCA Margin Calculations

Since the LOCA limits are used to define the operating limits of the core, the operating limits power distributions, as described in Section 2.4, are used in this calculation. The LOCA margin is calculated for each node in the core, but only the most limiting value is used in the determination of the AFD - power level limits. The equations below show how the LOCA margin, LOCAM, is calculated.

LOCAM = Min {(LOCAMX(z) - LHR(x,y,z)) * 100 / LOCAMX(z)}

Where:

LOCAMX(z)	= Axially dependent maximum allowable linear heat rate in kw/ft.
LHR(x,y,z)[1]	= NP(x,y,z) * FP * AVGLHR * RADLOC(x,y,e) * UCT * TILT * RPF * AMF
NP(x,y,z)[1]	= Nodal power from the power distribution calculation.
FP	= Fraction of core power level, including power level uncertainty.
AVGLHR	= Total core power divided by the total length of fuel rods in the core, kw/ft, accounting for fuel densification and thermal expansion.
RADLOC(x,y,e)	= Burnup (e) dependent maximum rod assembly power factor.
UCT	= Uncertainty factor on the pin total peak, including engineering hot channel factor and rod bow if not included in the LOCA analysis (see Section 3.1).
TILT	= Factor to account for a peaking increase due to an allowed quadrant tilt (see Section 3.2).
RPF	= Factor to account for the power deposited in the fuel rod.
AMF	= Additional Margin Factor, optionally used to incorporate additional design margin.

The values for LOCAMX(z) are derived from the Technical Specification limits on F_Q. Typical limiting values are shown in Figure 6.

The uncertainty on power level and the factor to account for power deposited in the fuel will be used only if these factors were not accounted for in the limits on F_Q.

[1] For SIMULATE, LHR does not include the RADLOC factor, since NP is the SIMULATE three dimensional local peak pin value.

4.3. LOFA DNB Margin Calculations

The LOFA DNB limits are also used to define the operating limits, so the operating limits power distributions, as described in Section 2.4, are used in this calculation. The DNB margin calculation is based on a set of Maximum Allowed Total Peak (MATP) curves that are calculated with a NRC approved thermal-hydraulic method (e.g., Reference 14). The MATP curves are determined for several power levels (e.g., 100, 75 and 50% power). The input power distributions are selected to match the power level of each set of MATP curves. Sample MATP curves for LOFA DNB are shown in Figure 10. The DNB margin is computed for each assembly in the core, but only the minimum margin for each power distribution is used in the determination of the AFD - power limits. DNB margin, DNBM, is calculated as:

$$DNBM = \mathrm{Min}\left\{\frac{MARP(x,y)-RPP(x,y)}{MARP(x,y)}\right\}*100$$

Where:

MARP(x,y)	= MATP(z,AP(x,y))/(AP(x,y)*UCA)
AP(x,y)	= Axial peak in an assembly, on an assembly normalized basis.
UCA	= Assembly axial peak uncertainty factor (see Section 3.1).
MATP(z)	= Maximum allowed total peak, at the axial plane of the axial peak.
RPP(x,y)[2]	= RNP(x,y) * RADLOC(x,y,e) * AMF * TILT * UCR
RNP(x,y)[2]	= Normalized assembly power from the power distribution calculation.
RADLOC(x,y,e)	= Burnup (e) dependent maximum rod to assembly power factor.
UCR	= Uncertainty factor on the pin radial peak, including engineering hot channel factor and rod bow if not included in the DNB analysis (see Section 3.1).
AMF	= Additional margin Factor, optionally used to incorporate additional design margin.

[2] For SIMULATE, RPP does not include the RADLOC factor, since RNP is the SIMULATE two dimensional peak pin value.

TILT = Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).

The axial uncertainty factor will be included only if it has not been accounted for in the MATP curves.

4.4. RPS DNB Margin Calculations

The rest of the DNB margin calculations are used to validate the RPS limits, so the operating limits restrictions on power distributions are not applied. The methodology for computing RPS DNB margin is the same as in Section 4.3, however the MATP curves are different. Table 3 lists the conditions at which the RPS MATP curves were generated and the conditions of the power distributions that will be used for each set of MATP curves.

4.5. Centerline Fuel Melt Margin Calculations

The centerline fuel melt limit is also used to validate the RPS limits, so the operating limits restrictions on power distributions are not applied in the calculation. Since there usually is a positive margin for centerline fuel melting, only the power distributions at 118% power are used for the centerline fuel melt margin calculations. A positive margin at 118% power will preclude negative margins at lower power levels. If the 118% power level results show negative margins, lower power levels will be analyzed to fully define the AFD - power level limit. The equations below show how the margin for centerline fuel melt is calculated. Note that the linear heat rate is calculated similarly to the LOCA margin calculation. Each node in the core model is analyzed, but only the minimum margin for a power distribution is used to determine the AFD - power level limits.

$$CFMM = Min\left\{\frac{MAXLHR - LHR(x,y,z)}{MAXLHR}\right\} * 100$$

Where:

MAXLHR	= Maximum allowable linear heat rate in kw/ft.
LHR(x,y,z)[3]	= NP(x,y,z) * FP * AVGLHR * RADLOC(x,y,e) * TILT * RPF * UCT * AMF
NP(x,y,z)[3]	= Nodal power from the power distribution calculation.
FP	= Fraction of core power level, including power level uncertainty.
AVGLHR	= Total core power divided by the total length of fuel rods in the core, kw/ft, accounting for fuel densification and thermal expansion.
RADLOC(x,y,e)	= Burnup(e) dependent maximum rod to assembly power factor.
UCT	= Uncertainty factor on the pin total peak, including engineering hot channel factor and rod bow if not included in the fuel mechanical analysis (see Section 3.1).
TILT	= Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).
RPF	= Factor to account for the power generated in the fuel rod.
AMF	= Additional Margin Factor, optionally used to incorporate additional design margin.

The uncertainty on power level and the factor to account for power deposited in the fuel will be used only if these factors were not accounted for in the limiting heat generation rate.

[3] For SIMULATE, LHR does not include the RADLOC factor, since NP is the SIMULATE three dimensional local peak pin value.

4.6. Determining the AFD - Power Level Limits

The individual values of margin for each power distribution and margin calculation are collected into a database. For each power level and margin calculation, the margin data is plotted against AFD. The data points are connected by drawing lines between points with an equal independent parameter. Control rod position is usually chosen as this independent parameter, which means that different points along these lines represent different xenon time steps. The limit is set to preclude operation with negative peaking margin. At lower power levels, core conditions may not produce an AFD at the desired AFD limit. For this case, the AFD limit from the upper power level is extrapolated to the lower power level and the core conditions are verified to yield non-negative margins. Figures 7 and 8 shows an example plot of LOCA and LOFA DNB margin plotted against AFD, connected by equal rod position lines.

The operating AFD limits are determined by selecting the limiting of either the LOCA margin results or the LOFA DNB margin results at the various power levels analyzed. The AFD limits may be interpolated between rod position if the rod position chosen for the rod insertion limit was not explicitly modeled when the power distributions were generated. The bounding AFD envelope is adjusted to account for measurement system (two segment power-range excore nuclear detectors) uncertainties. The uncertainties account for the excore detector calibration error and drift between calibrations.

The DNB margin calculations performed for the RPS OTΔT AFD Trip penalty, f(ΔI), provide AFD limits [

] The power - AFD penalty is determined by selecting the limiting breakpoints and slopes defined by the [

] The uncertainty associated with the f(ΔI) function is combined with the uncertainties of the other OTΔT function input parameters in determining the adjusted K_1 constant in the setpoint equation (References 2, 3), or the f(ΔI) function is adjusted to account for the AFD uncertainties.

The centerline fuel melt protection criterion is associated with the OPΔT Trip f(ΔI) penalty function. Since the OPΔT f(ΔI) function is usually zero, the check performed at 118% power is adequate to verify that the penalty is not required. Should the centerline fuel melt margin calculations result in an AFD limit at 118% power, lower power levels would be analyzed in order to define the power - AFD penalty. The penalty could then be incorporated into the OPΔT trip function or the required protection could be provided by the OTΔT function.

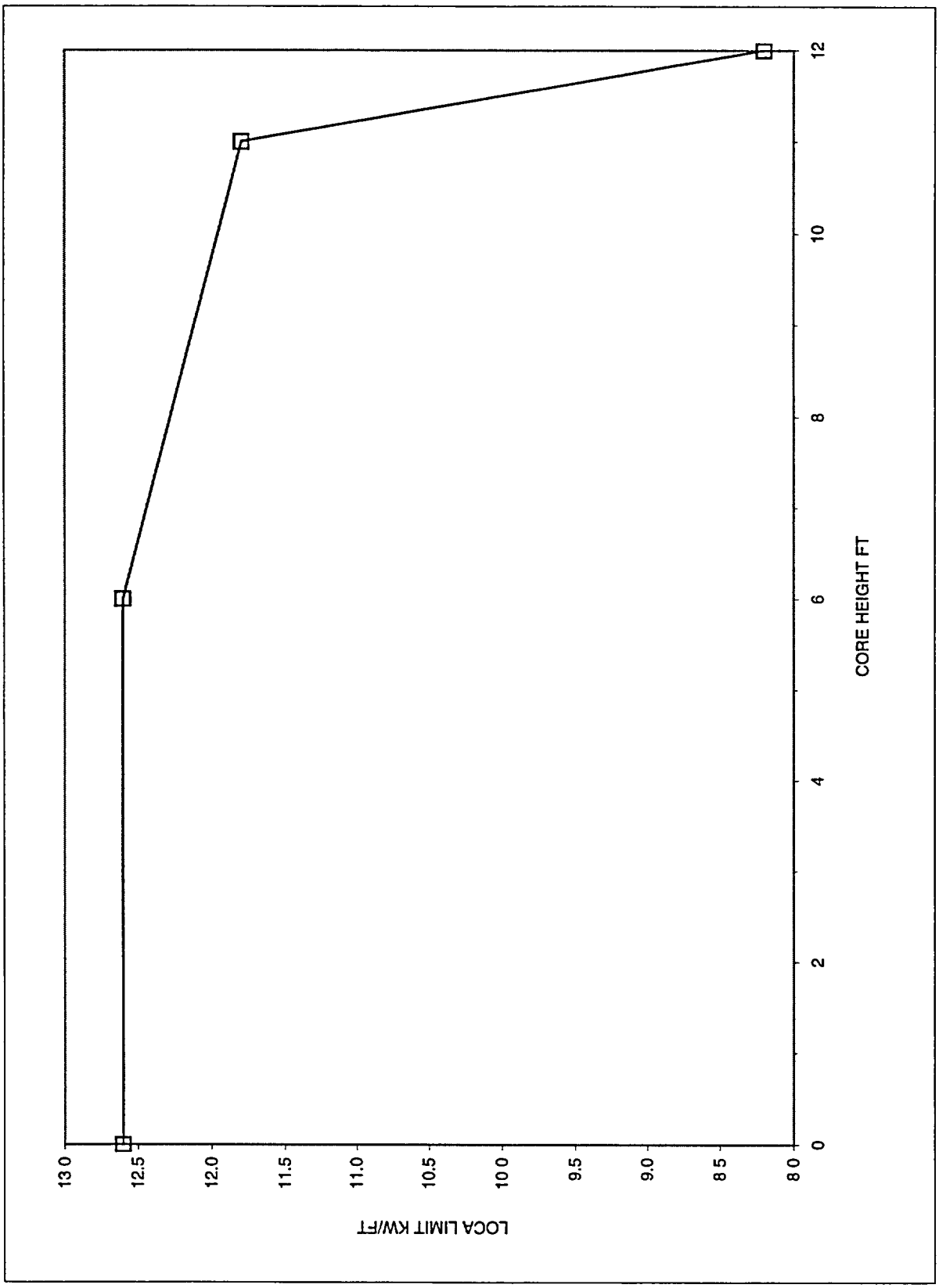
4.7. Control Rod Insertion Limits

The rod insertion limits are assumed when the operational AFD - power level limits are set. However, further iteration on the limits may be necessary depending on the results of the shutdown margin and ejected rod analyses. Adjustments are made to the rod insertion limits and AFD - power level limits as necessary.

Table 3

Typical RPS MATP Curve Conditions
and Conditions of the Power Distributions
used for each set of MATP Curves

Figure 6
Typical LOCA Linear Heat Rate Limits vs Core Height



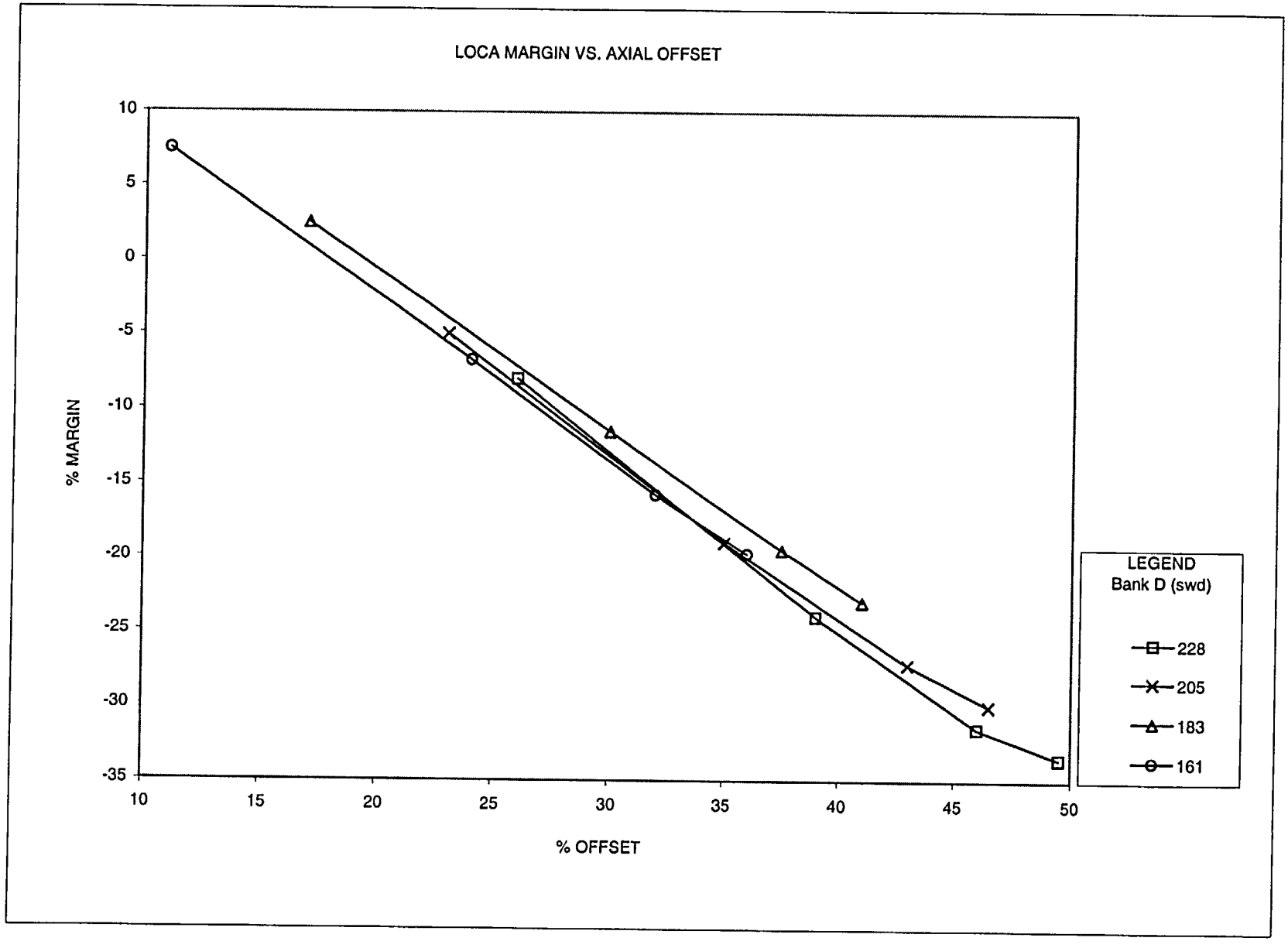


Figure 7
Sample LOCA Margin Plot

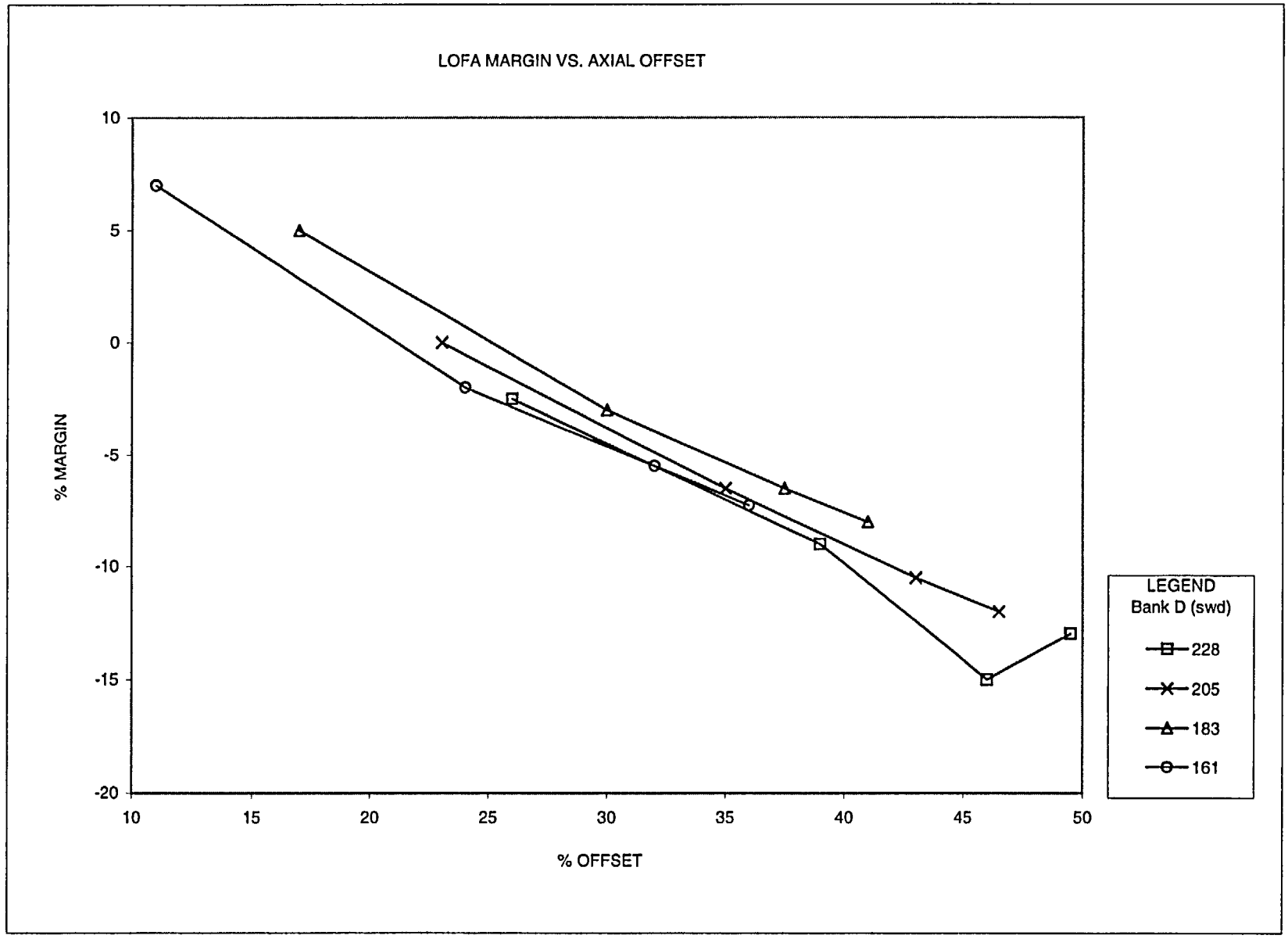


Figure 8
Sample LOFA Margin Plot

5. BASE LOAD LCO LIMITS

If the operational limits for a particular fuel cycle are too restrictive for normal operation, then a set of base load limits can be defined that may allow power operation at 100% power. Base load is defined as operating the reactor within a relatively narrow AFD band about a plant measured AFD target and within a limited power range. By limiting the allowed AFD - power level space, extra margin can be gained in the power distribution monitoring factors (see Section 6).

Base load limits and monitoring factors are computed the same as the operational limits, only the xenon transients will be re-defined so that they will be restricted to the base load operating band about a predicted AFD target. The power level at which the plant will be allowed to enter base load will be greater than or equal to the power level of the xenon transients.

6. POWER DISTRIBUTION SURVEILLANCE

The AFD - power level limits are set to preserve the power peaking assumptions in the LOCA analysis and to protect the fuel from damage during a LOFA when the power distribution is skewed in the axial direction. Similary, f(ΔI) limits are set to preclude RPS limits from being exceeded during Condition II transients. Because only steady state power distributions can be measured with reasonable accuracy, the limits on the measured power distribution are reduced by pre-calculated factors that account for perturbations from steady state conditions to applicable limits.

6.1. LOCA F_Q Surveillance Methodology

The Technical Specification (2, 3) LOCA F_Q limit that must be satisfied within the AFD - power level operating limits is:

$$F_Q^M(x,y,z) \leq \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q^M(x,y,z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

Where: P = relative thermal power.
$K(Z)$ = normalized F_Q as a function of core height (see Figure 9).
F_Q^{RTP} = the LOCA limit at rated thermal power (RTP).

This criterion is a Technical Specification (2, 3) limiting condition for operation (LCO).

Using definitions from Section 4.2, the reduced limits for the measured F_Q are specified as:

$$F_Q^M(x,y,z)*UMT*MT*TILT \leq [\qquad\qquad]$$

Where:
$F_Q^M(x,y,z)$ = The measured total peak in location x,y,z.

UMT = The measurement uncertainty factor on the total peak, provided in the Technical Specifications (2, 3).

MT = Manufacturing tolerance factor (or engineering hot channel factor), provided in the Technical Specifications (2, 3).

TILT = Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).

$F_Q^D(x,y,z)^4$ = $NP^D(x,y,z)$ * RADLOC(x,y,e) * UCT, design power distribution for F_Q.

$M_Q(x,y,z)$ = The LOCA margin remaining in location x,y,z in the calculated transient power distributions.

[ ]

6.2. CFM F_Q Surveillance Methodology

Using definitions from Section 4.5, the measured F_Q CFM surveillance limit is:

$$F_Q^M(x,y,z)*UMT*MT*TILT \leq [\qquad\qquad]$$

Where the parameters in the above equation are defined in Section 6.1, except:

[ ]

[ ]

6.3. LOFA DNB $F_{\Delta H}$ Surveillance Methodology

The Technical Specification (2, 3) $F_{\Delta H}$ limit that must be satisfied within AFD - power level operating limits is:

$$F^{M}_{\Delta H}(x,y) \leq MARP(x,y)*\left[1.0+\frac{1}{RRH}*(1.0-P)\right]$$

Where P is the relative thermal power. MARP(x,y) is the Maximum Allowed Radial Peak which is derived from the MATP curves (see Figure 10) by dividing the MATP by the axial peak term. This criterion is a Technical Specification (2, 3) LCO.

The limits for $F_{\Delta H}$ must be reduced for the same reason as the F_Q limits are reduced (see Section 6.1). Using definitions from Section 4.3, the reduced limit for monitoring $F_{\Delta H}$ is given in the following relationship:

$$F^{M}_{\Delta H}(x,y)*UMR*TILT \leq [\qquad\qquad]$$

Where:

$F^{M}_{\Delta H}(x,y)$ = Measured value of $F_{\Delta H}$

[ ]

UMR = Uncertainty factor on the measured radial peaks, provided in the Technical Specifications (2, 3).

TILT = Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).

6.4. Monitoring of Plant Measured Parameters

During power operations, the power distribution is continuously monitored by the ex-core nuclear instrumentation. The parameters of interest to power distribution monitoring are the core power level, the AFD and the quadrant power tilt. Limitations are imposed on these three parameters by the maneuvering analysis. The maneuvering analysis also imposes limits on control rod positions during power operations. The power distribution is also measured periodically by the in-core instrumentation system. The results of these measurements are used to verify that the core is behaving as predicted by the maneuvering analysis or to adjust the AFD - power level limits if it is not. The surveillance of these parameters is described below.

6.4.1. AFD - Power Level Limits

During normal operations, the combination of AFD and power level must be maintained within the operating limits that are provided by the maneuvering analysis. Example AFD - power level limits are shown in Figure 11. Since the operating limits are a Limiting Condition of Operation (instead of a Limiting Safety System Setting), the plant would be allowed to operate outside of the operating AFD - power level limits for short periods of time if necessary. This allowance is meant to be used to increase the plant availability during transient situations and is not meant to be used for normal operation.

If the power distribution is unusually limiting (because of severe power peaking, for example), then base load operation may be used if it provided for

by the maneuvering analysis. During base load operation, the measured AFD must be within a relatively small AFD band about a plant measured target AFD. The size of the AFD band is specified by the maneuvering analysis. Note that this target may or may not be within the AFD - power level operating limits. Base load may not be entered unless the plant has been relatively stable in AFD and power level for a period of time. The power level must be above the Allowed Power Level (APL - a value supplied by the maneuvering analysis) and the AFD must be within the AFD - power level operating limits. The power level may then be increased to a maximum of 100% rated thermal power or the Maximum Base Load Power (MBLP - a value described below).

6.4.2. Control Rod Insertion Limits

The control rods must be maintained within the insertion limits that were determined by the maneuvering analysis. Example limits are shown in Figure 12. These limits are a Limiting Condition of Operation, so operation outside of these limits is allowed for short periods of time.

6.4.3. Heat Flux Hot Channel Factor - $F_Q(x,y,z)$

The in-core instrumentation system is used periodically to measure $F_Q^M(x,y,z)$, which must always be within applicable limits. The LOCA limit is specified in the Technical Specifications (2, 3) and is shown in Section 6.1.

This limit on $F_Q^M(x,y,z)$ is a Limiting Condition of Operation, so operation outside of the limit is allowed for a short period of time to allow the operator to bring the reactor back within the limits without a reactor trip.

$F_Q^M(x,y,z)$ is usually measured at or near nominal conditions. To ensure that $F_Q^M(x,y,z)$ meets applicable limits for LOCA and CFM, the following limits are imposed at nominal conditions:

For nominal operation: $F_Q^M(x,y,z) \leq F_Q^L(x,y,z)^{OP}$ and
$F_Q^M(x,y,z) \leq F_Q^L(x,y,z)^{RPS}$, or

For base load operation: $F_Q^M(x,y,z) \leq F_Q^{Max\ BL}(x,y,z)$

$F_Q^L(x,y,z)^{OP}$ and $F_Q^L(x,y,z)^{RPS}$ are generated in the maneuvering analysis. These limits are specified in the Technical Specifications (2, 3) and are not imposed on the top or bottom 15% of the core. The limits on $F_Q^M(x,y,z)$ account for an appropriate measurement uncertainty, which is provided in the Technical Specifications (2, 3).

If $F_Q^M(x,y,z)$ exceeds $F_Q^L(x,y,z)^{OP}$ (LOCA limits), the AFD - power level limits must be adjusted by reducing the allowed AFD span (move the negative and positive AFD limits closer to the zero AFD point), so that positive margin would be maintained at the extremes of the AFD - power level operating limits. If $F_Q^M(x,y,z)$ exceeds $F_Q^L(x,y,z)^{RPS}$ (CFM limits), then a reduction is made to the OTΔT trip setpoints.

For base load operation, reactor power must be reduced until the above limit on $F_Q^{Max\ BL}(x,y,z)$ is satisfied. For base load operation, reactor thermal power may not exceed the Maximum Base Load Power (MBLP), which is defined as:

$$MBLP = \underset{\substack{over \\ (x,y,z)}}{Min} \frac{F_Q^{Max\ BL}(x,y,z)*100\%}{F_Q^M(x,y,z)}$$

Note that this is equivalent to saying that $F_Q^M(x,y,z)$ may not exceed $F_Q^{Max\ BL}(x,y,z)$ for base load operation.

6.4.4. Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(x,y)$

$F_{\Delta H}^M(x,y)$ is measured at the same time that $F_Q^M(x,y,z)$ is measured with the in-core instrumentation system. $F_{\Delta H}^M(x,y)$ must be within the maximum allowed values used in the maneuvering analysis (see Figure 10 for sample MATP curves). This limit is a Limiting Condition of Operation, so operation

outside of this limit is permitted for a period of time to allow the operator to bring the reactor back within the limit without a reactor trip.

$F^{M}_{\Delta H}(x,y)$ is usually measured at or near nominal conditions. To ensure that $F^{M}_{\Delta H}(x,y)$ meets applicable limits for LOFA, the following limits are imposed at nominal conditions:

For nominal operation: $F^{M}_{\Delta H}(x,y) \leq F^{L}_{\Delta H}(x,y)^{SURV}$

For base load operation: $F^{M}_{\Delta H}(x,y) \leq F^{Max\ BL}_{\Delta H}(x,y)$

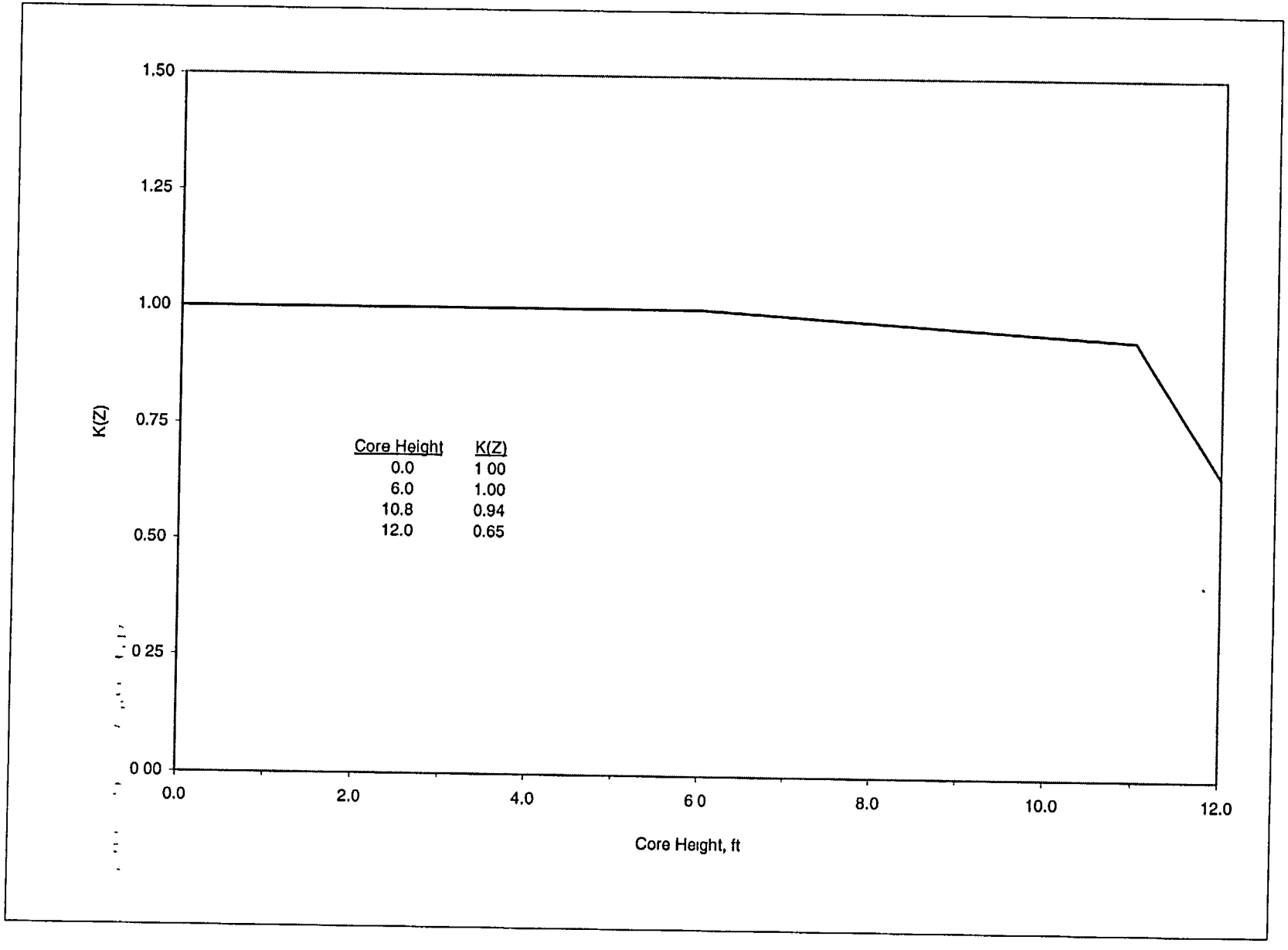
If the appropriate relationship is not satisfied, then the reactor power will be reduced until it is satisfied. The limits on $F^{M}_{\Delta H}(x,y)$ account for an appropriate measurement uncertainty, which is provided in the Technical Specifications (2, 3).

6.4.5. Quadrant Power Tilt

An allowance for a 2% quadrant power tilt was made in the AFD - power level operating limits and in the values of $F^{L}_{Q}(x,y,z)^{OP}$, $F^{L}_{Q}(x,y,z)^{RPS}$, $F^{Max\ BL}_{Q}(x,y,z)$, $F^{L}_{\Delta H}(x,y)^{SURV}$, and $F^{Max\ BL}_{\Delta H}(x,y)$. Thus, no action is required for an indicated quadrant power tilt of up to 2%. A quadrant power tilt larger than 2% is a Limiting Condition of Operation, so operation of the plant is allowed to continue for a period of time while the operator attempts to correct the condition.

Figure 9

K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height



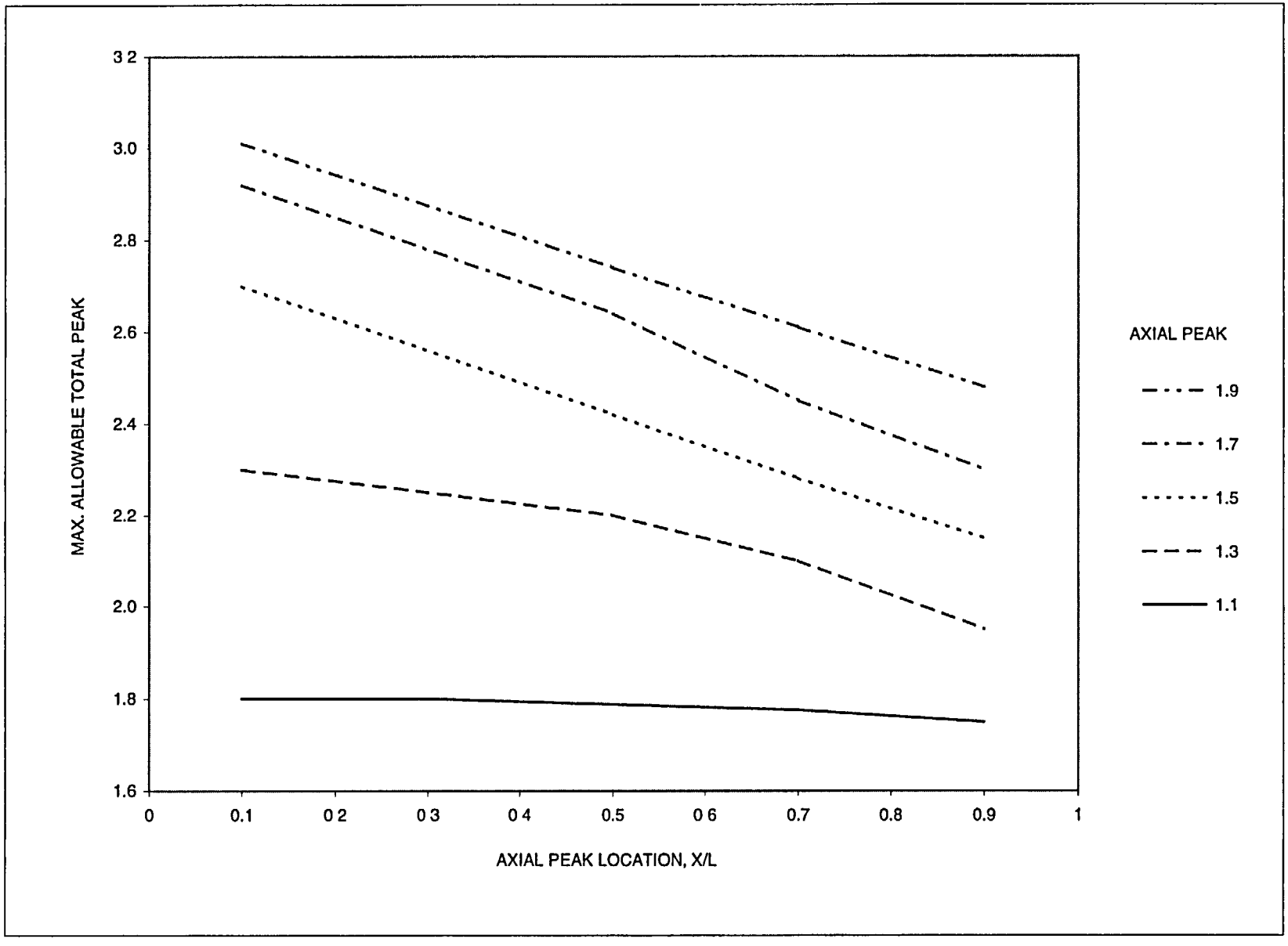


Figure 10
Sample LOFA DNB MATP Curves for 100% Power

Figure 11
Sample AFD - Power Level Operating Space

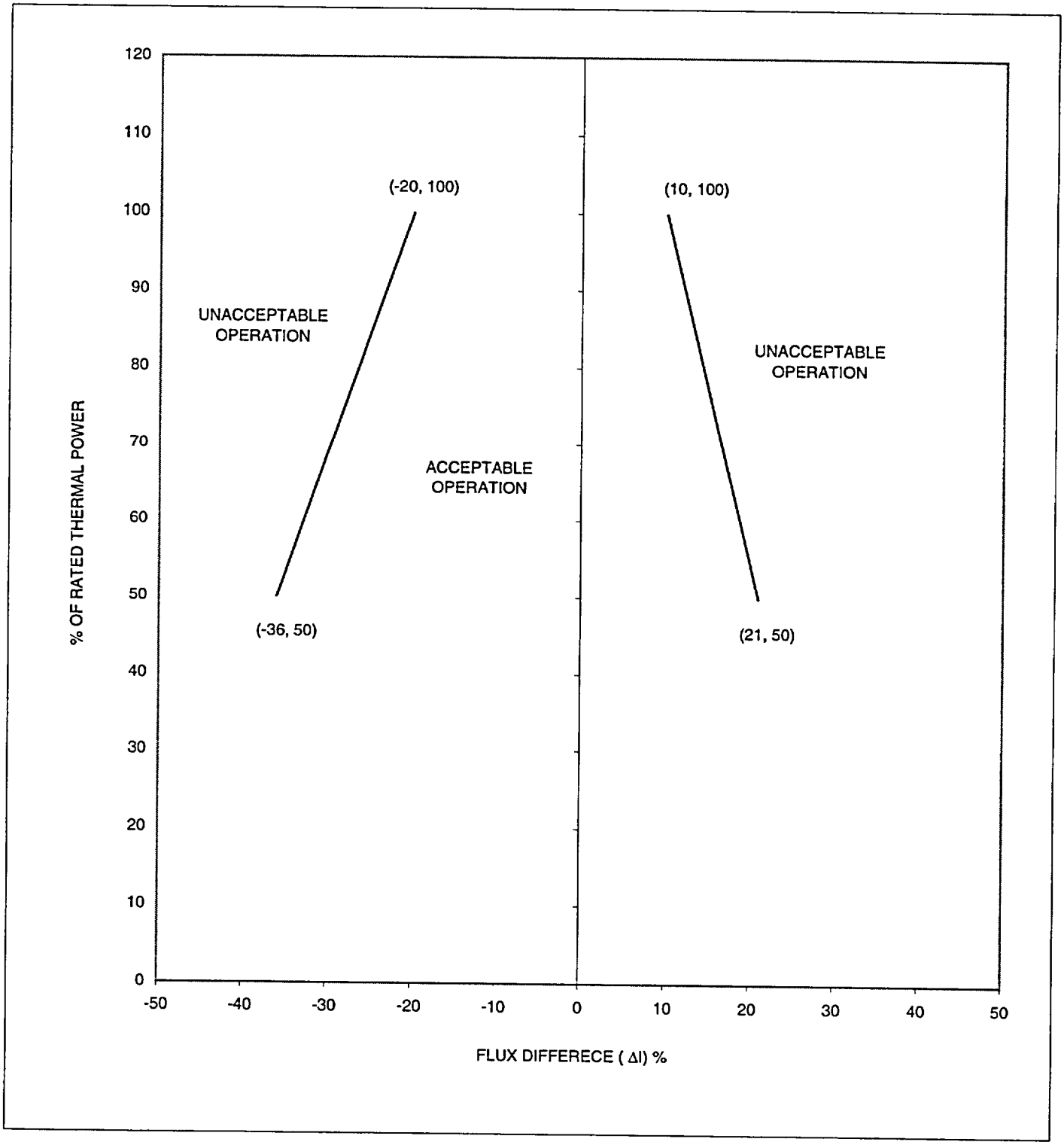
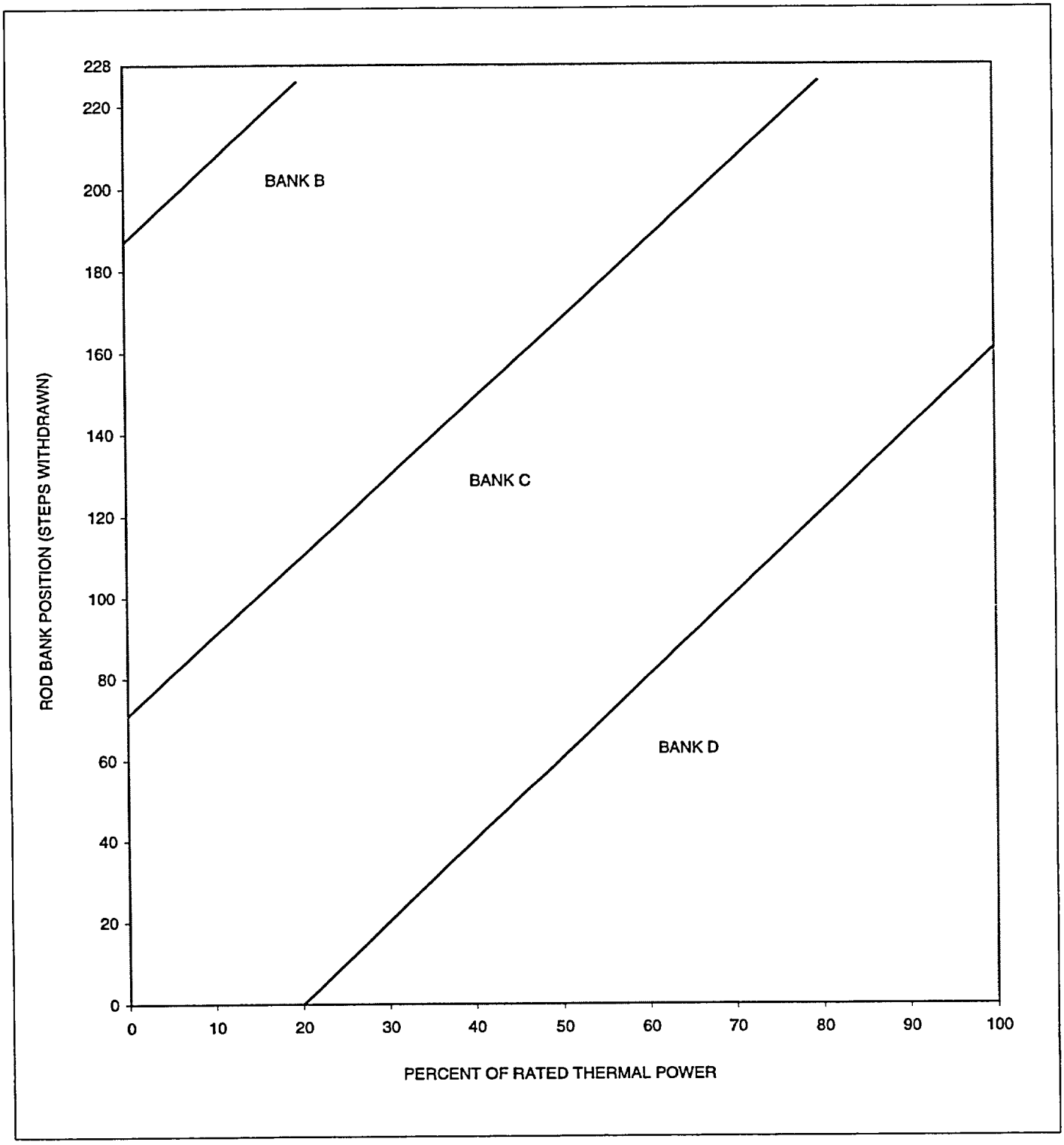


Figure 12
Control Rod Insertion Limits vs. Thermal Power



7. REFERENCES

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2. Technical Specifications for McGuire Nuclear Station Units No. 1 and 2, Docket Nos. 50-369/370.

3. Technical Specifications, Catawba Nuclear Station, Unit Nos. 1 and 2, Docket Nos. 50-413 and 50-414.

4. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Nuclear Physics Methodology for Reload Design", DPC-NF-2010A, June 1985.

5. Letter from Cecil O. Thomas, Chief Standardization and Special Projects Branch to H. B. Tucker, Vice President Nuclear Production, DPC, March 13, 1985.

6. Cobb, W. R., Eich, W. J., Tivel, D. E., "EPRI-CELL Code Description," EPRI-ARMP System Documentation, CCM-3, Part II, Chapter 5, October 1978.

7. Studsvik Energitechnik AB, "CASMO-2, A Fuel Assembly Burnup Program," Studsvik/NR-83/3, 1981.

8. Rothleder, B. M., Fisher, J. R., "EPRI-ARMP System Documentation", CCM-3, Part II, Chapter 14, September 1977.

9. Delp, D. L., et al, "FLARE - A Three Dimensional Boiling Water Reactor Simulator", GEAP-4598, July 1964.

10. User's manual for PDQ-XD, CE Computer Services, September 1986.

11. "Computer Code Certification for PDQEDIT," DPC internal document.

12. "Computer Code Certification for MARGINS," DPC internal document.

13. "Computer Code Certification for MARGINPLOT," DPC internal document.

14. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Core Thermal-Hydraulic Methodology using VIPRE-01", DPC-NE-2004P-A, Revision 1, SER Dated February 20, 1997 (DPC Proprietary).

15. "Duke Power Company, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P", DPC-NE-1004A, Revision 1, SER Dated April 26, 1996.

16. "Westinghouse Fuel Transition Report", DPC-NE-2009-P-A, SER Dated September 22, 1999 (DPC Proprietary).

APPENDIX A

Computer Program Description

EPRI-NODE-P

NODE (8) is a three dimensional nodal program that is derived from FLARE (9). NODE computes a three dimensional power distribution with thermal hydraulic feedback, the core multiplication factor, the fuel burnup distribution and maintains a reactivity inventory. The physics models within NODE account for the presence of control rods, fuel and moderator temperatures, fixed burnable poisons, soluble boron, fuel depletion, and time dependent xenon and iodine. The input to NODE is generated either from CASMO-2E (7) data or from EPRI-CELL (6) color set PDQ data.

PDQ07

PDQ07 (10) is an industry accepted multi-group, multi-dimensional, neutron diffusion depletion program. The Combustion Engineering version of PDQ that is used by DPC has been modified with a two dimensional thermal hydraulic feedback model to account for fuel and moderator temperature distributions. PDQ uses cross sections from either CASMO-2E or EPRI-CELL.

PDQEDIT

PDQEDIT (11) is a utility program that reads the PDQ system files. The program has several abilities, one of which is to produce radial local power factors from the mesh average power file.

MARGINS

MARGINS (12) is a program written by DPC that computes the margin to thermal limits for LOCA F_Q, DNB and centerline fuel melt. MARGINS requires three dimensional power distribution data for input. The output of MARGINS is a file that contains one entry per power distribution; the entry contains the case and limit type identifiers, the core axial offset and the core margin to the thermal limit evaluated.

MARGINPLOT

MARGINPLOT (13) is a program written by DPC that plots the MARGINS data and computes the zero margin intercepts for the thermal limits data.

SIMULATE-3

SIMULATE (15) is an advanced two-group nodal code written by Studsvik based on the QPANDA neutronics model. SIMULATE computes three dimensional nodal and pin power distributions accounting for fuel and moderator temperature, fuel burnup, xenon distributions, control rods, burnable absorbers, and soluble boron. Cross-section input to SIMULATE is provided from CASMO.

APPENDIX B

NRC/DPC Correspondence Regarding NRC Request for Additional Information

SAG

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

MAR 3 1989

RECEIVED
MAR 10 1989
DUKE POWER CO.
LICENSING

Mr. Hal B. Tucker, Vice President
Nuclear Production
Duke Power Company
P. O. Box 33189
Charlotte, NC 28242

Dear Mr. Tucker:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE NUCLEAR DESIGN METHODOLOGY FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS, TOPICAL REPORT DPC-NE-2011P

The Reactor Systems Branch has reviewed the subject topical report and has concluded that additional information is required for us to complete this review.

Please submit the responses to the questions in the enclosure within 45 days of the receipt of this letter to enable the staff to complete its review. If you need any clarification, please contact Lambros Lois of my staff at 301-492-0890.

Sincerely,

M. Wayne Hodges

M. Wayne Hodges, Chief
Reactor Systems Branch
Division of Engineering & Systems Technology

Enclosure:
As stated

REQUEST FOR ADDITIONAL INFORMATION
DPC-NE-2011P

1. How do operating limits obtained via this methodology compare to limits based on the use of the present RAOC methodology?

2. Is the potential increase in the available margin associated with the subject methodology due solely to the use of three-dimensional analyses/monitoring, or do other aspects contribute?

3. There is no indication in that the methodology employed in generating and using the LOFA DNB MATP curves has been reviewed and accepted by the NRC.

4. The procedure for generating power distributions appears to involve running two xenon transients at each of three times in a cycle, followed by using xenon distributions from each transient/time-in-life to calculate instantaneous power distributions associated with various combinations of power level, inlet temperature and control rod bank position, as well as those occurring during the course of several anticipated transients.

   It appears that only four xenon distributions from each transient at each time in life are used along with the statepoint configurations given in Table 2. Please clarify/elaborate as to how many power level/inlet temperature/control rod/xenon statepoints are evaluated at each time in life.

5. Is there demonstrated assurance that the power distributions resulting from the above analyses are indeed conservative with respect to those that might occur, and that they sufficiently span the AFD/rod insertion power level operating spaces to permit an accurate determination of operating limits?

6. What is the basis for the 15 minute limit assumed in the analysis of the boron dilution accident?

7. Radial local factors appear to be obtained from a nominal all-rods-out depletion calculation for the cycle and are, therefore, only functions of assembly type and burnup. However, local peaking should also be affected by transient xenon, control presence, etc. What is the basis for not accounting for these effects?

8. What are the other components of UCT in addition to those specifically mentioned in 3.1?

9. How are the axial peaking due to grid spacers and densification spike effects accounted for in the margin calculations?

10. Please explain the basis for the use of SC in the CFMM calculation, and its form.

11. Please explain why the uncertainties considered in the linear heat rate equation for the CFMM calculation are different from those used in obtaining LOCAM given that they refer to the same basic quantity.

12. The definition of the TILT factor varies while its value appears to be constant. Please explain/elaborate.

13. Since the maneuvering analysis involves two xenon transients at three times in core life there are six F_Q^D and six $F_{\Delta H}^D$ design distributions available for comparisons to measurements. Have the errors introduced by the subsequent interpolation on cycle burnup and power level been quantified and included in the analysis? How are mismatches between the measured and design data associated with AFD and control rod position differences accounted for?

14. Are M_Q and $M_{\Delta H}$ minimum values over the cycle?

15. How are possible increases in peaking between measurements due to mechanisms other than tilt (e.g. burnup) accounted for in the F_Q and $F_{\Delta H}$ surveillance?

16. What are the similarities/differneces between base load operation and CAOC?

17. Under what conditions would the AFD target and operating band for base load operation not fall within the normal AFD-power level operating limits?

18. Why are the uncertainties associated with $F^D_{\Delta H}$ and $F^T_{\Delta H}$ in 6.2 different?

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

March 28, 1989

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Subject: McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nuclear Station
Docket Numbers 50-413 and -414
Topical Report DPC-NE-2011P,
"Nuclear Design Methodology for Core
Operating Limits of Westinghouse Reactors";
Response to Request for Additional Information

Attached are responses to questions regarding the subject topical report, which were transmitted by letter dated March 3, 1989.

Please note that the proprietary nature of the original topical report, as identified in my April 27, 1988 transmittal letter and accompanying affidavit, is maintained in the responses to these questions. Therefore, they should be withheld from public disclosure.

Very truly yours,

H. B. Tucker

SAG154/lcs

xc: Mr. Darl S. Hood, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Kahtan Jabbour, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. S. D. Ebneter, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
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Mr. W. T. Orders
NRC Resident Inspector
Catawba Nuclear Station

Mr. P. K. Van Doorn
NRC Resident Inspector
McGuire Nuclear Station

bxc: R. L. Gill, Jr.
P. G. LeRoy
J. S. Warren
R. H. Clark
T. C. Geer
G. D. Seeburger
R. O. Sharpe - MNS
R. M. Glover - CNS
File: MC, CN-801.01

Q1 How do operating limits obtained via this methodology compare to limits based on the use of the present RAOC methodology?

A1 The operating space AFD limits from this method are expected to be a few percent wider than the current RAOC limits.

Q2 Is the potential increase in the available margin associated with the subject methodology due solely to the use of three-dimensional analyses/monitoring, or do other aspects contribute?

A2 The margin increase is due primarily to analysis of three-dimensional power distributions, as opposed to the 1D/2D synthesized power distribution that the RAOC limits are based on.

Q3 There is no indication in that the methodology employed in generating and using the LOFA DNB MATP curves has been reviewed and accepted by the NRC.

A3 The general methodology for generating DNB MATP curves has previously been approved by the NRC as applied to Oconee Nuclear Station in the SER for the topical report, "Duke Power Company, Oconee Nuclear Station, Reload Design Methodology," NFS-1001A, April 1984. A topical report describing the codes and methods used by Duke Power for generating DNB MATP limits specifically for Westinghouse reactors was submitted to the NRC in January 1989 under the title, "Duke Power Company, McGuire and Catawba Nuclear Stations, Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004. When approved, this methodology will be used to generate DNB MATP curves for setting the core limits.

Q4 The procedure for generating power distributions appears to involve running two xenon transients at each of three times in a cycle, followed by using xenon distributions from each transient/time-in-life to calculate instantaneous power distributions associated with various combinations of power level, inlet temperature and control rod bank position, as well as those occurring during the course of several anticipated transients.

It appears that only four xenon distributions from each transient at each time in life are used along with the statepoint configurations given in Table 2. Please clarify/elaborate as to how many power level/inlet temperature/control rod/xenon statepoints are evaluated at each time in life.

A4 The matrix of statepoints shown below will be used as an initial guide and may be modified as experience is accumulated. A power distribution will be analyzed for each statepoint in the matrix below for[ ] That is, a set of[ ]three-dimensional power distributions will be analyzed to set limits.

List of State Points

[ ]

Q5 Is there demonstrated assurance that the power distributions resulting from the above analyses are indeed conservative with respect to those that might occur, and that they sufficiently span the AFD/rod insertion power level operating spaces to permit an accurate determination of operating limits?

A5 Yes. [

]

As shown in the response to question 4, the statepoint conditions will span the allowable rod insertion limits and the accident condition rod insertions as described in section 2.4 of the report. The power distributions will generally span the AFD space, although some extrapolation on AFD may be required at times. Therefore, the AFD/rod insertion space will be sufficiently analyzed to accurately determine the operating limits.

Q6 What is the basis for the 15 minute limit assumed in the analysis of the boron dilution accident?

A6 The 15 minute limit is based on the operator action time acceptance criteria of the Standard Review Plan, section 15.4.6-II.

Q7 Radial local factors appear to be obtained from a nominal all-rods-out depletion calculation for the cycle and are, therefore, only functions of assembly type and burnup. However, local peaking should also be affected by transient xenon, control presence, etc. What is the basis for not accounting for these effects?

A7 Duke Power has examined the effects of control rods and transient xenon on local peaking factors using both two-dimensional and three-dimensional models. In general, it has been observed that the limiting nodes in a specific case are located away from the inserted control rods. That is, the peak nodal power occurs in an unrodded plane and/or an assembly removed from the rodded assemblies by several assembly pitches. Therefore, the intra-assembly flux distribution of the limiting node is relatively unaffected by the flux gradients induced locally near the rodded assembly. Similarly, the transient xenon distributions, while significantly skewed globally, do not cause significant changes in local power distributions.

Q8 What are the other components of UCT in addition to those specifically mentioned in 3.1?

A8 UCT is defined in Reference 4 of the report to be

$$1 + (.031/1.375) + \sqrt{(.03)^2 + (.035)^2 + (.02)^2} = 1.073.$$

The term (.031/1.375) accounts for a small bias in the calculated power distributions.

Q9 How are the axial peaking due to grid spacers and densification spike effects accounted for in the margin calculations?

A9 In the development of the observed reliability factors the calculated peaks did not include any grid effects while the measured data did. Therefore, the effects of the grid on peaking are inherently included in the observed reliability factors which are applied to the calculated values.

Current fuel designs used by Duke Power specify fuel pellet density greater than or equal to 95% of theoretical density. Results of hot cell and gamma scan measurements on fuel rods containing pellets of these densities have not shown any significant gap formation. Thus, no power peaking penalty will be taken for densification power spikes.

Q10 Please explain the basis for the use of SC in the CFMM calculation, and its form.

A10 A rod bow penalty is applied to the calculated peak when computing CFMM. However, since rod bow is considered to be independent of the calculational uncertainty, it is statistically combined with the engineering and power distribution factors in the equation for UCT found

in Reference 4 of the report. The algebraic derivation is shown below:

$$UCT = 1 + .031/1.375 + \sqrt{(.03)^2 + (.035)^2 + (.02)^2}$$

$$SC = 1 + .031/1.375 + \sqrt{(.03)^2 + (.035)^2 + (.02)^2 + (RBOW-1)^2}$$

$$\sqrt{(.03)^2 + (.035)^2 + (.02)^2} = UCT - 1 - .031/1.375$$

$$SC = 1 + .035/1.375 + \sqrt{(UCT - 1 - .031/1.375)^2 + (RBOW-1)^2} .$$

Q11 Please explain why the uncertainties considered in the linear heat rate equation for the CFMM calculation are different from those used in obtaining LOCAM given that they refer to the same basic quantity.

A11 The only difference is that the rod bow penalty is not applied to the LOCA limits, since any increase in peaking will be compensated for by the increased coolant flow.

Q12 The definition of the TILT factor varies while its value appears to be constant. Please explain/elaborate.

A12 The magnitude of the tilt factor is the same in all sections and the correct definition in all sections is "peaking increase due to allowable quadrant tilt."

Q13 Since the maneuvering analysis involves two xenon transients at three times in core life there are six F_Q^D and six $F_{\Delta H}^D$ design distributions available for comparisons to measurements. Have the errors introduced by the subsequent interpolation on cycle burnup and power level been quantified and included in the analysis? How are mismatches between the measured and design data associated with AFD and control rod position differences accounted for?

A13 The values of F_Q^D and $F_{\Delta H}^D$ from the design power distributions are not the values that are compared to measurements. [

This is very similar to the current monitoring methods which apply burnup-dependent W(Z) transient peaking factors to the measured peaks.]

[ ]

The impact on peaking of differences between the measured and design data for AFD are inherently included in the uncertainty factors which are applied to the predicted peaks. The uncertainty factor used is an observed nuclear reliability factor developed by matching reactor power and rod positions between predicted and measured statepoints. The calculated AFD was allowed to vary from the measured value in these calculations, although these differences are generally within 2%. The impact of control rod position differences between measured and design data is considered negligible since power distribution maps are usually taken at nearly all-rods-out conditions.

Q14 Are M_Q and $M_{\Delta H}$ minimum values over the cycle?

A14 [ ]

Q15 How are possible increases in peaking between measurements due to mechanisms other than tilt (e.g., burnup) accounted for in the F_Q and $F_{\Delta H}$ surveillance?

A15 If F_Q^M is greater than F_Q^{Max}, then the AFD power level space is reduced by an appropriate amount such that F_Q^T, at the new AFD limit, will be within the LOCA limits.

If $F^M_{\Delta H}$ is greater than $F^{Max}_{\Delta H}$, then power level will be reduced until the limit is met.

If the margins to the limits are found to be decreasing over successive measurements, then either the measurement frequency will be increased or the margins will be reevaluated with an additional penalty to account for the expected peaking increase to the next measurement.

Q16 What are the similarities/differences between base load operation and CAOC?

A16 The only significant difference is the power level at which the mode of operation may be entered. Base load operation is typically entered at 80% power after stabilizing the plant at the target AFD. CAOC is used for the full range of power operation.

Q17 Under what conditions would the AFD target and operating band for base load operation not fall within the normal AFD-power level operating limits?

A17 This condition is not expected to occur since the AFD-power level limits will be set each cycle with a cycle specific three-dimensional core model. However, operating for a significant period of time at reduced power may cause the AFD target to be outside of the operating AFD space. If this condition should occur, the surveillance of the measured peaking will ensure that the allowable limits are not exceeded and tighter AFD limits would be used to minimize potential transient peaking.

Q18 Why are the uncertainties associated with $F^D_{\Delta H}$ and $F^T_{\Delta H}$ in 6.2 different?

A18 A typographical error was made in the equation for $F^T_{\Delta H}$. The equation that was intended is:

[ ]

However, in further research it was discovered that a rod bow penalty does not need to be applied to a limit that is related to DNB. This approach has previously been approved by the NRC in the SER to "Duke Power Company Oconee Nuclear Station Reload Design Methodology II," DPC-NE-1002A, October 1985. Thus, the uncertainties in $F^D_{\Delta H}$ and $F^T_{\Delta H}$ should be the same. The correct equation for $F^T_{\Delta H}$ is:

[ ]

Also, the rod bow penalty should be removed from the calculation of DNBM. In section 4.3 of the report, the equation for RPP(x,y) should be:

[ ]

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

June 26, 2002

Mr. M. S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church St
Charlottte, NC 28202

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: REQUEST FOR ADDITIONAL INFORMATION - APPLICATION FOR CHANGES TO TECHNICAL SPECIFICATIONS (TAC NOS. MB3343, MB3344, MB3222 AND MB3223)

Dear Mr. Tuckman:

The Nuclear Regulatory Commission is reviewing your application dated October 7, 2001, entitled "License Amendment Request applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to Bases 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003" and has identified a need for additional information as identified in the Enclosure. These issues were discussed with your staff on June 6, 2002. Please provide a response to this request within forty-five (45) days of receipt of this letter so that we may complete our review.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosure: Request for Additional Information

cc w/encl: See next page

cc:
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3825 Barrett Drive
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4800 Concord Road
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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST APPLICABLE TO

TECHNICAL SPECIFICATION 5.6.5, CORE OPERATING LIMITS REPORT,

REVISIONS TO BASES 3.2.1 and 3.2.3

REVISIONS TO TOPICAL REPORTS DPC-NE-2009-P,

DPC-NF-2010, DPC-NE-2011-P, AND DPC-NE-1003

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

Topical Reports Numbered DPC-NE-2009-P Duke Power Company Westinghouse Fuel Transition Report and DPC-NF-2010-A, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design

1. Please provide a detailed qualitative technical justification for the requested changes to the topical reports (methodologies), DPC-NE-2011 and DPC-NF-2010. (i.e., why are these changes being made?).

2. To expedite the review process, please provide a qualitative and quantitative technical basis for each of the changes in these topical reports.

3. Please provide validation data that bench-marks the results of comparisons between the old and the new models (changes).

4. If the changes to these topical reports and methodologies impact the safe operation of the reactor core, please provide the safety significance (impact) of each of these changes.

5. Please provide the basis for why the proposed changes to the above stated topical reports should be found acceptable.

Topical Report Numbered DPC-NF-2010-A, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design

1. In the revision history section on page ii, the licensee provides the staff with the reason for the submittal. Since this is a licensing action, please list those Technical Specification(s), Bases, FSAR sections, conformance to regulatory documents, criteria, generic letters, etc. that are impacted by the request for these changes within the licensing framework.

2. Section 4.2.4.2, second paragraph. Please provide clarification of this change and the technical justification for it. Please provide a comparison between the old sentence and the new sentence.

3. In Attachment 7a, "Detailed Listing of the Changes to DPC-NF2010A," it is stated in many places, that "this change is made to avoid difficulties with the literal interpretation of the original description." Please provide clarification of this statement with a supporting example.

4. Section 4.2.4.4, fifth paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

5. Section 8.1, first paragraph. Is the added equation the same as that in the current version of the DPC-NF-2010A topical? If not, please provide technical justification for its use.

6. Section 9.1.5, first paragraph. Please provide clarification of this change and the technical justification for it. Please provide a comparison between the old sentence and the new sentence.

Topical Report Numbered DPC-NE-2011-P-A, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors

1. The description of the transient conditions was changed in Tables 1 and 2, of Section 2.5. It is not clear to the staff exactly what was changed. Please clarify.

2. From section 6.1, please explain what is meant by "updated the equation."

3. From section 6.1, please provide further clarification of this statement.

4. Section 6.2, were is UMR listed in section 6.2? Please provide original definition and new definition for comparison.

Topical Report Numbered DPC-NE-1003, Revision 1 McGuire Nuclear Station and Catawba Nuclear Station Rod Swop Methodology Report for Startup Physics Testings, Revision 1

1. Appendix A of topical report DPC-NE-1003, Revision 1, contains two versions of Duke Power Company's rod swap measurement procedure PT/O/A/4150/11A: Attachment 3 (dated June 1986) and Attachment 4 (dated April 1984). There are differences in these two versions of the procedure. For example, in the Attachment 3 version, Steps 12.2.2 and 12.2.3, respectively, specify the insertion of bank 1 until the indicated reactivity is approximately -20 pcm, and the withdrawal of reference bank until the indicated reactivity is approximately +20 pcm; whereas in the Attachment 4 version, the insertion and withdrawal of bank 1 and reference bank, respectively, of steps 12.2.1 and 12.2.2 specify reactivity change of ~~+/-~~ 10 pcm.

a. Since the Attachment 3 version of procedures is more recent, why is the Attachment 4 version referenced in Revision 1 of the topical report (Reference 2)?

b. Which of these two versions of rod swap measurement procedures will be used for McGuire and Catawba Units?

2. In the Attachment 3 version of rod swap measurement procedures PT/O/A/4150/11A, Step 12.1.3 states that: "Repeat steps 12.2.1 and 12.2.2 until the previously inserted bank is fully withdrawn."

   Is there a typographic error in the words "steps 12.2.1 and 12.2.2"? Should correct words be "steps 12.1.1 and 12.1.2"?

3. The equation in Section 3, Measurement Procedure, of the topical report for calculating the inferred rod worth of bank x is different from the equation in Step 12.5.3 of the Attachment 3 procedures. The difference appears to be due to the initial height of the reference bank for performing the rod swap measurement of the measured bank.

   Clarify the exact procedure to be used in the rod swap test, and make all necessary corrections in the topical report and the procedures to be consistent.

4. The third sentence in Section 3 of the topical report is revised to read: "All other banks are then exchanged with the reference bank or other test banks at constant boron conditions until the measured bank is fully inserted." It is stated, in Attachment 9a, "Detailed Listing of Changes to DPC-NE-1003A," that the third sentence in Section 3 is revised to make the report consistent with current procedures. The "Revision History" in the topical report states that this revision [Revision 1] also reflects a refinement in the rod swap to make use of two test banks.

   a. What are the current procedures? What is the date of the current procedures?

   b. Are the current procedures the same or different from the ones in Attachment 3? The Attachment 3 procedures do not include the exchange of a test bank with the other test bank.

   c. If the current procedures are different from those of Attachment 3 or 4, provide a copy of the procedures, and appropriately reference them in the report.

   d. Is the statement in "Revision History" referring to this revision? Please explain what the statement means.

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Michael S. Tuckman
Executive Vice President
Nuclear Generation

August 7, 2002

U. S. Nuclear Regulatory Commission
Washington D.C. 20555-0001
ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 370

Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 414

Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343, and MB3344) and License Amendment Request Supplement

This purpose of this letter is to provide Duke Energy Corporation's (Duke) response to an NRC request for additional information (RAI) and to supplement a Duke license amendment request (LAR) previously submitted pursuant to 10CFR50.90. Please note that some of the information contained in this submittal package has been determined to be proprietary and is being submitted pursuant to 10CFR2.790. This proprietary information is discussed below.

Duke submitted[1] a LAR applicable to McGuire and Catawba Technical Specifications (TS) 5.6.5.a and 5.6.5.b. Also included in this submittal were proposed revisions to the four Duke Topical Reports listed below.

[1] Reference 1: Letter, Duke Energy Corporation to U.S. Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Dated October 7, 2001, SUBJECT: License Amendment Request Applicable to Technical Specification 5.6.5, Core Operating Limits Report; Revisions to Bases 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003

- DPC-NE-2009-P, *Duke Power Company Westinghouse Fuel Transition Report*, Revision 1;

- DPC-NF-2010, *Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design*, Revision 1;

- DPC-NE-2011-P, *Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*, Revision 1;

- DPC-NE-1003, McGuire Nuclear Station and Catawba Nuclear Station *Rod Swap Methodology Report for Startup Physics Testing*, Revision 1.

The NRC RAI[2] asked questions on these topical reports. As described below, the Duke responses to these questions are included in the attachments to this letter.

In a subsequent submittal,[3] Duke proposed another LAR for McGuire and Catawba TS 5.6.5, but this LAR was only applicable to TS 5.6.5.b. The information contained herein explains the necessary coordination for changing TS 5.6.5.b for McGuire and Catawba. This LAR implements the provisions of an NRC approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler.[4] The NRC has approved and issued this LAR for both McGuire[5] and Catawba.[6] Implementation of the

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[2] Reference 2: Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation, Dated June 26, 2002, SUBJECT: Request for Additional Information, Application for Changes to Technical Specifications (TAC Nos. MB3222, MB3223, MB3343, and MB3344

[3] Reference 3, Letter, Duke Energy Corporation to U.S. Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Dated December 20, 2001, SUBJECT: License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report – Oconee, McGuire, and Catawba Technical Specification 5.6.5

[4] TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR"

[5] Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation Dated July 10, 2002, SUBJECT: McGuire Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3702 and MB3703)

[6] Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation Dated July 2, 2002, SUBJECT: Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3728 and MB3729)

referenced industry traveler eliminates the need for the changes Duke proposed to McGuire and Catawba TS 5.6.5.b in Reference 1. The LAR supplement transmitted herein deletes the proposed changes to McGuire and Catawba TS 5.6.5.b contained in Reference 1. The attached McGuire and Catawba TS pages (both marked and reprinted versions) update Reference 1 such that it contains the latest approved version of the affected TS pages and only applies to McGuire and Catawba TS 5.6.5.a. The affected TS pages are:

McGuire Units 1 and 2 Pages: 5.6-2, 5.6-3, B3.2.1-11, and B3.2.3-4; and

Catawba Units 1 and 2 Pages: 5.6-3, B3.2.1-11, and B3.2.3-4.

As shown, conforming Bases changes have been made and the necessary Bases pages are also included.

The attachments to this letter are listed and described below.

- Attachment 1 provides the Duke response to the NRC's general questions on Topical Reports DPC-NF-2010 and DPC-NE-2011-P.

- Attachment 2 provides the Duke response to the NRC's specific questions on Topical Report DPC-NF-2010.

- Attachments 3a and 3b provide the Duke responses to the NRC's specific questions on Topical Report DPC-NE-2011-P. Attachment 3a is the proprietary version and Attachment 3b is the non-proprietary version.

- Attachment 4 provides the Duke response to the NRC's specific questions on Topical Report DPC-NE-1003.

- Attachment 5 provides the Duke response to an NRC concern on Topical Report DPC-NE-2009-P. This concern was not included in the NRC's RAI,[2] however it was discussed during an NRC/Duke telephone conference held on July 24, 2002.

- Attachments 6a and 6b provide a marked copy of the existing approved Technical Specifications pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively. These marked copies show the proposed changes.

- Attachments 7a and 7b provide the reprinted Technical Specifications and Bases pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively.

Duke has determined that the revisions contained in this LAR supplement, as shown in Attachments 6a, 6b, 7a, and 7b have no impact on the determination of no significant hazards consideration that was included in Reference 1.

This submittal package contains information that Duke considers proprietary. This information is contained within the proprietary version of the response to the NRC questions on Topical Report DPC-NE-2011-P that is provided as Attachment 3a to this letter. In accordance with 10CFR2.790, Duke requests that this information be withheld from public disclosure. An affidavit that attests to the proprietary nature of this information is included with this letter. A non-proprietary version of this response is also provided as Attachment 3b to this letter.

Inquiries on this matter should be directed to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

xc w/Attachments:

C. P. Patel (Addressee Only)
NRC Senior Project Manager (CNS)
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R. Wingard, Director
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South Carolina Bureau of Land and Waste Management
2600 Bull Street
Columbia, SC 29201

M. S. Tuckman, affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to me: August 7, 2002
Date

Mary P. Dehus, Notary Public

My commission expires: JAN 22, 2006

SEAL

bxc w/Attachments:

M. T. Cash
C. J. Thomas
G. D. Gilbert
L. E. Nicholson
K. L. Crane
K. E. Nicholson
J. M. Ferguson (2) - CN01SA
L. J. Rudy
G. A. Copp
R. L. Gill
P. M. Abraham
G. G. Pihl
D. R. Koontz
R. C. Harvey
MNS Master File - MG01DM
Catawba Master File - CN04DM
NRIA/ELL

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NC Municipal Power Agency No. 1
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Raleigh, NC 27626-0513

T. R. Puryear
NC Electric Membership Corporation
CN03G

Piedmont Municipal Power Agency
121 Village Drive
Greer, SC 29651

Attachment 1

Responses to Request for Additional Information

Topical Reports Numbered DPC-NE-2011-P, Revision 1, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors and DPC-NF-2010, Revision 1, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design (TAC NOS. MB3343, MB3344, MB3222, MB3223)

General

Subsequent to receiving the NRC RAI package, a clarification of Questions 1, 2, and 3 was obtained from the NRC during a conference call on Thursday July 18, 2002. Responses to all questions in the NRC RAI are given below, and responses to Questions 1, 2, and 3 take into account the clarification received from the NRC.

Question 1. Please provide a detailed qualitative technical justification for the requested changes to the topical reports (methodologies), DPC-NE-2011 and DPC-NF-2010. (i.e., why are these changes being made?).

Response

Subsequent to the approval of the current version of these reports, there have been various changes in calculation methods and plant operating philosophy. Therefore, sections of these topical reports affected by these changes have been reviewed and updated to improve clarity and continuity in order to avoid ambiguities and inconsistencies that could be misconstrued. These revisions do not change approved methods nor introduce new methods. These changes and justifications were identified and described in the October 7, 2001 DEC submittal.

Question 2. To expedite the review process, please provide a qualitative and quantitative technical basis for each of the changes in the above stated topical reports.

Response

Qualitative and quantitative bases for each change to DPC-NF-2010 and DPC-NE-2011-P are provided in Attachments 7a and 8a, respectively in the License Amendment Request package submitted by Duke with a cover letter date of October 7, 2001.

Question 3. Please provide validation data, bench-marking the results of comparisons between the old and the new models (changes).

Response

These revisions do not change approved methods nor introduce new methods; therefore, additional benchmarking is not necessary.

Question 4. If the changes to these topical reports/methodologies impact the safe operation of the reactor core, please provide the safety significance (impact) of each of these changes?

Response

The methodology changes correspond to previously approved methodologies or licensing basis documents, or to administrative non-technical changes. Therefore, these changes do not impact the safe operation of the reactor core.

Question 5. Please provide the basis as to why the proposed changes to the above stated topical reports should be found acceptable.

Response
The purpose for these changes is to maintain the topical reports in a condition that is consistent with other current, NRC approved licensing related documents and to improve clarity and continuity in order to avoid ambiguities and inconsistencies that could be misconstrued. The changes do not change previously approved methodologies.

Question 1. The description of the transient conditions was changed in Tables 1 and 2, of Section 2.5. It is not clear to the staff what exactly was changed. Please clarify.

Response
The values in these tables are labeled "Typical", so the values shown are examples and may be changed to generate power distributions that are appropriate for the desired AFD span.

Table 1

[ ]

Table 2

[ ]

Questions 2 and 3. Section 6.1, Please explain what is meant by "updated the equation". Section 6.1, please provide further clarification of this statement.

Response
The equation in the current version of this topical report is not consistent with current Technical Specifications and COLR. The Technical Specifications and COLR supersede the current version of this topical report. As a result, the equation in the proposed version of the topical report is "updated" to make the topical report nomenclature consistent with the Technical Specifications and COLR. The equations in the current and proposed version of this topical report are described below.

The LOCA Fq limit equation in the current version of this topical report is as follows:

$Fq \leq 2.32 * K(Z) / P$ for $P > 0.5$
$Fq \leq 2.32 * K(Z) / 0.5$ for $P \leq 0.5$.

The LOCA Fq limit equation in the proposed version of this topical report is as follows:

$Fq \leq FqRTP * K(Z) / P$ for $P > 0.5$
$Fq \leq FqRTP * K(Z) / 0.5$ for $P \leq 0.5$.

Where: P = relative thermal power
K(Z) = normalized Fq limit as a function of core height (vendor specific penalty factor)
Fq = measured Fq
FqRTP = Fq limit at rated thermal power as defined in the COLR

The specific Fq limit of 2.32 was removed, because this value may be reload specific and the current process is to control the Fq limit in the COLR. By making the topical report consistent with the Technical Specifications and COLR, an inconsistency between Technical Specifications and DPC-NE-2011 is removed.

While developing this response, DEC noted a typographical error in Section 6.1 on Page 6-1 of the proposed version of this topical report (namely, several 'less than' (<) signs should have been 'less than or equal to' (≤) signs). A marked up copy and a reprinted copy of this page (Page 6-1) is is included at the end of Attachment 3b.

Question 4. Section 6.2, where is UMR listed in section 6.2? Please provide original definition and new definition for comparison.

Response
The changes listed in Attachment 8a of the LAR submitted by Duke correspond to the section numbering found in the current version of this topical report. Therefore, all the changes associated with Section 6.2 in Attachment 8a are located in Section 6.3 of the proposed version of the topical report. UMR is not used in Section 6.2 of the proposed version of the topical report but is used in Section 6.3.

Original Definition: In Section 6.2 of the current version of the topical report, UMR is defined "Uncertainty value for measured radial peaks, taken as 1.04 in the current Technical Specifications (2, 3)."

Proposed Definition: In Section 6.3 of the proposed topical report, UMR is defined "Uncertainty factor on the measured radial peaks, provided in the Technical Specifications (2, 3)."

This definition was updated to reflect that the value for UMR is to be found in the COLR as referenced by the Technical Specifications. This change is made to avoid a conflict if this value were to change in the future. As a result, the topical report now references Technical Specifications.

Attachment 3b - Non-Proprietary
Responses to Request for Additional Information
Topical Report Numbered DPC-NE-2011-P, Revision 1, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors
(TAC NOS. MB3343, MB3344, MB3222, MB3223)

The following page of this Attachment contains the marked up and reprinted page that is revised from the proposed version of this topical report. This page is being provided in response to Question 3.

6. POWER DISTRIBUTION SURVEILLANCE

The AFD - power level limits are set to preserve the power peaking assumptions in the LOCA analysis and to protect the fuel from damage during a LOFA when the power distribution is skewed in the axial direction. Similary, f(ΔI) limits are set to preclude RPS limits from being exceeded during Condition II transients. Because only steady state power distributions can be measured with reasonable accuracy, the limits on the measured power distribution are reduced by pre-calculated factors that account for perturbations from steady state conditions to applicable limits.

6.1. LOCA F_Q Surveillance Methodology

The Technical Specification (2, 3) LOCA F_Q limit that must be satisfied within the AFD - power level operating limits is:

$$F_Q^M(x,y,z) < \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q^M(x,y,z) < \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P < 0.5$$

Where: P = relative thermal power.

K(Z) = normalized F_Q as a function of core height (see Figure 9).

F_Q^{RTP} = the LOCA limit at rated thermal power (RTP).

This criterion is a Technical Specification (2, 3) limiting condition for operation (LCO).

Using definitions from Section 4.2, the reduced limits for the measured F_Q are specified as:

$$F_Q^M(x,y,z)*UMT*MT*TILT \leq [\qquad\qquad]$$

Where:

$F_Q^M(x,y,z)$ = The measured total peak in location x,y,z.

6. POWER DISTRIBUTION SURVEILLANCE

The AFD - power level limits are set to preserve the power peaking assumptions in the LOCA analysis and to protect the fuel from damage during a LOFA when the power distribution is skewed in the axial direction. Similary, f(ΔI) limits are set to preclude RPS limits from being exceeded during Condition II transients. Because only steady state power distributions can be measured with reasonable accuracy, the limits on the measured power distribution are reduced by pre-calculated factors that account for perturbations from steady state conditions to applicable limits.

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$$F_Q^M(x,y,z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

Where: P = relative thermal power.
$K(Z)$ = normalized F_Q as a function of core height (see Figure 9).
F_Q^{RTP} = the LOCA limit at rated thermal power (RTP).

This criterion is a Technical Specification (2, 3) limiting condition for operation (LCO).

Using definitions from Section 4.2, the reduced limits for the measured F_Q are specified as:

$$F_Q^M(x,y,z)*UMT*MT*TILT \leq [\qquad\qquad]$$

Where:

$F_Q^M(x,y,z)$ = The measured total peak in location x,y,z.

APPENDIX C

Original Issue NRC SER

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

January 24, 1990

Mr. H. B. Tucker, Vice President
Nuclear Production
Duke Power Company
P. O. Box 33189
Charlotte, NC 28242

Dear Mr. Tucker:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT DPC-NE-2011P, "DUKE POWER COMPANY NUCLEAR DESIGN METHODOLOGY FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS"

The staff has completed its review of the Topical Report DPC-NE-2011P, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors" submitted for NRC review by the Duke Power Company by letter dated April 27, 1988. Additional information was submitted on March 28, 1989. This topical report (DPC-NE-2011P) provides information and justification for the operating limits on power distribution, control rod insertion and power distribution inputs to the overpower-delta-T and overtemperature-delta-T reactor protection system trip functions. These limits are the axial flux difference for a given power level, the rod insertion limits and the f(delta-I) function of the overpower- and overtemperature-delta-T. These operating limits provide assurance that the peak local power is not greater than that assumed in the design basis transient and accident analyses. The limits are set such that the RPS will trip the reactor before fuel damage occurs. A three-dimensional reactor model power distribution is employed for the maneuvering analyses in several points in the core life. These power distributions are based on a set of conservative xenon distributions to ensure that the predicted power distributions are conservative with respect to those expected to occur. These power distributions are augmented by appropriate uncertainty factors.

We find the application of DPC-NE-2011P to be acceptable for referencing in license applications to the extent specified, and under the limitations delineated, in DPC-NE-2011P and the associated NRC technical evaluation. The evaluation defines the basis for acceptance of this topical report.

We do not intend to repeat our review of the matters found acceptable as described in DPC-NE-2011P when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the application of DPC-NE-2011P.

In accordance with procedures established in NUREG-0390, it is requested that the Duke Power Company publish accepted versions of this topical report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, Duke Power Company and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

for Ashok C. Thadani, Director
Division of Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
DPC-NE-2011P Evaluation

ENCLOSURE

SAFETY EVALUATION FOR THE TOPICAL REPORT DPC-NE-2011P
"DUKE POWER COMPANY, NUCLEAR DESIGN METHODOLOGY FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS"

1.0 INTRODUCTION

By letter dated April 27, 1988, the Duke Power Company submitted the Topical Report DPC-NE-2011P for NRC review (Ref. 1). Additional information was submitted on March 28, 1989 (Ref. 2). This topical report provides information and justification for the operating limits on power distribution, control rod insertion and power distribution inputs to the overpower-delta-T and overtemperature-delta-T reactor protection system trip functions. These limits are the axial flux difference for a given power level, the rod insertion limits and the f(delta-I) function of the overpower- and overtemperature-delta-T. These operating limits provide assurance that the peak local power is not greater than that assumed in the design basis transient and accident analyses. The limits are set such that the RPS will trip the reactor before fuel damage occurs. A three-dimensional reactor model power distribution is employed for the maneuvering analyses in several points in the core life. These power distributions are based on a set of conservative xenon distributions to ensure that the predicted power distributions are conservative with respect to those expected to occur. These power distributions are augmented by appropriate uncertainty factors.

The following evaluation incorporates our consultant's, BNL, contribution to this review. Restrictions to be observed in the application of this topical report are listed in Section 3.5.

2.0 SUMMARY OF THE TOPICAL REPORT

At first the report describes the three-dimensional nodal power and xenon distribution generation method which is based on an NRC approved version of

the EPRI-NODE-P code (Ref. 3). The local radial factors are estimated using a pin-by-pin PDQ-07 model. Power distributions are generated for different times in the cycle. Limiting xenon distributions are generated to assure conservatism. The power distribution is augmented by uncertainty factors which account for the (X-Y) power distribution calculation uncertainty, quadrant tilt and axial power distribution.

The general methodology for the limiting condition of operation and the reactor protection system limits is followed by the calculation of the LOCA margin and the estimation of the loss of flow DNB limits. In addition the reactor protection system margin, the centerline fuel melt margin, the axial flux difference power level limits and the control rod insertion limits are calculated.

The power distribution surveillance and their relation to the operation and transient limits are then estimated for the LOCA F_Q limits, the loss of flow DNB, $F_{\Delta H}$, axial flux difference power level limits, control rod insertion limits, the heat flux hot channel factor, the nuclear enthalpy rise hot channel factor and the quadrant power tilt.

Appendix A in the report gives a brief description of the computer codes used in the above calculations.

3.0 EVALUATION

The proposed methodology employs a three-dimensional reactor and cycle specific model in conjunction with xenon distributions obtained from a maneuvering analysis which simulates severe xenon transients. Bounding power distributions are then generated based on these severe xenon distributions, and various combinations of rod positions, inlet temperature, power level and cycle burnup. These power distributions are compared to operating and safety thermal limits to define or validate the axial flux difference (AFD) power level operating space, the rod insertion limits and the f(delta-I) penalty function employed in the OPΔT and/or the OTΔT trip functions of the Reactor Protection System (RPS) such that power distributions that might exceed the

respective thermal limits are prohibited. In addition to the xenon transient based power distributions, a number of anticipated transients (e.g., boron dilution, rod withdrawal, etc.) are analyzed in setting the RPS limits. A core monitoring/surveillance procedure which assures safe operation within the applicable limits is an integral part of the proposed methodology. This approach is an alternative to the Relaxed Axial Offset Control (RAOC) methodology (Ref. 4) currently in use at Duke Power Company's (DPC) McGuire and Catawba Nuclear Stations.

The present review considered the information provided in the topical report along with additional information provided by DPC in response to a request for additional information (RAI) (Ref. 5).

The computer codes and associated methodologies employed in the power distribution and peaking calculations have been previously reviewed by the NRC and found to be acceptable (Refs. 6 and 7). The shutdown margin and ejected rod analyses that enter into the setting of control rod insertion limits have also been approved by the NRC. A topical report describing the codes and methods to be used by DPC to generate the core thermal hydraulics (including hot rod) for Westinghouse (W) reactors is presently under review (Ref. 9).

In view of the above, and noting that the DPC methods for determining maximum allowable LOCA peaking and loss of flow accident (LOFA) DNB based operating limits and maximum allowable DNB and linear heat rate based RPS limits have been approved by the NRC, the acceptability of the proposed methodology hinges on the following major issues.

3.1 Operating Space AFD Limits

Since the proposed methodology represents a departure from currently accepted practice, any changes in limits relative to those obtained with the presently employed and approved RAOC methodology that represent a reduction in conservatism must be justified.

DPC has indicated that the proposed methodology will yield operating space AFD limits that are a few percent wider (less conservative) than the current RAOC

limits; this is due primarily to the use of explicit three-dimensional (3-D) power distributions as opposed to the synthesized 3-D power distributions on which RAOC is based. The increase in the available margin, and consequently the AFD operating space limits, is consistent with previous experience that supports a reduction in peaking when explicit 3-D power distributions are used as compared to synthesizing 3-D distributions from 1-D and 2-D calculations.

Under the proposed DPC methodology, if operating limits are too restrictive for normal operation, a set of limits can be defined that may still allow operation at full power. The resulting "base load" operation is typically used above 80 percent power and is similar to the widely used and accepted constant axial offset control (CAOC) approach. The xenon distributions used in setting the limits in this case are restricted to a relatively narrow operating band about a predicted AFD target.

It is therefore concluded that the DPC approach is acceptable with respect to AFD limits.

3.2 Conservatism of Power Distributions

In order to have confidence in the operating and RPS limits obtained by the proposed methodology, there must be demonstrated assurance that the power distributions resulting from the DPC approach are conservative with respect to those that might be reasonably expected to occur, and that they sufficiently span the AFD/rod-insertion power-level operating spaces to permit an accurate determination of limits.

DPC has determined through sensitivity studies that the power distributions employed in setting the operating and RPS limits are conservative. This is due in part to the severity of the xenon transients employed in the maneuvering analyses and conservative modelling assumptions. In addition, since the limits are based on the analyses of almost 3000 three-dimensional power distributions (resulting from a matrix of power level/rod position/inlet temperature/burnup and xenon distribution statepoints), DPC is confident that the operating limits can be determined accurately, and any extrapolation would

be minimal. A review of the statepoints (combinations of power level, rod insertion, etc.) and anticipated transients considered by DPC in generating bounding power distributions supports the conclusion that there is assurance that the power distributions assumed in the analyses of thermal limits are indeed conservative relative to the expected distributions, and this aspect of the DPC methodology is acceptable. It should be noted that the matrix of statepoints currently considered in the analysis may be modified as experience is accumulated. However, any reductions in the number of statepoints considered should be implemented only if there are no concomitant adverse effects (e.g., excessive interpolations required to set limits).

3.3 Uncertainties and Parameters in Margin and Monitoring Algorithms

The DPC methodology requires the determination of margins to linear heat rate and DNB thermal limits and the monitoring of the measured state to assure that operation is consistent with the DPC analyses performed to ensure that these limits will not be violated. Two linear heat rate related margins are determined - an operating limit based on LOCA considerations and an RPS limit that protects against centerline fuel melt. Similarly, two DNB related margins are also determined - an operating limit based on LOFA considerations and an RPS limit. In the core surveillance, precalculated factors based on the maneuvering analyses and the available margins are used to define an F_Q^{Max} and $F_{\Delta H}^{Max}$ which are then compared to measured values to determine whether the core is behaving as expected.

The equations used in the determination of the margins, including the uncertainties, were reviewed and found to be acceptable. The components of the margin equations used in the determination of linear heat rate and DNB are justified, and the values of the uncertainties applied have been previously reviewed and approved by the NRC.

Since only steady-state power distributions can be measured with reasonable accuracy, changes in the margins to limits accompanying deviations from steady-state conditions must be determined on the basis of calculations. The measured values of F_Q^{Max} and $F_{\Delta H}^{Max}$ are therefore compared to maximum

allowable values that account for the minimum margins determined in the maneuvering analysis to ensure that the limits on the measured values will be met at the extremes of the AFD-power level operating limits. If the measured values of F_Q^{Max} or $F_{\Delta H}^{Max}$ exceed their respective limits, then the AFD-power level limits and the f(delta-I) function in the OPΔT trip function are adjusted and/or the power level is reduced. The trends in the margins to the limits are monitored from measurement-to-measurement, and the measurement frequency is increased or an additional penalty is included in the margins if increased peaking is expected. Monitoring in the case of base load operation is similar. This monitoring philosophy is similar to that currently employed in connection with RAOC. The factors and uncertainties (and related methodologies) applied in the comparisons to measurements are justified, and the DPC methodology is acceptable.

3.4 Evaluation Summary

Based on the review of the topical report and the additional information provided, and recognizing that the NRC has reviewed and approved the computer codes and some components of the proposed methodology (e.g., the generation and use of DNB MATP curves), it is concluded that the DPC analysis represents an acceptable approach for determining and monitoring core operating and RPS limits for the McGuire and Catawba Nuclear Stations. The proposed methodology, however, should be confirmed by continued calculation-to-measurement comparisons, and monitoring of trends or any loss of conservatism. While the application of the methodology to other four-loop, 193-assembly W PWRs is acceptable, the appropriate, plant specific reactor systems aspects must be considered and justified.

3.5 Restrictions

The following restrictions are imposed on the use of the Nuclear Design Methodology described in DPC-NE-2011:

(1) Application of this methodology is to be limited to the McGuire and Catawba nuclear power stations,

(2) Application to other Westinghouse 193-assembly plants would be acceptable provided that plant-specific differences be considered and justified,

(3) Application of this methodology is contingent upon NRC approval of the Reload Design Thermal-Hydraulic Methodology DPC-NE-2004 (presently under NRC review) using the VIPRE-01 code, and

(4) Calculation of power and xenon distributions are limited to the use of the EPRI-NODE-P and the PDQ-07 codes.

4.0 REFERENCES

1. Letter from H. B. Tucker Duke Power Company to USNRC, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," dated April 27, 1988.

2. Letter from H. B. Tucker Duke Power Company to USNRC, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors - Response to Request for Additional Information," dated March 28, 1989.

3. Letter from C. O. Thomas NRC, to H. B. Tucker Duke Power Company, dated March 13, 1985.

4. WCAP-10216-PA, "Relaxation of Constant Axial Offset Control, F(q) Surveillance Technical Specification," June 1983.

5. Letter from H. B. Tucker (DPC) to NRC, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors - Response to Request for Additional Information," March 28, 1989.

6. DPC-NE-2010A, "Duke Power Company McGuire Nuclear Station, Catawba Nuclear Station, Nuclear Physics Methodology for Reload Design," June 1985.

7. Letter from C. O. Thomas (NRC) to H. B. Tucker (DPC), March 13, 1985.

8. NFS-1001A, "Duke Power Company, Oconee Nuclear Station, Reload Design Methodology," April 1984.

9. DPC-NE-2004, "Duke Power Company, McGuire and Catawba Nuclear Stations, Core Thermal-Hydarulic Methodology Using VIPRE-01," January 1989.

APPENDIX D

Revision 1 NRC SER

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
October 1, 2002

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

SUBJECT: McGUIRE NUCLEAR STATION, UNITS 1AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3222 AND MB3223)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 208 to Facility Operating License NPF-9 and Amendment No. 189 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated October 7, 2001, as supplemented by letter dated August 7, 2002.

The amendments revise TS 5.6.5.a by adding a few parameter limits currently included in the Core Operating Limits Report. In addition to the license amendment request, you also submitted revisions to four previously approved topical reports for the Nuclear Regulatory Commission staff review and approval. The enclosed Safety Evaluation also addresses these topical reports.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 208 to NPF-9
2. Amendment No. 189 to NPF-17
3. Safety Evaluation

cc w/encls: See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated October 7, 2001, as supplemented by letter dated August 7, 2002, Duke Power Company, et al. (DPC, the licensee), submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS).

Revisions were proposed for TS 5.6.5.a, Item 1, to add the moderator temperature coefficient (MTC) 60 parts per million (ppm) surveillance limit. The specific value of the surveillance limit was previously relocated to the Core Operating Limits Report (COLR). A new item 12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," is also proposed to be added to TS 5.6.5.a.

The initial submittal, dated October 7, 2001, proposed to change the dates and revision numbers for three of the Nuclear Regulatory Commission (NRC) approved analytical methods previously listed in TS 5.6.5.b, as listed below. The changes would reflect later versions of these topical reports that were also submitted with the October 7, 2001, submittal for NRC review and approval. As required by TS 5.6.5.b, only those methods listed within the TS as having been reviewed and approved by the NRC, can be used to determine the subject core operating limits. The subject core operating limits are listed in TS 5.6.5.a and their values are located in the COLR. A revision to a fourth report, DPC-NE-1003, was also submitted for NRC review and approval.

- DPC-NE-2009, Revision 1, "Duke Power Company Westinghouse Fuel Transition Report," August 2001.
- DPC-NF-2010, Revision 1, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," August 2001.
- DPC-NE-2011, Revision 1, "Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," August 2001.
- DPC-NE-1003, Revision 1, "McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing," August 2001.

The licensee in its letter of October 7, 2001, stated that, once approved, the approved topical report revisions, except for DPC-1003, Revision 1, will be listed in Section 5.6.5.b of the McGuire TS, to replace their respective original versions, and that the approved version of DPC-NE-2011-P, Revision 1, will also be listed in the references for TS Bases 3.2.1 and 3.2.3 to replace the existing reference to the original version, DPC-NE-2011-P-A.

However, on July 10, 2002, the NRC issued amendments numbered 203 and 184 to the McGuire Unit 1 and 2 operating licenses that effectively relocated the topical report revision numbers and dates from the TS 5.6.5.b list of approved methodologies to the COLR. Amendments 203 and 184 were consistent with the NRC Technical Specification Task Force (TSTF) Standard TS Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR." Accordingly, since this portion of its request is no longer needed in view of amendments 203 and 184, the licensee's letter dated August 7, 2002, eliminated the requests to change TS 5.6.5.b and proposed revisions to BASES 3.2.1 and 3.2.3 to make its submittal consistent with the implementation of amendments 203 and 184 at the McGuire Nuclear Station. Nonetheless, this Safety Evaluation sets forth the NRC staff's evaluation of the licensee's proposed changes to the topical reports listed above.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 (c)(2)(ii)(B), Criterion 2, specifies that a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier must be included in the TS limiting conditions for operation (LCO). Accordingly, the reactor operating parameters, which are the initial conditions for the safety analyses of the design basis transients and accidents, are included in the TS LCOs.

Since many parameter limits, such as core physics parameters, generally change with each reload core, licensees previously needed to request TS amendments to update these parameters for each refueling cycle. NRC Generic Letter (GL) 88-16 (Ref. 4) provides guidance for relocating the values of the cycle-specific core operating parameter limits from TS to the COLR, thus eliminating unnecessary burden on the licensees and the NRC to update these limits in the TS for each fuel cycle. The guidance includes adding the COLR in the TS administrative reporting requirement that also specifies (1) the cycle-specific parameters included in the COLR, and (2) the analytical methods that the NRC has previously reviewed and approved to be used to determine the core operating parameters limits.

The McGuire TS 5.6.5, "Core Operating Limits Report (COLR)," conforms to GL 88-16 guidance. TS 5.6.5.a lists a set of parameters, including the reference to the actual TS number for each specified parameter. TS 5.6.5.b specifies the topical reports that are used for the determination of the core operating limits.

The proposed TS changes in this license amendment request are to revise the parameters listed in TS 5.6.5.a. These revisions are based on the guidance of GL 88-16.

3.0 STAFF EVALUATION

In this section, the staff will discuss the review of the revised versions of the four previously approved topical reports submitted for staff review, and the proposed TS changes.

3.1 Topical Reports Revisions

The licensee requested the NRC to review revisions to four topical reports that were previously approved and listed in TS 5.6.5.b as the approved methodologies used for the determination of the parameter limits in the COLR. Since the staff has reviewed and approved the original versions of these topical reports, the staff review of these revised versions concentrated on the revisions made to the approved reports.

3.1.1 DPC-NE-2009, Revision 1

Topical report, DPC-NE-2009-P-A, (Ref. 5), provides general information about the Robust Fuel Assembly (RFA) design and describes methodologies used for reload design analyses to support the licensing basis for use of RFAs in the McGuire and Catawba reload cores. These methodologies include fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The NRC approved the report in September 1999.

Revision 1 of DPC-NE-2009, as amended by the August 7, 2002, letter (Ref. 2), consists of the following minor changes to its Chapter 6, "UFSAR Accident Analyses."

(A) Update of the reference list in Section 6.7 as follows:

- Update reference 6-25, WCAP-10054-P-A Addendum 2, to Revision 1, dated July 1997.
- Correct reference 6-35, WCAP-8354, with proprietary topical report number, and designate the second report as a non-proprietary report.
- Add reference 6-39, Westinghouse letter NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," dated July 15, 1999 (Ref. 6).

(B) Addition of a paragraph to Section 6.5.1, "Small Break LOCA," to explain that the Westinghouse small break LOCA NOTRUMP Evaluation Model includes the error corrections and model enhancements described in a few Westinghouse annual notifications required by 10 CFR 50.46, including the 1998 annual notification referenced in Reference 39.

The first two changes in the reference list are editorial and merely provide the latest version of the approved topical report or identify the proprietary and non-proprietary versions of a topical report. Reference 6-39, Westinghouse letter NSD-NRC-99-5839, is the annual notification of the changes to the LOCA evaluation models during 1998. This notification documented the following error corrections or model enhancements to the NOTRUMP small break LOCA Evaluation Model:

- A programming error correction on the SBLOCTA rod-to-rod radiation model, that is not modeled in licensing basis analyses and therefore, has no impact on the small break LOCA results.

- A logic simplification to the NOTRUMP droplet fall model that produces insignificant differences in results.

- A change in the reactor coolant pump heat in NOTRUMP that is not used in the evaluation model and therefore, has no impact on the small break LOCA results.

- A modification of NOTRUMP steam generator tube condensation heat transfer logic for a foreign plant that does not affect standard Westinghouse Pressurized Water Reactor calculations.

- An extension of reactor coolant conditions to allow for the NOTRUMP point kinetics calculations to be performed for cases that experience core uncovery conditions prior to reactor trip. For typical small break LOCA analyses, the reactor trips long before any threat of core uncovery and therefore, the change has no impact on peak cladding temperature calculations.

- A programming change in SBLOCTA code to allow for modeling of variable length blankets on either ends of the rod that involves no changes to the thermal-hydraulic fuel rod model, nor the solution technique.

Since the changes documented in the Westinghouse annual notice have insignificant impact on the small break LOCA analyses, the staff concludes the addition of Reference 6-39 is acceptable. Therefore, Revision 1 of DPC-NE-2009-P-A, as modified in the August 7, 2002, letter, is acceptable.

3.1.2 DPC-NF-2010, Revision 1

Topical Report DPC-NF-2010, (Ref. 7), describes DPC's Nuclear Design Methodology for McGuire and Catawba Nuclear Stations. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology that DPC intends to use to perform design calculations and to provide operational support, and the development of statistical factors.

Revision 1 of DPC-NF-2010, updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as shutdown margin, boron and control rod worth, axial and radial peaking factors, and cycle length, as well as numerous editorial changes.

During the review, the staff also identified a few discrepancies associated with administrative changes. In response to the staff's request for additional information (Ref. 2), the licensee provided further changes to Revision 1 of the topical report. These modifications include clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NF-2010 and the responses to the requests for additional information pertaining to these changes. The staff has concluded

that the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report and that there are no unreviewed methodology or regulatory issues. Therefore, the staff finds the changes to be acceptable.

3.1.3 DPC-NE-2011, Revision 1

Topical Report DPC-NF-2011, (Ref. 9), describes the methodology for performing a maneuvering analysis for four-loop plants, such as the McGuire and Catawba Nuclear Stations. The licensee has developed this methodology as an alternate to the existing Relaxed Axial Offset Control (RAOC) Methodology. The licensee pointed out that this maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of the licensee's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower delta-temperature and over-temperature delta-temperature reactor protection system (RPS) trip functions.

Revision 1 of DPC-NE-2011, updates the report to include editorial changes, and to permit the use of certain methods approved subsequent to the implementation of the original version, such as the CASMO-3/SIMULATE-3P methodology (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as power peaking factors, axial and radial power distributions, and cycle length, as well as numerous editorial changes.

In response to the NRC staff's request for additional information (Ref. 2), the licensee provided additional information regarding cycle depletion times to clarify issues associated with power peaking versus burnup as a function of cycle time. The licensee's amendment request also included clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NE-2011-A and the responses to the requests for additional information pertaining to the requested changes. Since the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report, the staff finds the changes to be acceptable.

3.1.4 DPC-NE-1003, Revision 1

Topical Report DPC-NE-1003 (Ref. 10), describes the measurement procedure used to determine the inferred bank worth and the calculation procedures used to develop the rod swap correction factor that accounts for the effect of a test bank on the partial integral worth of the reference bank. The NRC approved the report in May 1987 (Ref. 11) for rod worth measurement of reload cores for McGuire and Catawba Stations, Units 1 and 2.

Revision 1 of DPC-NE-1003 updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect the revision of the rod swap measurement procedures, and various editorial changes. In response to staff questions, the licensee, in its letter of August 7, 2002, provided the current version of the control rod worth measurement rod swap procedures, PT/0/A/4150/11A, dated January 19, 1996. The staff review of this current control rod worth measurement procedure has found it to be acceptable. The licensee, in the August 7, 2002, letter also modified the equation in Section 3 of the topical report for the calculation of the inferred rod bank worth from the

measured reference bank worth and bank height. This change is consistent with the equation described in step 12.12.5 of the current measurement procedures of January 19, 1996. Therefore, Revision 1 of DPC-NE-1003, as modified in the August 7, 2002, letter, is acceptable.

3.2 Proposed TS Changes

This section addresses the staff's evaluation of the proposed changes to TS 5.6.5.a regarding the cycle-specific operating parameters specified in the COLR. The staff review of these TS changes are based on the guidance of GL 88-16.

TS 5.6.5.a provides a list of core operating limits that are established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The values of the limits are located in the COLR. For McGuire Nuclear Station, Units 1 and 2, the licensee proposed to revise the list by:

(1) adding "60 ppm" to Item 5.6.5.a.1 regarding the moderator temperature coefficient (MTC) surveillance limit for Specification 3.1.3, and

(2) adding Item 5.6.5.a.12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2."

These changes are evaluated below.

3.2.1 MTC 60 ppm Surveillance Limit

McGuire TS LCO 3.1.3 specifies that the MTC be maintained within the LCO limits, which are based on the safety analysis assumptions. For verification that these LCO limits are met, the Surveillance Requirements of TS 3.1.3 also place surveillance limits for conducting the end of cycle MTC measurement at boron concentrations of 300 ppm and 60 ppm. The LCO limits and the 300 ppm and 60 ppm surveillance limits are specified in the COLR. However, TS Item 5.6.5.a.1 operating limits does not currently identify the 60-ppm surveillance limit.

The proposed change to the McGuire TS would add the 60 ppm surveillance limit in Item 5.6.5.a.1. The new TS would read "Moderator Temperature Coefficients BOL and EOL limits and 60 ppm and 300 ppm surveillance limit for Specification 3.1.3." The NRC approved incorporating the 60-ppm surveillance limits into the COLR during the Improved Technical Specifications conversion in 1998 (Ref. 12 and 13); however, reference to this surveillance was not included in TS Item 5.6.5.a.1 at that time. The proposed TS change to include the 60 ppm surveillance limit in TS Item 5.6.5.a.1 provides consistency with previously approved requirements and, therefore, it is acceptable.

3.2.2 Relocation of Hot Channel Factors Surveillance Penalty Factors to COLR

Surveillance Requirements in TS 3.2.1 and 3.2.2, respectively, require that the heat flux hot channel factor, F_q (x,y,z), and the enthalpy rise hot channel factor, $F_{\Delta h}$ (x,y), be measured every 31 effective full power days (EFPD) during equilibrium conditions using the incore detector system to verify they are within the respective limits. To address the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, penalty factors are applied to these hot channel factors if their margins to the respective limits have decreased since the previous surveillance. These margin-decrease penalty factors are

calculated by projecting the limiting hot channel factors over the 31 EFPD surveillance intervals with the maximum changes at the limiting core location, and are based on reload core design. In Section 8, "Improved Technical Specification Changes," of DPC-NE-2009, the licensee proposed to replace the penalty factors with tables of penalty value as a function of burnup in the COLR to facilitate cycle-specific updates. TS Item 5.6.5.b.14 lists topical report DPC-NE-2009-P-A that includes (in response to a staff question during the review of DPC-NE-2009) the approved methodology used to calculate these burnup-dependent penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin decrease penalty factors in the COLR acceptable, as stated in the staff's Safety Evaluation supporting license Amendment Nos. 188 and 169, respectively, for McGuire Nuclear Station, Units 1 and 2 (Ref. 14).

The proposed changes to the McGuire TS would add Item 5.6.5.a.12 that reads: "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2." The addition of TS Item 5.6.5.a.12 would make it consistent with the previous staff approval of including these surveillance penalty factors in the COLR and, therefore, this proposed change is acceptable.

4.0 SUMMARY

The staff has reviewed the revisions to four previously approved topical reports described in Section 1.0 of this Safety Evaluation, and the proposed changes to McGuire Nuclear Station, Units 1 and 2, TS 5.6.5.a related to the COLR. Based on our evaluation, described in Section 3 of this Safety Evaluation, the staff concludes that the these topical report revisions, as amended by the August 7, 2002, letter, and the TS changes are acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change recordkeeping, reporting, or administrative procedure requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (67FR 54680). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to BASES 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

2. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369 and 370; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413 and 414; Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343 and MB3344) and License Amendment Request Supplement," August 7, 2002.

3. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report - Oconee, McGuire, and Catawba Technical Specifications 5.6.5," December 20, 2001.

4. Letter from Dennis Crutchfield, USNRC, to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

5. DPC-NE-2009-P-A, "Duke Power Company Westinghouse Fuel Transition Report," December 1999.

6. Letter from J. S. Galembush, Westinghouse Electric Company, to US Nuclear Regulatory Commission, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," NSD-NRC-99-5839, July 15, 1999.

7. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985..

8. DPC-NE-1004A, Revision 1, "Nuclear Design Methodology Using CASMO-3/ SIMULATE-3P," SER dated April 26, 1997.

9. DPC-NE-2011, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," March 1990.

10. DPC-NE-1003, "Rod Swap Methodology Report for Startup Physics Testing," December 1986.

11. Letter from Darl Hood, USNRC, to H. B. Tucker, Duke Power Company, "Rod Swap Methodology Report for Startup Physics Testing, McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 62981, 62982, 62983, 62984)," May 22, 1987.

12. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2, (TAC Nos. M98964 and M98965)," September 30, 1998.

13. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M95298 and M95299)," September 30, 1998.

14. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "McGuire Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2411 and MA2412)," September 22, 1999.

15. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Catawba Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2359 and MA2361)," September 22, 1999.

16. Letter from Robert F. Martin, USNRC, to David L. Rehn, Catawba Site, Duke Power Company, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)," March 25, 1994.

Principal Contributor: Y. Hsii
A. Attard

Date: October 1, 2002

EC050

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

RECEIVED
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REGULATORY COMPLIANCE

October 1, 2002

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3343 AND MB3344)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 202 to Facility Operating License NPF-35 and Amendment No.195 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated October 7, 2001, as supplemented by letter dated August 7, 2002.

The amendments revise TS 5.6.5.a by adding a few parameter limits currently included in the Core Operating Limits Report. In addition to the license amendment request, you also submitted revisions to four previously approved topical reports for the Nuclear Regulatory Commission staff review and approval. The enclosed Safety Evaluation also address these topical reports.

A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Chandu P. Patel

Chandu P. Patel, Project Manager, Section 1
Project Directorate II /RA/
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 202 to NPF-35
2. Amendment No. 195 to NPF-52
3. Safety Evaluation

cc w/encls: See next page

Catawba Nuclear Station

cc:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated October 7, 2001, as supplemented by letter dated August 7, 2002, Duke Energy Corporation, et al. (DEC, the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS).

Revisions were proposed for TS 5.6.5.a, Item 1, to add the moderator temperature coefficient (MTC) 60 parts per million (ppm) surveillance limit. The specific value of the surveillance limit was previously relocated to the Core Operating Limits Report (COLR). Two new items were also proposed to be added to TS 5.6.5.a. These two items are (1) Item 12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," and (2) Item 13, "Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9.2."

The initial submittal, dated October 7, 2001, proposed to change the dates and revision numbers for three of the Nuclear Regulatory Commission (NRC) approved analytical methods previously listed in TS 5.6.5.b, as listed below. The changes would reflect later versions of these topical reports that were also submitted with the October 7, 2001, submittal for NRC review and approval. As required by TS 5.6.5.b, only those methods listed within the TS as having been reviewed and approved by the NRC, can be used to determine the subject core operating limits. The subject core operating limits are listed in TS 5.6.5.a and their values are located in the COLR. A revision to a fourth report, DPC-NE-1003, was also submitted for NRC review and approval.

- DPC-NE-2009, Revision 1, "Duke Power Company Westinghouse Fuel Transition Report," August 2001.
- DPC-NF-2010, Revision 1, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," August 2001.
- DPC-NE-2011, Revision 1, "Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," August 2001.

- DPC-NE-1003, Revision 1, "McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing," August 2001.

The licensee in its letter of October 7, 2001, stated that, once approved, the approved topical report revisions, except for DPC-1003, Revision 1, will be listed in Section 5.6.5.b of the Catawba TS, to replace their respective original versions, and that the approved version of DPC-NE-2011-P, Revision 1, will also be listed in the references for TS Bases 3.2.1 and 3.2.3 to replace the existing reference to the original version, DPC-NE-2011-P-A.

However, on July 2, 2002, the NRC issued amendments numbered 199 and 192 to the Catawba Unit 1 and 2 operating licenses that effectively relocated the topical report revision numbers and dates from the TS 5.6.5.b list of approved methodologies to the COLR. Amendments 199 and 192 were consistent with the NRC Technical Specification Task Force (TSTF) Standard TS Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR." Accordingly, since this portion of its request is no longer needed in view of amendments 199 and 192, the licensee's letter dated August 7, 2002, eliminated the requests to change TS 5.6.5.b and proposed revisions to BASES 3.2.1 and 3.2.3 to make its submittal consistent with the implementation of amendments 199 and 192 at the Catawba Nuclear Station. Nonetheless, this Safety Evaluation sets forth the NRC staff's evaluation of the licensee's proposed changes to the topical reports listed above.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulation* (10 CFR) Section 50.36 (c)(2)(ii)(B), Criterion 2 specifies that a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier must be included in the TS limiting conditions for operation (LCO). Accordingly, the reactor operating parameters, which are the initial conditions for the safety analyses of the design basis transients and accidents, are included in the TS LCO.

Since many parameters limits, such as core physics parameters, generally change with each reload core, licensees need to request TS amendments to update these parameters for each refueling cycle. NRC Generic Letter (GL) 88-16 (Ref. 4) provides guidance for relocating the values of the cycle-specific core operating parameter limits from TS to the COLR, and thus eliminates the unnecessary burden on the licensees and the NRC to update these limits in the TS each fuel cycle. The guidance includes adding the COLR in the TS administrative reporting requirement that also specifies (1) the cycle-specific parameters included in the COLR, and (2) the analytical methods that the NRC has previously reviewed and approved to be used to determine the core operating parameters limits.

The Catawba TS 5.6.5, "Core Operating Limits Report (COLR)," conforms to the GL 88-16 guidance. TS 5.6.5.a lists a set of parameters, including the reference to the actual TS number for each specified parameter. TS 5.6.5.b specifies the topical reports that are used for the determination of the core operating limits.

The proposed TS changes in this license amendment request are to revise the parameters listed in TS 5.6.5.a. These revisions are based on the guidance of GL 88-16.

3.0 STAFF EVALUATION

In this section, the staff will discuss the review of the revised versions of the four previously approved topical reports submitted for staff review, and the proposed TS changes.

3.1 Topical Reports Revisions

The licensee requested the NRC to review revisions of four topical reports that were previously approved and listed in TS 5.6.5.b as the approved methodologies used for the determination of the parameter limits in the COLR. Since the staff has reviewed and approved the original versions of these topical reports, the staff review of these revised versions will concentrate on the revisions made to the approved reports.

3.1.1 DPC-NE-2009, Revision 1

Topical report, DPC-NE-2009-P-A, (Ref. 5), provides general information about the Robust Fuel Assembly (RFA) design and describes methodologies used for reload design analyses to support the licensing basis for use of the RFA design in the McGuire and Catawba reload cores. These methodologies include fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The NRC approved the report in September 1999.

Revision 1 of DPC-NE-2009-A, as amended by the August 7, 2002, letter (Ref. 2), consists of the following minor changes to Chapter 6, "UFSAR Accident Analyses:"

(A) Update of the reference list in Section 6.7 as follows:

- Update reference 6-25, WCAP-10054-P-A Addendum 2, to Revision 1, dated July 1997.
- Correct reference 6-35, WCAP-8354, with proprietary topical report number, and designate the second report as a non-proprietary report.
- Add reference 6-39 a Westinghouse letter NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," dated July 15, 1999 (Ref. 6).

(B) Addition of a paragraph to Section 6.5.1, "Small Break LOCA," to explain that the Westinghouse small break LOCA NOTRUMP Evaluation Model includes the error corrections and model enhancements described in a few Westinghouse annual notifications required by 10 CFR 50.46, including the 1998 annual notification referenced in Reference 39.

The first two changes in the reference list are editorial and merely provide the latest version of the approved topical report or identify the proprietary and non-proprietary versions of a topical report. Reference 6-39, the Westinghouse letter NSD-NRC-99-5839, is the annual notification of the changes to the LOCA evaluation models during 1998. This notification documented the following error corrections or model enhancements to the NOTRUMP small break LOCA Evaluation Model:

- A programming error correction on the SBLOCTA rod-to-rod radiation model that is not modeled in licensing basis analyses and therefore, has no impact on the small break LOCA results.

- A logic simplification to the NOTRUMP droplet fall model that produces insignificant differences in results.

- A change in the reactor coolant pump heat in NOTRUMP that is not used in the evaluation model and therefore, has no impact on the small break LOCA results.

- A modification of NOTRUMP steam generator tube condensation heat transfer logic to a foreign plant that does not affect standard Westinghouse Pressurized Water Reactor calculations.

- An extension of reactor coolant conditions to allow for the NOTRUMP point kinetics calculations to be performed for cases that experience core uncovery conditions prior to reactor trip. For typical small break LOCA analyses, the reactor trips long before any threat of core uncovery and therefore, the change has no impact on peak cladding temperature calculations.

- A programming change in SBLOCTA code to allow for modeling of variable length blankets on either ends of the rod that involves no changes to the thermal-hydraulic fuel rod model, nor the solution technique.

Since the changes documented in the Westinghouse annual notice have insignificant impact on the small break LOCA analyses, the staff concludes the addition of Reference 6-39 is acceptable. Therefore, Revision 1 of DPC-NE-2009-P-A, as modified in the August 7, 2002, letter, is acceptable.

3.1.2 DPC-NF-2010A, Revision 1

Topical Report DPC-NF-2010A, (Ref. 7), describes Duke Power Company's Nuclear Design Methodology for McGuire and Catawba Nuclear Stations. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology the licensee intends to use to perform design calculations and to provide operational support, and the development of statistical factors.

Revision 1 of DPC-NF-2010A, updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as shutdown margin, boron and control rod worth, axial and radial peaking factors, and cycle length, as well as numerous editorial changes.

During the review, the staff also identified a few discrepancies associated with administrative changes. In response to the staff's request for additional information (Ref. 2), the licensee provided further changes to Revision 1 of the Topical report. These modifications include clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NF-2010A and the responses to the requests for additional information pertaining to these changes. The staff has concluded

that the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report and that there are no unreviewed methodology or regulatory issues. Therefore, the staff finds the changes acceptable.

3.1.3 DPC-NE-2011, Revision 1

Topical Report DPC-NE-2011, (Ref. 9), describes the methodology for performing a maneuvering analysis for four-loop plants, such as McGuire and Catawba Nuclear Station. The licensee has developed this methodology as an alternate to the existing Relaxed Axial Offset Control Methodology. The licensee pointed out that this maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of the licensee's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower delta-temperature and over-temperature delta-temperature reactor protection system trip functions.

Revision 1 of DPC-NE-2011, updates the report to include editorial changes, and to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P methodology (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as power peaking factors, axial and radial power distributions, and cycle length, as well as numerous editorial changes.

In response to the NRC staff's request for additional information (Ref. 2), the licensee provided additional information to the staff regarding cycle depletion times to clarify issues associated with power peaking versus burnup as a function of cycle time. The licensee's amendment request also included clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NE-2011-A and the responses to the requests for additional information pertaining to the requested changes. Since the changes to this topical report consists mostly of administrative changes and clarifications to the original NRC approved topical report, the staff find the changes acceptable.

3.1.4 DPC-NE-1003, Revision 1

Topical Report DPC-NE-1003 (Ref. 10) describes the measurement procedure used to determine the inferred bank worth and the calculation procedures used to develop the rod swap correction factor that accounts for the effect of a test bank on the partial integral worth of the reference bank. The NRC approved the report in May 1987 (Ref. 11) for rod worth measurement of reload cores for McGuire and Catawba Stations, Units 1 and 2.

Revision 1 of DPC-NE-1003 updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect the revision of the rod swap measurement procedures, and various editorial changes. In response to staff questions, the licensee, in its letter of August 7, 2002, provided the current version of the control rod worth measurement rod swap procedures, PT/0/A/4150/11A, dated January 19, 1996. The staff review of this current control rod worth measurement procedure has found it acceptable. The licensee in the August 7, 2002, letter also modified the equation in Section 3

of the topical report for the calculation of the inferred rod bank worth from the measured reference bank worth and bank height. This change is consistent with the equation described in step 12.12.5 of the current measurement procedures of January 19, 1996. Therefore, Revision 1 of DPC-NE-1003, as modified in the August 7, 2002, letter, is acceptable.

3.2 Proposed TS Changes

This section addresses the staff's evaluation of the proposed changes to TS 5.6.5.a regarding the cycle-specific operating parameters specified in the COLR. The staff review of these TS changes are based on the guidance of GL 88-16.

TS 5.6.5.a provides a list of core operating limits that are established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The valves of the limits are in the COLR. For Catawba Units 1 and 2, the licensee proposed to revise the list by:

(1) adding "60 ppm" to Item 5.6.5.a.1 regarding the moderator temperature coefficient (MTC) surveillance limit for Specification 3.1.3,

(2) adding Item 5.6.5.a.12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," and

(3) adding Item 5.6.5.a.13, "Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9.2."

These changes are evaluated below.

3.2.1 MTC 60 ppm Surveillance Limit

Catawba TS LCO 3.1.3 specifies that the MTC be maintained within the LCO limits, which are based on the safety analysis assumptions. For verification that these LCO limits are met, the Surveillance Requirements of TS 3.1.3 also places surveillance limits for conducting the end of cycle MTC measurement at 300 ppm and 60 ppm boron concentration. The LCO limits and the 300-ppm and 60-ppm surveillance limits are specified in the COLR. However, TS Item 5.6.5.a.1 operating limits does not currently identify the 60-ppm surveillance limit.

The proposed change to the Catawba TS would add the 60-ppm surveillance limit in Item 5.6.5.a.1. The new TS would read "Moderator Temperature Coefficients BOL and EOL limits and 60 ppm and 300 ppm surveillance limit for Specification 3.1.3." The NRC approved incorporating the 60-ppm surveillance limits into the COLR during the Improved Technical Specifications conversion in 1998 (Ref. 12 and 13); however, reference to this surveillance was not included in TS Item 5.6.5.a.1 at that time. The proposed TS change to include the 60-ppm surveillance limit in TS Item 5.6.5.a.1 provides consistency with previously approved requirements and, therefore, it is acceptable.

3.2.2 Relocation of Hot Channel Factors Surveillance Penalty Factors to COLR

Surveillance Requirements in TS 3.2.1 and 3.2.2, respectively, require that the heat flux hot channel factor, F_q (x,y,z), and the enthalpy rise hot channel factor, $F_{\Delta h}$ (x,y), be measured every 31 effective full power days (EFPD) during equilibrium conditions using the incore detector

system to verify they are within the respective limits. To address the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, penalty factors are applied to these hot channel factors if their margins to the respective limits have decreased since the previous surveillance. These margin-decrease penalty factors are calculated by projecting the limiting hot channel factors over the 31 EFPD surveillance intervals with the maximum changes at the limiting core location, and are based on reload core design. In Section 8, "Improved Technical Specification Changes," of DPC-NE-2009, the licensee proposed to replace the penalty factors with tables of penalty value as functions of burnup in the COLR to facilitate cycle-specific updates. TS Item 5.6.5.b.14 lists topical report DPC-NE-2009-P-A that includes (in response to a staff question during the review of DPC-NE-2009) the approved methodology used to calculate these burnup-dependent penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin decrease penalty factors in the COLR acceptable as stated in the staff's safety evaluation supporting license amendment Nos. 180 and 172, respectively for Catawba Units 1 and 2 (Ref. 15).

The proposed changes to the Catawba TS would add Item 5.6.5.a.12, that reads: "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2." The addition of TS Item 5.6.5.a.12 would make it consistent with the previous staff approval of including these surveillance penalty factors in the COLR and, therefore, this proposed change is acceptable.

3.2.3 Reactor Makeup Water Pumps Combined Flow Rates Limit

The relocation of the reactor makeup water pumps combined flow rates limit for the boron dilution mitigation system from Catawba TS 3.3.9 and 3.9.2 to the COLR was approved by the NRC as described in a letter dated March 25, 1994 (Ref. 16). The reactor makeup water pumps flow rate limit is included in the Catawba COLR.

The proposed changes to the Catawba TS would add Item 5.6.5.a.13, "Reactor makeup water pumps combined flow rates limit for Specification 3.3.9 and 3.9.2," to TS 5.6.5.a. The addition of this item would make the TS 5.6.5.a list consistent with the core operating limits included in the Catawba COLR and is therefore, acceptable.

4.0 SUMMARY

The staff has reviewed the revisions of four previously approved topical reports described in Section 1.0 of this Safety Evaluation, and the proposed changes to Catawba Nuclear Station, Units 1 and 2, TS 5.6.5.a related to the COLR. Based on our evaluation described in Section 3 of this Safety Evaluation, the staff concludes that the these topical report revisions, as amended by the August 7, 2002, letter, and the TS changes are acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding [67 FR 54680]. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to BASES 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

2. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369 and 370; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413 and 414; Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343 and MB3344) and License Amendment Request Supplement," August 7, 2002.

3. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report - Oconee, McGuire, and Catawba Technical Specifications 5.6.5," December 20, 2001.

4. Letter from Dennis Crutchfield, USNRC, to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

5. DPC-NE-2009-P-A, "Duke Power Company Westinghouse Fuel Transition Report," December 1999.

6. Letter from J. S. Galembush, Westinghouse Electric Company, to US Nuclear Regulatory Commission, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," NSD-NRC-99-5839, July 15, 1999.

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