DPC-NE-2009-A

Section E

Topical Report

(includes updated and revised pages listed in Sections D, F, G, and H)

Duke Power Company Westinghouse Fuel Transition Report

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1.0 INTRODUCTION

Duke Power Company is currently using Framatome Cogema Fuels (FCF) Mark-BW fuel assemblies in the McGuire and Catawba reactors. Duke Power will transition to the 17x17 Westinghouse 0.374 Robust Fuel Assembly (RFA) design described in Chapter 2 of this report. This topical report presents the information required to support the licensing basis for the use of the RFA design in McGuire and Catawba reload cores.

This report describes the core design, fuel rod design, and thermal-hydraulic analyses that are performed to show that all licensing criteria are met for each reload core. This report also discusses the UFSAR Chapter 15 transient and accident analyses methodology that is applicable to each reload design. Previously approved methodologies used by Duke Power Company to perform core design, thermal-hydraulic design, and UFSAR Chapter 15 Non-LOCA analyses for the Mark-BW fuel will be used to analyze the RFA design with the revisions described in Chapters 3, 5, and 6, respectively.

Chapter 4 describes the fuel rod design analysis methodology that will be used to analyze the RFA design. Although the fuel rod analysis methodology is new for Duke Power, the methods are essentially identical to the NRC-approved Westinghouse methods. The Westinghouse LOCA analysis methodology is described in Section 6.5. Section 6.6 presents an improved methodology that will be used to perform the nuclear analysis portion of the rod ejection accident (REA) analysis for McGuire and Catawba. The new methodology is based on the SIMULATE-3K computer code.

Chapter 7 discusses the licensing and analysis approach Duke Power will use for reconstitution of the RFA design. Chapter 8 describes the Technical Specification changes that will be made due to the transition to the RFA design and the analysis methodology described in this report.

2.0 FUEL DESIGN

Duke Power is transitioning to the Westinghouse 17x17 0.374 robust fuel assembly design for the McGuire and Catawba reactors. For the remainder of this report the fuel design will be referred to as simply the RFA design The RFA design is based on the VANTAGE + fuel assembly design, licensed by the NRC in Reference 2-1. The RFA design used at McGuire and Catawba will include the following features initially licensed with the VANTAGE + fuel design:

- ZIRLO™ clad fuel rods,
- ZIRLO™ guide thimbles, instrumentation tubes and mid-grids (both structural and Intermediate Flow Mixing (IFM) grids),
- 0.374 inch fuel rod OD,
- Zirconium diboride Integral Fuel Burnable Absorbers (IFBAs),
- Mid-enriched annular axial blanket pellets,
- High burnup fuel skeleton, and
- Debris Filter Bottom Nozzle (DFBN).

In addition to the VANTAGE + fuel design features listed above, the RFA design used at McGuire and Catawba will incorporate the following features that were licensed using the Fuel Criteria Evaluation Process (Reference 2-2) via Reference 2-3:

- Increased guide thimble and instrumentation tube OD (0.482 inch),
- Modified Low Pressure Drop (MLPD) structural mid-grids, and
- Modified Intermediate Flow Mixing (MIFM) grids.

The RFA design used at McGuire and Catawba will include the following additional features to help mitigate debris failures:

- Pre-oxide coating on the bottom of the fuel rods and
- Protective bottom grid with longer fuel rod end-plugs.

The RFA design used at McGuire and Catawba will include the following feature to help mitigate Incomplete Rod Insertion (IRI):

- fuel rods positioned on the bottom nozzle

The three features listed above will be evaluated using the 10CFR50.59 process

One new feature that will be added to the McGuire and Catawba RFA design is a Quick Release Top Nozzle (QRTN). This top nozzle design is similar to the Reconstitutable Top Nozzle (RTN) design, but has been modified for easier removal. This design change will be licensed by Westinghouse using the Fuel Criteria Evaluation Process (Reference 2-2) and notification will be made to the NRC. Westinghouse sent notification per Reference 2-2 to the NRC in Reference 2-6 confirming batch implementation of the QRTN at McGuire and Catawba. 2

The Westinghouse RFA is designed to be mechanically and hydraulically compatible with the FCF Mark-BW fuel (Reference 2-4) that is currently used at McGuire and Catawba. The basic design parameters of the RFA are compared to those of the Mark-BW fuel assembly in Table 2-1.

The IFM grids are non-structural members whose primary function is to promote mid-span flow mixing. Therefore, the design bases for the IFM grids are to avoid cladding wear and interactive damage with grids of the neighboring fuel assemblies during fuel handling. Westinghouse fuel with IFM grids has been flow tested both adjacent to another assembly with IFM grids and adjacent to an assembly without IFM grids. There was no indication of adverse fretting wear of the fuel rods by the standard structural or IFM grids (Reference 2-5). No adverse fretting wear is expected in transition cores with the Westinghouse RFA design and Mark-BW fuel since the Mark-BW fuel is very similar to Westinghouse fuel assembly designs without IFM grids

2.1 References

2-1 S. L. Davidson & T. L. Ryan, "Vantage+ Fuel Assembly Reference Core Report", WCAP-12610-P-A, April 1995.

2-2 S. L. Davidson (Ed.), "Westinghouse Fuel Criteria Evaluation Process", WCAP-12488-P-A, October 1994.

2-3 NSD-NRC-97-5189, Letter from N. J. Liparulo (Westinghouse) to J. E. Lyons (USNRC), "Transmittal of Response to NRC Request for Information on Wolf Creek Fuel Design Modifications", June 30, 1997.

2-4 "Mark-BW Mechanical Design Report", BAW-10172P-A, December 1989.

2-5 S. L. Davidson (Ed.), "Reference Core Report Vantage 5 Fuel Assembly", WCAP-10444-P-A, September 1985.

2-6 LTR-NRC-02-2, Letter from Henry A. Sepp (Westinghouse) to J.S Wermeil (USNRC), "Fuel Criterion Evaluation Process (FCEP) Notification of Quick Release Top Nozzle (QRTN) Design", January 15, 2002. 2

Table 2-1

Comparison of Robust Fuel Assembly and Mark-BW Fuel Assembly Design Parameters

	17x17 Robust Fuel Assembly Design	17x17 Mark-BW Fuel Assembly Design
Fuel Assembly Length, in.		
Assembly Envelope, in.		
Fuel Rod Pitch, in		
Fuel Rod Material		
Fuel Rod Clad OD, in.		
Fuel Rod Clad Thickness, in.		
Fuel/Clad Gap, mils		
Fuel Pellet Diameter, in.		
Fuel Stack Height, in.		
Guide Thimble Material		
Outer Diameter of Guide Thimbles, in (upper part)		
Inner Diameter of Guide Thimbles, in. (upper part)		
Outer Diameter of Guide Thimbles, in. (lower part)		
Inner Diameter of Guide Thimbles, in (lower part)		
Outer Diameter of Instrument Guide Thimbles, in.		
Inner Diameter of Instrument Guide Thimbles, in.		
End Grid Material		
Intermediate Grid Material		
Intermediate Flow Mixing Grid Material		

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3.0 CORE DESIGN

3.1 Introduction

The nuclear characteristics of the Westinghouse RFA design and the Mark-BW fuel design are almost identical due to similar dimensional characteristics of the fuel pellet, fuel rod and cladding. As a result, the methods and core models used to perform transition and full core analyses of the Westinghouse RFA design are the same as those currently licensed and employed in reload design analyses for McGuire and Catawba.

3.2 Reload Design Methodology

The development of core models, core operational imbalance limits and the evaluation of key physics parameters used to confirm the acceptability of UFSAR Chapter 15 accidents will be performed in compliance with the approved methodology defined in References 3-1 through 3-4. Conceptual transition core designs using the Westinghouse RFA design have been evaluated and show that current reload limits remain bounding with respect to key physics parameters. In the event that one of the key parameters is exceeded, the evaluation process described in Reference 3-3 would be performed.

The introduction of the Westinghouse RFA design is not expected to change the magnitude of the nuclear uncertainty factors described in Reference 3-1. However, the use of zirconium di-boride Integral Fuel Burnable Absorbers (IFBA) is a fuel design change which is different from the burnable absorber types modeled in Duke's current benchmarking database. The NRC SER for Reference 3-1 requires Duke to re-benchmark the nuclear code package and assure that the nuclear uncertainties remain appropriate for significant changes in fuel design. While the introduction of the IFBA burnable absorber is not considered significant, the nuclear uncertainties in Reference 3-1 were re-evaluated and confirmed to be bounding.

Duke explicitly modeled Seqouyah Unit 2 Cycles 5, 6, and 7 and performed statistical analysis of the nuclear uncertainty factors as described in Reference 3-1. These cores were chosen because they are very similar to McGuire and Catawba and contained both IFBA and Wet Annular Burnable Absorber (WABA) fuel. The results of the statistical analysis are shown in Table 3-1 and show that the current licensed nuclear uncertainty factors bound those for the Westinghouse fuel with a combination of IFBA and/or WABA burnable absorbers. Boron concentrations, rod worths, and isothermal temperature coefficients were also predicted and found to agree well with the measured data. A 10CFR50.59 USQ evaluation has been performed to demonstrate that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties are applicable to the Westinghouse RFA design described in this report.

In all nuclear design analyses, both the Westinghouse RFA and the Mark-BW fuel are explicitly modeled in the transition cores. When establishing Operating and RPS limits (i.e. LOCA kw/ft, DNB, CFM, transient strain), the fuel specific limits or a conservative overlay of the limits are used.

The nuclear design related Technical Specification limits were reviewed for transition and full core reloads comprised of the Westinghouse RFA design. The only change required to the Technical Specifications is to replace the factor used to account for possible increases in $F_{\Delta H}$ and Fq between flux maps with a burnup dependent factor (see Chapter 8 for additional details).

In summary, the steady-state physics codes, methodology and nuclear uncertainty factors remain unchanged for the transition to the Westinghouse RFA design. The evaluation of conceptual core designs with the RFA design indicate that key physics parameters assumed in the UFSAR Chapter 15 accident analyses remain bounding. The introduction of the IFBA burnable poison design will require that the factor used to account for the possible increase in peaking over a 31 EFPD surveillance period be replaced by a burnup dependent factor (see Chapter 8).

3.3 References

3-1 "Design Methodology Using CASMO-3/SIMULATE-3P," DPC-NE-1004A (Revision 1), SER dated April 26, 1996.

3-2. "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," DPC-NE-2011PA, March 1990.

3-3 "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," DPC-NE-3001-PA, November 1991.

3-4 "Duke Power Company McGuire Nuclear Station, Catawba Nuclear Station Nuclear Physics Methodology," DPC-NF-2010A, June 1985.

Table 3-1

Nuclear Uncertainty Factors

(Statistically combined factors without Engineering Hot Channel Factor)

Parameter	Westinghouse Fuel with IFBA/WABA	DPC-NE-1004A
$F_{\Delta h}$	1.027	1.028
F_z	1.049	1.053
F_q	1.049	1.061

4.0 FUEL ROD ANALYSIS

This chapter describes Duke Power's fuel rod mechanical reload analysis methodology for Westinghouse fuel. The fuel rod analysis methodology discussed in this Chapter is essentially identical to Westinghouse's approved methodology. The analyses will be performed using the NRC approved Westinghouse fuel performance code, PAD, described in Section 4.1. Fuel rod mechanical analyses for Mark-BW fuel at McGuire and Catawba will continue to be performed using the NRC-approved methodology given in Reference 4-12.

The fuel rods are designed to meet the requirements of 10CFR50, Appendix A, "General Design Criteria" (Reference 4-1), specifically Criterion 10 "Reactor Design", which states: "The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation including the effects of anticipated operational occurrences."

To meet this requirement and the requirements of Section 4.2 of the Standard Review Plan (SRP) (Reference 4-2), Westinghouse has established specific fuel design criteria associated with Condition I and II operation (Reference 4-3). Section 4.2 of this report describes each of the fuel rod design criteria which are evaluated as required by SRP 4.2 for Condition I and II operation. A description of the fuel rod analysis methodology which is used to show that the design criteria are met each cycle is also provided.

Detailed fuel rod design analyses consider parameters such as the pellet/clad diametral gap, the size and density of the pellet, the gas plenum volume, and the helium prepressurization. Using the approved fuel performance models in PAD (Reference 4-4 and 4-14), the analyses also 2
consider effects such as fuel densification and swelling, cladding creep, cladding corrosion, fission gas release and other physical properties which vary with burnup. The integrity of the fuel rods is ensured by designing the rods and operating the core to prevent excessive fuel temperatures, excessive fuel rod internal gas pressures, and excessive cladding stresses and strains. This is achieved by verifying that the conservative design criteria described in Section 4.2 are satisfied during Condition I and II events over the life of the fuel.

The fuel rod analyses must consider the uncertainties associated with design models and variations in as-built dimensions. Due to the empirical basis of the performance models used in the design codes (e.g , fission gas release, clad creep, etc), there is variability in the data used for model validation. To have confidence that the extremes of the performance spectrum are covered, deviations from best estimate model projections must be accounted for Each model which has a significant effect on fuel rod performance includes uncertainty bands defined to bound 95 % of the data. These uncertainty bands are used to define conservative upper bound uncertainty levels in the model predictions. These uncertainty levels are considered in the fuel rod analyses, assuring that all fuel rods in a core will satisfy the design criteria

The fuel rod analyses also consider the variations in rod dimensions and fuel fabrication characteristics. Typically drawing tolerances which are assumed to represent at least a 2 sigma bound are used in fuel rod analyses Actual as-built measurements and bounding values based on measured standard deviations may be used for critical fuel parameters. The typical method for including model, rod dimension, and fuel characteristic uncertainties is by statistical convolution. 2

The fuel rod for the RFA design is identical to the fuel rod for the VANTAGE+ design, thus the licensed pin burnup for the Westinghouse RFA design is 60,000 MWd/mtU (Reference 4-3) Using the Westinghouse Fuel Criteria Evaluation Process (FCEP) (Reference 4-13), the burnup limit can be increased to 62,000 MWd/mtU for specific reload cores.

Fuel rod analyses or evaluations to verify that a generic analysis is applicable must be performed for each reload cycle. Typically, generic analyses are completed that are expected to envelope the operation of future fuel cycles. The generic fuel rod analyses are then shown to be valid for each reload cycle design. This chapter describes the generic fuel rod analysis methods. In most cases, the generic analyses are bounding for each fuel cycle design and no new analyses are required. Cycle specific fuel rod analyses may be performed to obtain additional margin

4.1 Computer Code

The PAD fuel performance code (Reference 4-4 and 4-14) is the main code used for evaluating fuel rod performance PAD iteratively calculates the interrelated effects of temperature, 2 pressure, cladding elastic and plastic behavior, cladding corrosion, fission gas release, and fuel densification and swelling as a function of time and power. PAD evaluates the power history of a rod as a series of steady-state power levels with instantaneous changes from one power level to another.

PAD divides the fuel rod into several axial segments and each segment is assumed to operate at a constant set of conditions over its length. Fuel densification and swelling, cladding stresses and strains, temperatures, burnup and fission gas release are calculated separately for each axial segment and the effects are integrated to obtain the overall fission gas release and rod internal pressure. The coolant temperature rise along the rod is calculated based on the flow rate and axial power distribution and the cladding surface temperature is calculated considering the effects of corrosion and the possibility of local boiling.

PAD considers the fuel pellet as a solid cylinder with allowances for dishing, chamfering, and pellet chipping. To calculate thermal expansion, fuel densification and swelling, and fission gas release, the pellet is divided into equal volume concentric rings and each ring is assumed to be at its average temperature during a given time step. Axial and radial thermal expansion, swelling and densification are determined for each ring and these effects are integrated over the entire fuel rod to calculate the length of the fuel column and the void volume to calculate the rod internal pressure.

The version of the PAD code used for the analyses in Revision 0 of this report was PAD 3.4 (Reference 4-4). In July of 2000, Westinghouse received approval for PAD 4.0 (Reference 4-14). This newest version of the code includes a revised cladding creep model and irradiation growth model as well as updated cladding and oxide thermal conductivity values. Duke Power is implementing PAD 4.0 in the same forward fit approach as outlined in Reference 4-14. When newer versions of the PAD code are approved by the NRC for use by Westinghouse, Duke Power plans to use the new versions for licensing analyses in the same manner. 2

4.2 Fuel Rod Design Bases and Analyses

The design bases for the RFA design that will be used in McGuire and Catawba are identical to those given in Reference 4-3 for Vantage+ fuel. The fuel rod design bases and analysis methodologies are described below

4.2.1 Fuel Rod Internal Pressure

The fuel rod internal pressure design basis is that the fuel system will not be damaged due to excessive fuel rod internal pressure (Reference 4-3 and 4-6). The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward clad creep during steady-state operation and (2) extensive DNB propagation to occur.

4 2.1.1 Analysis

Part 1 of this design basis precludes the cladding outward creep rate from exceeding the fuel solid swelling rate, and, thus, ensures that during steady-state operation the fuel-cladding gap will not re-open following contact, or increase in size. The PAD code is used to predict fuel rod internal pressures that are used to verify that the fuel rod internal pressure design basis is met The rod average burnup at which the diametral gap begins to increase due to the outward cladding creep rate is calculated. This allowable rod burnup is compared to predicted rod burnups for each reload design to confirm that the rod internal pressure criterion is met for all of the fuel

A bounding pin power history, similar to that shown in Fig. 4-1, is used to perform a generic rod internal pressure analysis. A cycle-specific rod internal pressure analysis may be performed using predicted limiting pin power histories if the bounding power history does not envelope the pin powers for a future core design. The transient gas release contribution to the rod internal pressure must be included in the rod internal pressure analyses. Both Condition I axial xenon oscillations and Condition II overpower transients are considered in calculating the rod internal pressure.

Sensitivity studies have been performed to determine the design parameters and PAD models which are the most significant contributors to the uncertainty in the rod internal pressure. An upper bound rod internal pressure is calculated to account for the impact of possible variations in design parameters or models. The bounding pressure is compared to a lower bound steady-state pressure limit

Part 2 of the rod internal pressure design basis deals with DNB propagation, which is discussed in Reference 4-6. The current methodology for calculating the frequency and expected location of fuel rods experiencing both DNB and internal pressure greater than the reactor coolant system pressure is consistent with that used for the evaluations documented in Reference 4-6. For each rod that is both in DNB and above system pressure, the number of additional rods in DNB due to propagation effects are calculated based on whether the neighboring rods are in DNB or above system pressure. A fuel rod which is both in DNB and above system pressure is assumed to balloon at the location of DNB. When the ballooned clad contacts its neighboring rods, it is assumed that these rods will also experience DNB as a result of the flow blockage. If one of these rods is also above system pressure, it would also balloon to contact its neighboring rods. This process is assumed to continue if any of the neighbor rods are above system pressure. The total number of rods in DNB initially, rods above system pressure, rods both in DNB and above system pressure, and rods in DNB due to propagation are calculated.

4.2.2 Cladding Stress

The cladding stress design basis is the fuel system will not be damaged due to excessive fuel cladding stress (Reference 4-3 and 4-9). The volume average effective stress calculated with the Von Mises equation considering interference due to uniform cylindrical pellet cladding contact, caused by thermal expansion, pellet swelling and uniform cladding creep, and pressure differences, is less than the ZIRLO™ 0.2 % offset yield stress, with due consideration of temperature and irradiation effects under Condition I and II modes of operation. While the cladding has some capability for accommodating plastic strain, the yield stress has been established as a conservative design limit.

4.2.2.1 Analysis

Excessive clad stress can arise due to rapid local power increases such that clad creep cannot accommodate the pellet thermal expansion. The clad stress criterion is applied to the volume average effective stress which occurs as a result of a Condition II transient local power increase. The primary mechanism which increases the clad stresses during a Condition II transient, relative to the steady-state stresses, is the differential thermal expansion between the pellet and the cladding.

[ ]

For each reload design, the allowable changes in local linear heat rate (Δ kw/ft) as a function of burnup are compared to predicted peaking changes that result from either Condition I or II events.

4.2.3 Cladding Strain

The cladding strain design basis is that the fuel system will not be damaged due to excessive fuel cladding strain (Reference 4-3 and 4-9). The design limit is that during steady-state operation, the total plastic tensile creep strain due to uniform cladding creep and uniform fuel pellet expansion associated with fuel swelling and thermal expansion is less than 1% from the unirradiated condition. The acceptance limit for fuel rod cladding strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than 1% from the pre-transient value (Reference 4-2).

4.2.3.1 Analysis

The intent of this criterion is to minimize the potential for clad failure due to excessive clad straining. This criterion addresses slow strain rate mechanisms where the effective clad stress never reaches the yield strength due to stress relaxation. Clad strain allowable local power limits (Δ kw/ft) are calculated using PAD and the methodology discussed above for calculating clad stress local power limits. Analyses have generally shown that the transient clad stress analyses are more limiting than the transient clad strain analyses (i.e., the clad stress Δ kw/ft limits are typically more restrictive than the clad strain Δ kw/ft limits).

4.2.4 Cladding Fatigue

The cladding fatigue design basis is that the fuel system will not be damaged due to excessive clad fatigue (Reference 4-3 and 4-9). The fatigue life usage factor is limited to less than 1.0 to prevent reaching the material fatigue limit.

4.2.4.1 Analysis

A cladding fatigue analysis is performed to consider the accumulated effects of short term, cyclic, cladding stress and strain resulting primarily from daily load follow operation. The accumulated effects of cyclic strains associated with normal plant shutdowns and returns to full power are also considered.

The fatigue model in PAD calculates the low cyclic fatigue and the fatigue life fraction of a fuel rod during load follow operation, as a function of time and irradiation history. The Langer-O'Donnell low cyclic fatigue model (Reference 4-7) constitutes the basic approach used in the fatigue analysis. The empirical factors used in the Langer-O'Donnell fatigue model have been modified to conservatively bound the results of Westinghouse test programs presented in Reference 4-8. The design equations follow the concepts of the fatigue design criterion given in the ASME Code, Section III:

The calculated pseudo-stress amplitude (S_a) is multiplied by 2 to obtain the allowable number of cycles (N_f)

The allowable cycles for a given S_a is five percent of N_f or a safety factor of 20 on the number of cycles.

The lower of the two allowable number of cycles is selected and the cumulative fatigue life fraction is then calculated as:

$$n_k/N_{fk} < 1.0$$

where:

N_k = number of cycles of mode k

N_{fk} = number of allowable cycles

PAD is used to analyze a spectrum of pin power histories to determine the fatigue life.

4 2.5 Fuel Clad Oxidation and Hydriding

The fuel clad oxidation and hydriding design basis is that fuel damage will not occur due to excessive clad oxidation or hydriding (Reference 4-3). To limit metal-oxide formation to acceptable values, the ZIRLO™ metal-oxide interface temperature is limited

] (Reference 4-3). The clad and structural component hydrogen pickup is limited to [ ] (Reference 4-3) at end of life to preclude loss of ductility due to hydrogen embrittlement by the formation of zirconium hydride platelets.

4.2.5.1 Analysis

A spectrum of pin power histories, including a bounding power history similar to that shown in Fig. 4-1, are analyzed to verify that the cladding metal-oxide interface temperature limits are met during steady-state operation and during Condition II local power increases. For each steady-state power history, the temperature of the metal-oxide interface is calculated. The oxide layer on the fuel is calculated using the ZIRLO™ corrosion model described in Reference 4-3. At various times during the steady-state depletion, Condition II local power increases are simulated The local power is increased until the cladding metal-oxide interface temperature is equal to the transient cladding temperature limit An analysis is performed for each reload which verifies that the local power limit associated with the transient cladding temperature limit is not exceeded during Condition II events (Reference 4-11).

The methodology for calculating the hydrogen pickup of the cladding is the same as that described above for calculating the metal-oxide interface temperature. In addition to the zirc-oxide buildup on the cladding, the hydrogen pickup resulting from the corrosion process is calculated. Corrosion and percent metal wastage for the grids and thimbles is also calculated.

4.2.6 Fuel Temperature

The fuel temperature design basis is that fuel rod damage will not occur due to excessive fuel temperatures (Reference 4-3). The fuel system and protection system are designed to assure that for Condition I and II events, the calculated centerline fuel temperature does not exceed the fuel melting temperature. The melting temperature of unirradiated UO_2 is taken as 5080 °F, decreasing by 58 °F per 10,000 MWd/mtU of fuel burnup (Reference 4-3). A centerline fuel temperature of 4700 °F has been selected by Westinghouse as the design limit for fuel temperature analyses, References 4-9 and 4-10.

4 2.6.1 Analysis

The PAD code (Reference 4-4 and 4-14) is used to verify that the fuel temperature design limit is met. Using a fuel centerline temperature limit of 4700 °F covers both the reduction in melt temperature with burnup and manufacturing and modeling uncertainties. PAD is used to calculate the fuel centerline temperature and the local linear heat rate to prevent fuel melting or linear heat rate to melt (LHRTM). As explained in Reference 4-11 an analysis is performed for 2 each reload which verifies that this local power limit is not exceeded for Condition I and II events.

4.2.7 Fuel Clad Flattening

From Reference 4-3, the design basis for fuel clad flattening is that fuel rod failures will not occur due to clad flattening.

4.2.7 1 Analysis

Westinghouse demonstrated in Reference 4-5 that clad flattening will not occur for current Westinghouse fuel designs. Based on post irradiation examination and in-core flux data Westinghouse confirmed that significant axial gaps in the fuel column due to densification will not occur for current Westinghouse fuel. Therefore, it was concluded that clad flattening will not occur.

A new clad flattening evaluation is required only if any of the following fuel rod design parameters change: cladding creep properties, cladding thickness, fuel densification, rod prepressure, and as-fabricated pellet-clad gap. All of these parameters are related to the fuel design itself; they are not affected by a particular reload core design. For each new region of fuel; the cladding thickness, fuel rod prepressure, and as-fabricated pellet-clad gap will be verified to be within the range of parameters considered in Reference 4-5.

4.2 8 Fuel Rod Axial Growth

From Reference 4-3, the fuel rod growth design basis is that the fuel rods will be designed with adequate clearance between the fuel rod end plugs and the top and bottom nozzles to accommodate the difference in the growth of the fuel rods and the growth of the fuel assembly. The Westinghouse RFA was designed to assure that there is no interference between the fuel rods and the fuel assembly top and bottom nozzles during the design life of the fuel.

4.2.8.1 Analysis

The fuel rod growth model described in Reference 4-4 and Reference 4-14 is used to show that the fuel rod growth criterion is met. The rod growth analysis assumes upper bound fuel rod growth, lower bound fuel assembly growth, minimum initial fuel rod to nozzle gap, upper bound rod fast fluence, and nominal differential thermal expansion between the fuel rod cladding and the fuel assembly structure. A generic analysis is performed to calculate the maximum allowable rod average burnup for which the rod to nozzle gap is zero. For the current RFA design, the allowable rod burnup with respect to the rod growth criterion is greater than the licensed burnup limit of 60,000 MWd/mtU. Using the Westinghouse Fuel Criteria Evaluation Process (FCEP) (Reference 4-13), the burnup limit can be increased to 62,000 MWd/mtU for specific reload cores 2

4.3 References

4-1 Title 10, Chapter 1, Code of Federal Regulations - Energy, Part 50, "Domestic Licensing of Production and Utilization Facilities", Appendix A, "General Design Criteria for Nuclear Power Plants".

4-2 "Section 4.2, Fuel System Design", Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, NUREG-0800, Rev. 2, US Nuclear Regulatory Commission, July 1981

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4-4 Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations", WCAP-10851-P-A, August 1988.

4-5 P. J Kersting, et al., "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel", WCAP-13589-A, March 1995.

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4-8 S. L. Davidson and J A. Iorii, "Reference Core Report 17x17 Optimized Fuel Assembly", WCAP-9500-P-A, May 1982.

4-9 S. L. Davidson (ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel", WCAP-10125-P-A, December 1985.

4-10 S. L. Ellenberger, et al., "Design Bases for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions", WCAP-8745-P-A, September 1986

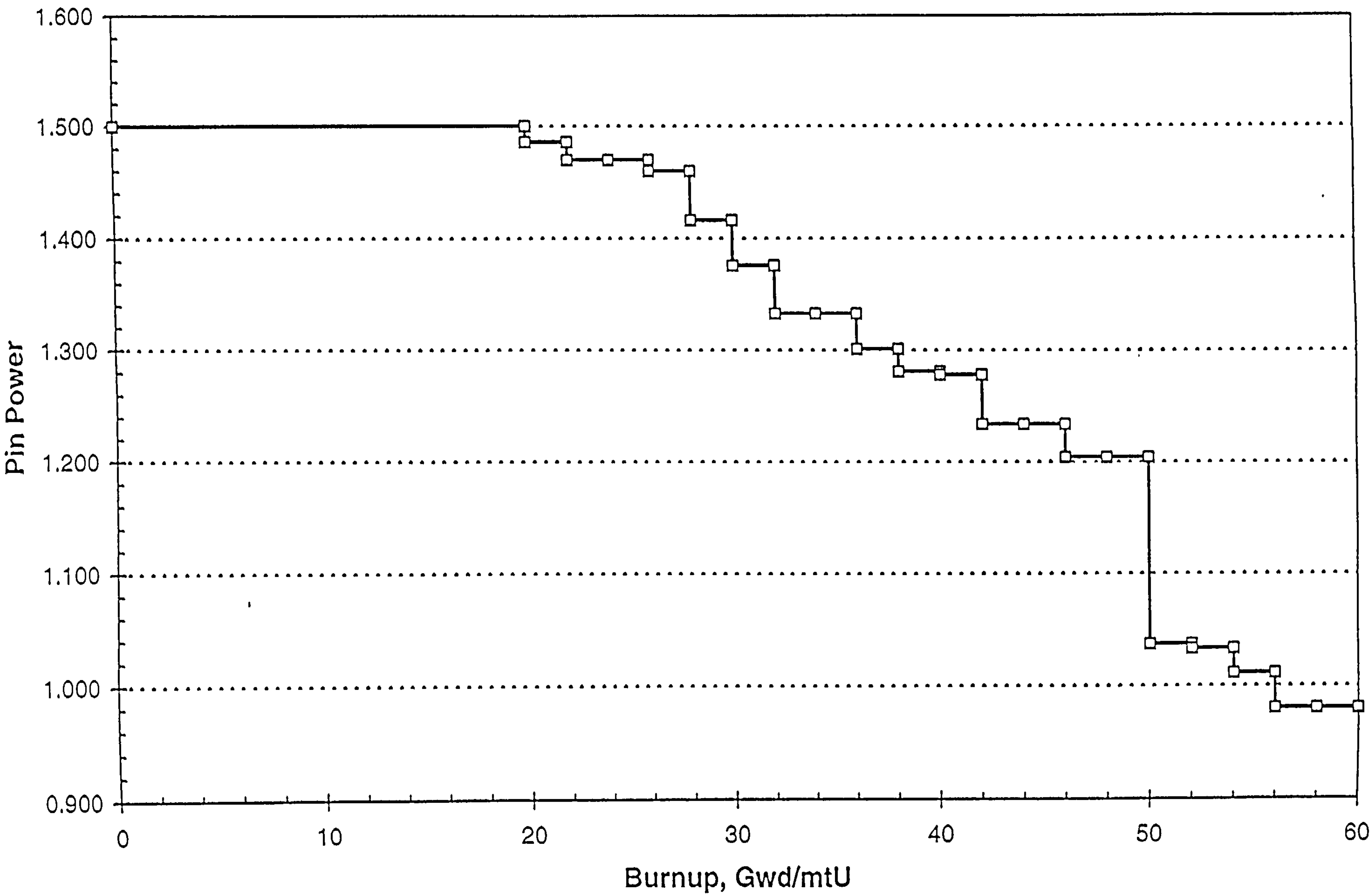
4-11 "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011P-A, March 1990.

4-12 "Duke Power Company Fuel Rod Mechanical Reload Analysis Methodology Using TACO3", DPC-NE-2008P-A, SER dated April 3, 1995.

4-13 S. L. Davidson (Editor), "Westinghouse Fuel Criteria Evaluation Process", WCAP-12488P-A, October 1994.

4-14 Foster, J. P., et al., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)", WCAP-15063-P-A Revision 1 with Errata, July 2000. 2

Fig. 4-1 Typical Bounding Pin Power History



5.0 THERMAL-HYDRAULIC ANALYSIS

Steady-state thermal-hydraulic analyses for the Westinghouse RFA design will be performed using the NRC approved methodology given in References 5-1 and 5-4. Reference 5-1 describes the VIPRE-01 core thermal-hydraulic models used for steady state analyses at McGuire and Catawba. The only changes necessary to perform core thermal-hydraulic analyses for the Westinghouse RFA design are to specifically model the fuel (dimensions, form loss coefficients, etc.) and to use the WRB-2M critical heat flux (CHF) correlation (Reference 5-2) The RFA design, VIPRE-01 models, and the WRB-2M CHF correlation are discussed in Sections 5.1, 5.2, and 5.3, respectively.

DPC-NE-2005P-A (Reference 5-4) describes Duke Power's NRC-approved methodology for calculating a Statistical Core Design (SCD) DNBR limit for application to pressurized water reactors. Individual appendices to the report list information necessary to complete the calculations for specific plants and fuel types. This includes the fuel data for the VIPRE-01 model, parameter uncertainties, the CHF correlation, and the range of conditions analyzed. The remainder of Chapter 5 is written in the same format as an appendix to Reference 5-4. Sections 5.1 through 5.3 list the plant specific data, models, and CHF correlation. Section 5 4 lists the range of statepoint conditions analyzed and Section 5.5 describes the key parameters and associated uncertainties. The statistical design limit, or SDL, which will be used for licensing analyses for Westinghouse Robust fuel at McGuire and Catawba is discussed in Section 5.6. Section 5.7 discusses how the impact of the geometric and hydraulic differences between the resident Mark-BW fuel and the Westinghouse RFA design is addressed and determines the SDL for RFA/Mark-BW transition cores.

Unless otherwise noted, all VIPRE-01 modeling inputs listed in Reference 5-1 for the 17x17 fuel at McGuire and Catawba are unchanged. The thermal-hydraulic SCD analysis discussed in this chapter was performed using the approved methodology given in the main body of Reference 5-4.

5.1 Plant Specific Data

This analysis is for the McGuire and Catawba plants (four-loop Westinghouse PWR's) with the RFA design. The Robust fuel design includes 0.374 OD fuel rods and non-structural Intermediate Flow Mixing (IFM) grids in the upper three spans to improve DNB performance. This design also includes the fuel reliability features of a debris filtering bottom and a protective grid between this nozzle and the first structural grid. See Chapter 2 of this report for a complete description of the fuel design.

The parameter uncertainties and statepoint ranges were selected to bound the McGuire and Catawba unit and cycle-specific values (see Sections 5.4 and 5.5)

5.2 Thermal-Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference 5-3 and the McGuire/Catawba eight channel model approved in Reference 5-1 are used in this analysis. The 2
VIPRE-01 models approved in Reference 5-1 for the Mark-BW fuel are used to analyze the RFA design with the following changes:

1) The RFA design geometry information is listed in Table 5-1. Applicable form loss coefficients as per the vendor were used in the models. Also, the axial noding was adjusted to be compatible with the Westinghouse WRB-2M CHF correlation.

2) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI model. Correspondingly, the subcooled void model was changed from the Levy to EPRI model.

3) The reference pin peak described in Reference 5-1 was increased from 1.60 to 2
1.67. The associated pin power distribution was also updated based on this higher value.

4) The reference axial power profile (symmetric chopped cosine) peak to average value described in Reference 5-1 was increased from 1.55 to 1.60.

With respect to Item 2), the Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 (void fractions of 0.85) and is discontinuous at higher values (Reference 5-3).

The EPRI bulk void model is essentially the same as the Zuber-Findlay bulk void model except for the equation used to calculate the drift velocity (Reference 5-3). This eliminates the discontinuity at high qualities and void fractions. Therefore, the EPRI model covers the full range (i.e., void fraction range, 0 - 1.0) of void fractions required for performing DNB calculations. Also, for overall void model compatibility, the subcooled void model was changed from the Levy model, as specified in Reference 5-1, to the EPRI correlation.

To evaluate the impact of changing bulk void models on DNB predictions, fifty-one RFA critical heat flux test data points (Reference 5-2) were compared using both the Levy/Zuber-Findlay and EPRI/EPRI subcooled void / bulk void model combinations in VIPRE-01. These data points cover a pressure range of 1519 to 2426 psia and an inlet temperature range 397.4 to 617.6°F. The mass flux at the MDNBR location varied from 1.48 to 3.02 Mlbm/hr-ft^2. The void fraction at the MDNBR location varied from 0.309 to 0.697. The equilibrium quality at the MDNBR location varied from 0.07 to 0.254 The results of this comparison are as follows:

	Levy/Zuber-Findlay	EPRI/EPRI
Minimum DNBR (Avg.)	1.029	1.028

The minimum DNBR results show a minimal difference of 0.1% (0.001 in DNB). Therefore, the EPRI bulk void model and EPRI subcooled void correlation will be used in RFA analyses.

The changes related to Items 3) and 4) above are due to the RFA fuel design containing 2
significant DNBR margin due to the addition of the IFM grids. This DNB margin is applied in core design space by increasing the reference radial and axial peaking. With respect to radial peaking, all three models described in Reference 5-1 (8, 12, and 75 channel models) are based on the maximum pin power value. Therefore, all three models were updated to the new peak pin value of 1.67. The resulting pin power distributions from this change are shown in Figures 5-1, 5-2, and 5-3.

5.3 Critical Heat Flux Correlation

The WRB-2M critical heat flux correlation described in Reference 5-2 is used for all statepoint analyses. This correlation was developed by Westinghouse for application to the RFA design. As discussed in Reference 5-2 the WRB-2M correlation was developed with the VIPRE-01 thermal-hydraulic computer code. This correlation was programmed into the Duke Power version of VIPRE-01 and will be used in DNBR calculations for the RFA design, except for the following: 2

1. steam line break transient (see Section 6.2.2).

2. the non-mixing vane span of the RFA fuel (below the first mixing vane zircaloy grid). For this region of the fuel, the BWU-N CHF correlation will be applied (Reference 5-5) to the RFA fuel. 2

5.4 Statepoints

The statepoint conditions evaluated in this analysis are listed in Table 5-2. These statepoints cover the range of conditions to which the statistical DNBR limit will be applied. The range of key parameter values evaluated in this analysis are listed on Table 5-5.

5.5 Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table 5-3. The uncertainties were selected to bound the values calculated for each parameter at McGuire and Catawba.

5.6 DNB Statistical Design Limit

The statistical DNBR value for each statepoint evaluated is listed on Table 5-4. Section 1 of Table 5-4 contains the 500 case runs and Section 2 contains the 5000 case runs. The number of cases was increased from 3000 to 5000 as described in Attachment 1 of the main body of Reference 5-4. The DNBRs calculated for all of the statepoints are normally distributed. As shown in Section 2 of Table 5-4 the maximum statepoint statistical DNBR value is [ ].

Therefore, the statistical design limit (SDL) using the WRB-2M CHF correlation for the RFA design at McGuire/Catawba is conservatively determined to be 1.30.

5.7 Transition Cores

A transition core model is used to determine the impact of the geometric and hydraulic differences between the resident FCF Mark-BW fuel and the Westinghouse RFA design. The 8 channel model described in Reference 5-1 is used to evaluate the impact of transition cores containing the RFA design. In Figure 5 of Reference 5-1, the RFA design is used instead of Mark-BW fuel. Therefore, the limiting assembly (Channels 1 through 7) is modeled as the RFA design and the remainder of the core (Channel 8) is modeled as Mark-BW fuel. The transition core analysis models each fuel type in their respective locations with the correct geometry. The form loss coefficients for each fuel design are input so the effect of crossflow out of the IFM grid spans in the limiting channel is calculated.

To evaluate the impact of the transition core on the statistical DNBR limit, the most limiting full core statepoint (Statepoint 12 on Table 5-4) was evaluated using the 8 channel transition core model. This case is designated as statepoint 12TR in Sections 1 and 2 of Table 5-4. The statistical DNBR calculated using the transition core model (statepoint 12TR) is slightly greater than the Statistical DNBR value for the full RFA core (statepoint 12) at both the 500 and 5000 cases levels. As shown in Section 2 of Table 5-4, this value is still less than 1.30. Therefore, the statistical design limit of 1.30 is bounding for RFA/Mark-BW transition cores as well as full RFA cores 2

For initial transition reload cycles, a transition core DNBR penalty is determined for the RFA design using the 8 channel RFA/Mark-BW transition core model. For subsequent cycles where the RFA fuel composes greater than 80% of the assemblies incore, the 75 channel model shown in Figure 5-3 and described in Reference 5-1 is used to determine a transition core penalty. In either case, a conservative penalty is applied for all DNBR analyses in transition cycles to bound the effects of mixed cores. 2

5.8 References

5-1 DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, Rev 1, February 1997.

5-2 WCAP-15025-P, Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, Westinghouse Energy Systems, February 1998.

5-3 VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

5-4 DPC-NE-2005P-A, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, Rev 2, June 1999. 2

5-5 BAW-10199P-A, The BWU Critical Heat Flux Correlations, Framatome Cogema Fuels, April 1996.

Table 5-1

RFA Design Data

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, inches (Nominal)	0.374
Guide tube diameter, inches (Nominal)	0.482
Fuel rod pitch, inches (Nominal)	0.496
Fuel Assembly pitch, inches (Nominal)	8.466
Fuel Assembly length, inches (Nominal)	160.0

GENERAL FUEL CHARACTERISTICS

Component	Material	Number	Location/Type
Grids	Inconel	1	Lower Protective
	Inconel	2	Upper and Lower Non-Mixing Vane
	ZIRLO™	6	Intermediate Mixing Vane
	ZIRLO™	3	Intermediate Flow Mixing (Non-structural)
Nozzles	304SS	1	Debris Filtering Bottom
	304SS	1	Removable Top

Table 5-2

McGuire/Catawba SCD Statepoints, WRB-2M Correlation

Stpt No.	Power* (% RTP)	RCS Flow** (K gpm)	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak (FΔH)
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
12TR***						

* 100% RTP = 3411 Megawatts Thermal

** Mass flow rate should be calculated using the given core inlet temp.

*** TR - transition core model

Table 5-3

McGuire/Catawba Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type Of Distribution
Core Power*	+/- 2% / 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 4.0% / 2.43%	Normal
$F^E_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.41% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	+/- 10.73% / 6.52%	Normal
Code/Model	[ ]	Normal

* Percentage of 100% RTP (3411 MWth)

Table 5-3 (Continued)

McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
Core Power	The core power uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from normally distributed random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method (SRSS). Since the uncertainty is calculated from normally distributed values, the parameter distribution is also normal.
Core Flow	
Measurement	Same approach as core power.
Bypass Flow	The core bypass flow is the parallel core flow paths in the reactor vessel (guide thimble cooling flow, head cooling flow, fuel assembly/baffle gap leakage, and hot leg outlet nozzle gap leakage) and is dependent on the driving pressure drop. Parameterizations of the key factors that control ΔP, dimensions, loss coefficient correlations, and the effect of the uncertainty in the driving ΔP on the flow rate in each flow path, was performed. The dimensional tolerance changes were combined with the SRSS method and the loss coefficient and driving ΔP uncertainties were conservatively added to obtain the combined uncertainty. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as pressure.
$F^N_{\Delta H}$	
Measurement	This uncertainty is the measurement uncertainty for the movable incore instruments. A measurement uncertainty can arise from instrumentation drift or reproducibility error, integration and location error, error associated with the burnup history of the core, and the error associated with the conversion of instrument readings to rod power. The uncertainty distribution is normal.

Table 5-3 (Continued)

McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^E_{\Delta H}$	This uncertainty accounts for the manufacturing variations in the variables affecting the heat generation rate along the flow channel. This conservatively accounts for possible variations in the pellet diameter, density, and U_{235} enrichment. This uncertainty distribution is normal and was conservatively applied as one-sided in the analysis to ensure the MDNBR channel location was consistent for all cases.
Spacing	This uncertainty accounts for the effect on peaking of reduced hot channel flow area and spacing between assemblies. The power peaking gradient becomes steeper across the assembly due to reduced flow area and spacing. This uncertainty distribution is normal and was conservatively applied as one-sided to ensure consistent MDNBR channel location.
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty distribution is applied as normal.
Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one of the physics code's axial nodes. The uncertainty distribution is conservatively applied as uniform.
DNBR Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normal.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is applied as normal.

Table 5-4

McGuire/Catawba Statepoint Statistical Results

SECTION 1
WRB-2M Critical Heat Flux Correlation
500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
12TR*				

* TR - transition core model

Table 5-4 (Continued)

McGuire/Catawba Statepoint Statistical Results

SECTION 2

WRB-2M Critical Heat Flux Correlation

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
7				
11				
12				
12TR*				

* TR - transition core model

Table 5-5

McGuire/Catawba Key Parameter Ranges

WRB-2M CHF Correlation

Parameter	Maximum	Minimum
Core Power* (% RTP)		
Pressure (psia)		
T inlet (deg. F)		
RCS Flow (Thousand GPM)		
FΔH, Fz, Z		

* 100% RTP = 3411 Megawatts Thermal

All values listed in this table are based on the currently analyzed statepoints (Table 5-2). Ranges are subject to change based on future statepoint conditions.

FIGURE 5-1
8 CHANNEL MODEL - GEOMETRY AND REFERENCE POWER DISTRIBUTION

FIGURE 5-2
12 CHANNEL MODEL - GEOMETRY AND REFERENCE POWER DISTRIBUTION

FIGURE 5-3
75 CHANNEL MODEL - GEOMETRY AND REFERENCE POWER DISTRIBUTION

6.0 UFSAR ACCIDENT ANALYSES

DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology" (Reference 6-1), DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology" (Reference 6-2), and DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology" (Reference 6-3) describe the Duke Power NRC-approved models and methodology for analyzing UFSAR Chapter 15 Non-LOCA transients and accidents. DPC-NE-3004-PA, "Mass and Energy Release and Containment Response Methodology" (Reference 6-4), describes the Duke Power NRC-approved models and methodology for analyzing UFSAR Chapter 6.2 mass and energy release accidents and containment response.

UFSAR Chapter 15 non-LOCA analyses will continue to be performed according to the methodologies described previously in Reference 6-1, Reference 6-2, and Reference 6-3, except as noted in Sections 6.1-6.3, respectively. LOCA mass and energy release analyses (UFSAR Chapter 6.2) will continue to be performed according to the methodology described in Reference 6-4, except as noted in Section 6.4. LOCA analyses (UFSAR Chapter 15.6.5) will be performed by Westinghouse as described in Section 6.5.

6.1 Thermal-Hydraulic Transient Analysis Methodology (DPC-NE-3000)

DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology" (Reference 6-1), serves as the Duke Power Company response to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Action," which requires that licensees performing their own safety analyses demonstrate their analytical capabilities. Reference 6-1 describes the RETRAN-02 (Reference 6-21) system transient thermal-hydraulic models, and the VIPRE-01 (Reference 6-20) core thermal-hydraulic models developed for Oconee, McGuire, and Catawba Nuclear Stations. The previous comparisons of computer code results to experimental data, plant operational data, and other benchmarked analyses, continue to demonstrate the analytical capability to perform non-LOCA transient thermal-hydraulic analyses. Changing from Mark-BW to the RFA design does not affect this conclusion.

A review of Reference 6-1 indicates that only portions of Chapter 3 (McGuire/Catawba Transient Analyses) currently do not support the RFA design from a technical standpoint. Chapters 2 and 4 pertain to Oconee Nuclear Station only, and therefore remain unaffected. Chapter 5 pertains to McGuire/Catawba RETRAN benchmark analyses, which continue to demonstrate analytical capability to perform non-LOCA transient thermal-hydraulic analyses regardless of fuel type. Chapters 1 (Introduction) and 6 (Summary) are affected from an editorial standpoint only.

6.1.1 Plant Description (Section 3.1 in DPC-NE-3000)

The only difference with respect to the plant description will be the change from Mark-BW fuel to the RFA design. Chapter 2 of this report gives a complete description of the RFA design.

6.1.2 McGuire/Catawba RETRAN Model (Section 3.2 in DPC-NE-3000)

Volumes [ ] in the primary system nodalization scheme represent the reactor core region from the [ ] Dimensional changes due to the change to the RFA design will require minor changes to these volume calculations, as well as associated junction and heat conductor calculations.

6.1.3 McGuire/Catawba VIPRE Model (Section 3.3 in DPC-NE-3000)

The McGuire/Catawba simplified [ ] channel model in Reference 6-1 is used for analyzing the RFA design. As described in Chapter 5, the reference radial pin power distribution remains unchanged, but the peak pin is increased from 1.50 to 1.60 and the WRB-2M CHF correlation (Reference 6-5) and the SCD limit developed in Chapter 5 are used. The axial node size is adjusted to be compatible with the WRB-2M CHF correlation. The RFA design geometry is listed in Table 5-1 and applicable form loss coefficients are used. The remaining code inputs and options remain identical to that originally approved in Reference 6-1.

No transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses. As discussed in Reference 6-1, the [ ] channel model used for transient analyses was originally developed with additional conservatism over the 8 channel model used for steady-state analyses to specifically minimize the impact of changes in core reload design methods or fuel assembly design. Should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly.

6.2 Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology (DPC-NE-3001)

DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology" (Reference 6-2), describes the Duke Power Company methodologies for simulating the UFSAR Chapter 15 events characterized by multidimensional reactor transients (rod ejection, steam line break, and dropped rod), and for systematically confirming that reload physics parameters important to Chapter 15 transients and accidents are bounded by values assumed in the licensing analyses (the Safety Analysis Physics Parameters (SAPP) methodology). The SAPP methodology remains unchanged when analyzing the RFA design. Thermal-hydraulic changes for analyzing the RFA design in rod ejection, steam line break, and dropped rod accidents are discussed in the sections that follow.

6.2.1 Rod Ejection

The changes presented in Section 6.1 also apply to the rod ejection accident. The nuclear analysis of the rod ejection accident using SIMULATE-3K is presented in Section 6.6. The remainder of the rod ejection thermal-hydraulic methodology presented in Reference 6-2 remains unchanged.

6.2.2 Steam Line Break

The changes presented in Section 6.1 also apply to steam line break, with the exception of the CHF correlation. Since the WRB-2M CHF correlation pressure range of applicability is not acceptable for steam line break analyses (see Chapter 5 of this report for ranges of applicability), the W3-S CHF correlation will continue to be used as originally documented in Reference 6-2.

The remainder of the steam line break thermal-hydraulic methodology presented in Reference 6-2 remains unchanged, except for the selection of subcooled and bulk void models for offsite power lost (OSPL) cases for reasons described in Chapter 5. The [ ] for steam line break cases for which offsite power is lost. This is acceptable since the [ ] gives more conservative DNBR results for steady-state cases (according to Reference 6-1), and preliminary studies of steam line break cases show no difference in results.

6.2.3 Dropped Rod

The changes presented in Section 6.1 also apply to the dropped rod transient. The remainder of the dropped rod thermal-hydraulic methodology presented in Reference 6-2 remains unchanged.

6.3 UFSAR Chapter 15 System Transient Analysis Methodology (DPC-NE-3002)

DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology" (Reference 6-3) documents the conservative modeling assumptions used by Duke Power Company in performing the NSSS primary and secondary system analyses of UFSAR Chapter 15 accidents. It covers all applicable non-LOCA accidents in UFSAR Sections 15.1-15.6, except those already discussed in Reference 6-2. There are no changes to Reference 6-3 with respect to analyzing the RFA design.

6.4 Mass and Energy Release and Containment Response Methodology (DPC-NE-3004)

DPC-NE-3004-PA, "Mass and Energy Release and Containment Response Methodology" (Reference 6-4), describes the Duke Power Company methodology for simulating the mass and energy release from high energy line breaks (LOCA and steam line break) and the resulting containment response to demonstrate that the containment peak pressure and temperature limits are not exceeded. Since the fuel stored energy for the RFA design is similar to that for the Mark-BW fuel, there are no changes anticipated for Reference 6-4 with respect to the RFA design except the RETRAN related changes described in Section 6 1 of this report. Similar changes to

the RELAP5 model, which is used to model the mass and energy release from LOCAs, are also anticipated. The RETRAN and RELAP5 model changes for the RFA design are not significant enough to require reanalyses. Future reanalyses will incorporate the RFA design model revisions.

6.5 LOCA Analyses

Large and small break LOCA analyses will be performed by Westinghouse using approved versions of the Westinghouse Appendix K LOCA evaluation models. All features employed have been approved by the NRC as required and annual model reports for the evaluation models have been supplied to the NRC, the most recent of which is found in Reference 6-22. Therefore, no NRC review of the evaluation model features is necessary, and only methodology with respect to analyzing McGuire/Catawba will be presented in this section. New LOCA analyses will be performed to support the licensing of McGuire/Catawba during the transition and full core operation of the RFA design.

6.5.1 Small Break LOCA

For small break LOCAs (SBLOCAs) due to breaks less than 1 ft^2, Westinghouse developed the NOTRUMP computer code (Reference 6-23) to calculate the transient depressurization of the reactor coolant system (RCS) as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 6-24, 6-25, 6-26, and 6-27) was developed and licensed by Westinghouse to determine the RCS response to design basis SBLOCAs, and to address NRC concerns expressed in NUREG-0737, Item II.K.3.30.

In addition, several model enhancements have been made to the evaluation model and 1
implemented via the 10 CFR 50.46 process. These enhancements or changes were determined to be non-significant as defined by 10 CFR 50.46. Westinghouse reported these enhancements to the NRC in annual notification reports (References 6-22, 6-28 and 6-39) and implemented them on a forward fit basis. Duke did not report these changes in their annual 10 CFR 50.46 reports since the Westinghouse SBLOCA analysis using these enhancements had not been implemented

for McGuire and Catawba during this time period. The purpose of identifying these enhancements in this report is to clearly identify the SBLOCA analysis method to be used to support McGuire and Catawba. 1

The NRC approved noding scheme for the NOTRUMP Evaluation Model is shown in Reference 6-24, although minor noding changes to facilitate the modeling of broken loop ECCS were instituted and reported to the NRC in Reference 6-28. Peak cladding temperature (PCT) calculations are performed with the LOCTA-IV code (Reference 6-29) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. Additional modifications to the LOCTA-IV code to allow the modeling

of annular fuel pellets in the axial blankets have been reviewed and approved by the NRC in Reference 6-27. The axial shape chosen for McGuire/Catawba SBLOCA will be based on the desired core operating limits and axial offset control strategy so as to bound all burnups and operating cycles.

Due to the nature of SBLOCA transients, the rod heatup and resulting calculated PCT is insensitive to transition core effects, and an evaluation is performed to demonstrate that this is a valid assumption. Therefore, SBLOCA will generally have no additional penalty for transition core effects.

6.5.2 Large Break LOCA

For the Westinghouse large break LOCA (LBLOCA) methodology, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. The most recent version of the 1981 Westinghouse Large Break LOCA ECCS Evaluation Model with BASH (Reference 6-30) will be used to perform the LBLOCA analysis for the transition of McGuire/Catawba to the RFA design. A description of the various aspects of the Westinghouse LOCA analysis methodology can be found in WCAP-8339 (Reference 6-31). This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the acceptance criteria. The SATAN-VI (Reference 6-32), WREFLOOD (Reference 6-33), BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail in References 6-30 and 6-34. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling through and subsequent to the blowdown, refill, and reflood phases of the LOCA. The LOTIC computer code (Reference 6-35) calculates the minimum containment backpressure transient required for LBLOCA analyses in Appendix K to 10 CFR Part 50. Although there have been several updates to the original SATAN-VI code, the most notable upgrade is delineated in Reference 6-34.

The WREFLOOD code has been replaced by the REFILL code as reported in Reference 6-36. The REFILL code is identical to the section of the WREFLOOD code that modeled the refill phase of the transient. There has also been a recent change (the incorporation of the REFILL and LOCTA codes directly into the BASH code as subroutine modules) in the methodology for execution of the

BASH Evaluation Model as reported in Reference 6-37. In addition, the LOTIC code has been coupled with the BASH code so that the codes run interactively. The BASH Evaluation Model now utilizes the SATAN code for the blowdown calculations, the BASH code for the refill and reflood phases with interactive LOTIC calculations for containment backpressure, and the LOCBART code for the fuel rod heatup calculations. The most recent version of the LOCBART code employs an improved grid heat transfer model which has been approved by the by NRC in Reference 6-38.

An input parameter that affects LOCA analysis results is the assumed axial power shape at the beginning of the accident. The methodology employed by Westinghouse is termed ESHAPE (Explicit SHape Analysis for Pct Effects). The ESHAPE methodology is based upon explicit analysis of the LBLOCA transient with a set of bounding skewed axial power shapes to supplement the base analysis performed with the chopped cosine power shape. The limiting case break, as demonstrated with a chopped cosine, will be reanalyzed using skewed power shapes and typically demonstrate that the chopped cosine power shape is limiting.

As required in Appendix K to 10 CFR 50, a minimum of a three break spectrum will be analyzed. In addition, as required in the NRC Safety Evaluation Report (SER) for the BASH Evaluation Model, a maximum Safety Injection flow case will be analyzed.

When assessing the effect of transition cores on the LBLOCA analysis, it must be determined whether the transition core can have a greater calculated peak cladding temperature (PCT) than a complete core of the RFA design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core. An evaluation will be performed to address the cross-flow effects due to any hydraulic mismatch between the current fuel and the Westinghouse fuel. If it is determined that a transition core penalty is required during the cycles that both fuels reside in the core, it will be applied as an adder to the LOCA results for a full core of the RFA design.

6.6 Rod Ejection Analysis Using SIMULATE-3K

This section presents an improved methodology to be used by Duke Power to perform the nuclear analysis portion of the rod ejection accident (REA) analysis for the McGuire and Catawba Nuclear Stations. The current approved REA analysis methodology is described in the topical report titled, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters" (Reference 6-2) and uses the computer code ARROTTA to perform the nuclear analysis portion of the REA calculation. A Safety Evaluation Report (SER) for this topical was received on November 15, 1991 (Reference 6-6). The new methodology is based on the SIMULATE-3K (Reference 6-9) computer code which employs a three-dimensional neutron kinetics model based on the QPANDA two-group nodal model to calculate three-dimensional power distributions, core reactivity, or a core power level for both static and transient applications.

The SIMULATE-3K methodology affords compatibility with the current SIMULATE-3P nuclear design methodology (Reference 6-8) and will enhance the generation of forcing functions (transient core power distribution and hot assembly peak pin power distribution) at bounding physics parameter conditions for input into fuel enthalpy, peak RCS pressure, and DNB calculations. The SIMULATE-3K cross section model is also more robust than that used by ARROTTA. The transition from ARROTTA to SIMULATE-3K will reduce the engineering resources required to perform future REA analyses and enhance the transition from Mark-BW fuel to Westinghouse RFA or other fuel types in the future.

The basic methodology described in Reference 6-2 for the nuclear analysis portion of the REA remains intact with only minor differences which are outlined in this report. All other methods described in Reference 6-2 remain unchanged, i.e. core thermal-hydraulic and system thermal-hydraulic analysis. To demonstrate the transient capability of SIMULATE-3K, comparisons between SIMULATE-3K and ARROTTA reference REA analyses at beginning-of-cycle (BOC) and end-of-cycle (EOC), hot full power (HFP) and hot zero power (HZP) conditions were performed. These comparisons demonstrate the acceptability of the physical and numerical models within the SIMULATE-3K code as compared to the current licensed methodology.

A description of the models employed and the benchmark calculations performed in the verification of the SIMULATE-3K computer code are presented in Section 6.6.1. This section also includes a comparison of ARROTTA and SIMULATE-3K REA results applicable to the McGuire and Catawba Nuclear Stations at BOC and EOC, HFP and HZP conditions.

Section 6.6.2 describes the nuclear analysis methodology to be used in the evaluation of the UFSAR Chapter 15 REA using SIMULATE-3K.

6.6.1 SIMULATION CODES AND MODELS

6.6.1.1 CASMO-3 & SIMULATE-3P

CASMO-3 is used to produce two energy group edits of homogenized cross sections, assembly discontinuity factors, fission product data, and pin power data for input to ARROTTA, SIMULATE-3P, and SIMULATE-3K core models. CASMO-3 is a multigroup, two dimensional transport theory code for burnup calculations on PWR or BWR fuel assemblies. The code models a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with integral burnable absorber, lumped burnable absorber rods, clustered discrete control rods, incore instrument channels, assembly guide tubes, and intra-assembly water gaps. The program utilizes a cross section library based on ENDF/B-IV with some data taken from ENDF/B-V. Reference 6-11 provides a detailed description of the theory and equations solved by CASMO-3. The use of CASMO-3 in this report is consistent with the previously approved methodologies of References 6-8 and 6-2.

SIMULATE-3P is used to set up the cycle-specific model and conditions for the REA. It may also be used to generate pin-to-assembly factors for the conversion of nodal powers to pin powers for the REA analyses. SIMULATE-3P is a three-dimensional, two energy group, diffusion theory core simulator program which explicitly models the baffle and reflector regions of the reactor. Homogenized cross sections and discontinuity factors developed with CASMO-3 are used on a coarse mesh nodal basis to solve the two group diffusion equations using the QPANDA neutronics model. A nodal thermal hydraulics model is incorporated to provide both fuel and moderator temperature feedback effects. Inter- and intra-assembly information from the coarse mesh solution is then utilized along with the pinwise assembly lattice data from CASMO-3 to reconstitute pin-by-pin power distributions in two and three dimensions. The program performs a macroscopic depletion of fuel with microscopic depletion of iodine, xenon, promethium, and samarium fission products. Reference 6-10 provides a detailed description of the theory and equations solved by SIMULATE-3P. The use of SIMULATE-3P in this report is consistent with the previously approved methodologies of References 6-8 and 6-2.

6.6.1.2 ARROTTA

ARROTTA is a three-dimensional, two energy group diffusion theory core simulator applicable for both static and transient kinetics simulations. Homogenized cross sections, discontinuity factors, and six groups of delayed neutron precursor data are generated with CASMO-3 and used on a coarse mesh nodal basis to solve the two energy group diffusion equations using the QPANDA neutronics model. The thermal-hydraulic model is comprised of both fluid dynamics and heat transfer models. Reference 6-12 provides a detailed description of the theory and equations solved by ARROTTA. The use of ARROTTA for the benchmark calculations performed in this report is consistent with the previously approved methodology documented in Reference 6-2.

6.6.1.3 SIMULATE-3K

6.6.1 3.1 Code Description

The SIMULATE-3K code (Reference 6-9) is a three-dimensional transient neutronic version of the SIMULATE-3P code (Reference 6-10). SIMULATE-3K uses the QPANDA full two-group nodal spatial model developed in SIMULATE-3P, with the addition of six delayed neutron groups. The program employs a fully-implicit time integration of the neutron flux, delayed neutron precursor, and heat conduction models. Beta is fully functionalized similar to other cross sections to provide an accurate value of beta for the time-varying neutron flux. The control of time step size may be determined either as an automated feature of the program or by user input. Use of the automated feature allows the program to utilize larger time steps (which may be restricted to a maximum size based on user input) at times when the neutronics are changing slowly and smaller time steps when the neutronics are changing rapidly.

Additional capability is provided in the form of modeling a reactor trip. The trip may be initiated at a specific time in the transient or following a specified excore detector response. Use of the excore detector response model to initiate the trip allows the user to specify the response of individual detectors as required to initiate the trip, as well as the time delay prior to release of the control rods. The velocity of the control rod movement is also controlled by user input.

The SIMULATE-3K thermal-hydraulic model includes a spatial heat conduction and a hydraulic channel model. The heat conduction model solves the conduction equation on a multi-region mesh in cylindrical coordinates. Temperature-dependent values may be employed for the heat capacity, thermal conductivity, and gap conductances. A single characteristic pin conduction calculation is performed consistent with the radial neutronic node geometry, with an optional calculation of the peak pin behavior available to monitor local maxima. A single characteristic hydraulic channel calculation is performed based on the radial neutronic node geometry. The model allows for direct moderator heating at the option of the user. This thermal-hydraulic model is used to determine fuel and moderator temperatures for updating the cross-sections, and may additionally be used to provide edits of fuel temperature throughout the transient.

The SIMULATE-3K program utilizes the same cross-section library and reads the same restart file (exposure and burnup-related information) as SIMULATE-3P. Executed in the static mode, SIMULATE-3K performs the same solution techniques, pin power reconstruction, and cross-section development as SIMULATE-3P. Additional features of SIMULATE-3K include the application of conservatism to key physics parameters through simple user input. Also, the inlet thermal-hydraulic conditions can be provided on a time dependent basis through user input.

6.6.1.3.2 SIMULATE-3K Code Verification

The SIMULATE-3K code has been benchmarked against many numerical steady state and transient benchmark problems by the code vendor, Studsvik of America, Inc. The results of these benchmarks are described in Reference 6-9 and show excellent agreement between

SIMULATE-3K and the reference solutions. Some of the SIMULATE-3K benchmarks which have been performed are: The fuel conduction and thermal-hydraulics model has been benchmarked against the TRAC code (Reference 6-13). The transient neutronics model has been benchmarked, using standard LWR problems, to reference solutions generated by QUANDRY (Reference 6-14), SPANDEX (Reference 6-15), NEM (Reference 6-16), and CUBBOX (Reference 6-17). Finally, a benchmark of the coupled performance of the transient neutronics and thermal-hydraulic models was provided by comparison of results from a standard NEACRP rod ejection problem to the PANTHER code (Reference 6-18). Steady-state components of the SIMULATE-3K model are implemented consistent with the CASMO-3/SIMULATE-3P methodology and performance benchmarks which were approved for use on all Duke Power reactors in Reference 6-8. In addition, a benchmark to ARROTTA for the Oconee REA analyses was performed in topical report DPC-NE-3005-P, "Oconee UFSAR Chapter 15 Transient Analysis Methodology (Reference 6-19).

6.6.1.3.3 SIMULATE-3K / ARROTTA REA Benchmark

The three dimensional neutron kinetics capability of the SIMULATE-3K code is demonstrated by comparing SIMULATE-3K and ARROTTA calculations for the reference rod ejection accident analyses performed at BOC and EOC, HFP and HZP conditions for McGuire and Catawba. For the REA benchmark, ARROTTA and SIMULATE-3K are used to calculate the core power level and nodal power distribution versus time during the rod ejection transient for the BOC and EOC, HFP and HZP REA cases. These comparisons demonstrate the acceptability of the physical and numerical models within SIMULATE-3K for application in the REA analyses for McGuire and Catawba Nuclear Station.

The reference core used in the benchmark calculations is a hypothetical Catawba 1 Cycle 15 core. This core represents typical fuel management strategies (i.e. core loadings and cycle lengths) currently being developed for reload core designs at McGuire and Catawba Nuclear Stations. The ARROTTA and SIMULATE-3K models for this core were then adjusted to produce a conservative initial condition Doppler and moderator temperature coefficient, ejected rod worth, Beta, and power distribution as described in the "Multidimensional Reactor Transients and Safety Analysis Physics Parameter" topical report DPC-NE-3001 (Reference 6-2).

The combination of these conservative input parameters produces conservative transient results. The assembly enrichments, burnable poison loading, and assembly exposures for the reference core are shown in Figure 6-1. The core consists of all Framatome Mark-BW fuel.

6.6.1.3.3.1 ARROTTA Analysis

The ARROTTA REA analysis is based on the methodology described in the "Multidimensional Reactor Transients and Safety Analysis Physics Parameters" topical report DPC-NE-3001 (Reference 6-2) with the exceptions that the initial power conditions have been increased to reflect a design pin FΔH of 1.6, and the ARROTTA model was updated to reflect the C1C15 reference core design.

The REA analyses of Reference 6-2 were made limiting by setting key physics parameters to conservative or bounding values. Utilizing this approach produces limiting results which are expected to bound future reload cycles. The ARROTTA model was adjusted to produce conservative MTC, DTC, Beta, and ejected rod worths as identified in Tables 6-3.

6.6.1.3.3.2 SIMULATE-3K Analysis

The SIMULATE-3K analysis is performed as described in DPC-NE-3001, Reference 6-2. The SIMULATE-3K model employed in this analysis was adjusted to be functionally equivalent to the ARROTTA model to account for differences in the two codes cross section model. Since ARROTTA is restricted to one node per fuel assembly in the radial direction, the SIMULATE-3K model was set up to be consistent with this assumption. The axial nodalization depends on the fuel assembly design, such as whether or not axial blanket fuel is being modeled. For the analysis presented, an axial nodalization of 18 equal length fuel nodes is used.

[ ]

Additional model adjustments were performed to produce limiting values for the Doppler temperature coefficient, moderator temperature coefficient, ejected rod worth, and Beta. Table 6-3 provides a summary of initial condition values for each of these parameters for the SIMULATE-3K analyses. Trip times were input to be consistent with the ARROTTA analyses.

6.6.1.3.3.3 Results

ARROTTA results from each of the four cases evaluated are summarized in Table 6-1. Results from the SIMULATE-3K cases are provided in Table 6-2. Table 6-3 lists the REA initial condition kinetics parameters for both the ARROTTA and SIMULATE-3K benchmarks. Core power versus time for each case is shown in Figures 6-2 through 6-4.

For the HFP cases, which begin at 102% power, core power increases rapidly as the control rod is ejected The ejected rod worth in these transients is not sufficient to achieve a prompt critical state. Power increases until Doppler feedback from increasing fuel temperature begins to turn the excursion around. Core power level continues to decrease as the fuel temperature approaches an equilibrium value. A reactor trip signal on high flux occurs very early in these transients but the conservative trip delay time prevents rod motion until after the peak core power occurs Additional conservatisms applied to the rate of rod insertion and scram worth minimizes the effect of the reactor trip until the rods approach the bottom of the reactor core.

The transients initiated from HZP differ from the at-power initial conditions in that the ejected rod worth is large enough to achieve a prompt critical core. The power increase continues after the control rod is fully ejected until the fuel heats up enough for Doppler feedback to turn the excursion around. Conservatisms on trip delay time, rate of rod insertion, and scram worth minimize the impact of the reactor trip.

These results showed good agreement between SIMULATE-3K and ARROTTA for the reference analyses. The transient power response and time of peak power statepoint agreed well. The nodal peak powers agreed well with the exception of the EOC HZP case. This was due to the unique combination of adjustments which had to be made for this case to duplicate ARROTTA's initial conditions as specified in Table 6-3. In conclusion, these comparisons demonstrate the acceptability of the physical and numerical models within the SIMULATE-3K code for application in analyses of the REA for McGuire and Catawba Nuclear Station.

6.6.2 Rod Ejection Nuclear Analysis

The current approved methodology for the REA utilizes the computer code ARROTTA (Reference 6-12) to perform nuclear analysis calculations. This section describes the use of SIMULATE-3K for the nuclear analysis calculations for the REA analyses as described in topical report, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters" DPC-NE-3001 (Reference 6-2).

6.6.2.1 REA Analytical Approach

The complexity of the core and system response to a rod ejection event requires the application of a sequence of computer codes. The rapid core power excursion is simulated with a three-dimensional transient neutronic and thermal-hydraulic model using the SIMULATE-3K code (Reference 6-9). [ ] The resulting transient core power distribution results are then input to VIPRE-01 (Reference 6-20) core thermal-hydraulic models. The VIPRE models calculate the fuel temperatures, the allowable power peaking to avoid exceeding the DNBR limit, and the core coolant expansion rate. The allowable power peaking is then used along with a post-ejected condition fuel pin census to determine the percentage of pins exceeding the DNB limit. The coolant expansion rate is input to a RETRAN-02 (Reference 6-21) model of the Reactor Coolant System to determine the peak pressure resulting from the core power excursion.

The remainder of this section will address how the nuclear analyses of the REA will be performed with SIMULATE-3K. The basic methodology, as described in Reference 6-2, remains unchanged with the exception of minor differences between SIMULATE-3K and ARROTTA which are discussed in the following section.

6.6.2.2 <u>SIMULATE-3K Nuclear Analysis</u>

The response of the reactor core to the rapid reactivity insertion from the control rod ejection is simulated with SIMULATE-3K code (Reference 6-9). SIMULATE-3K computes a three-dimensional power distribution (in rectangular coordinates) and reactivity or power level for both static and transient applications. SIMULATE-3K includes a prediction of individual pin powers. Modifications are made to the core model to ensure conservative results. These changes produce a rod ejection model which produces limiting results that are expected to bound future reload cycles. A complete description of the SIMULATE-3K code is discussed in Section 6.6.1.3 and Reference 6-9.

The SIMULATE-3K model geometry will typically be [ ] per fuel assembly in the radial direction. The axial nodalization depends on the fuel assembly design, such as whether or not axial blanket fuel is being modeled. The number of axial levels is chosen to accurately describe the axial characteristics of the fuel. For current fuel designs, a typical axial nodalization of 24 equal length fuel nodes in the axial direction is used. The SIMULATE-3K model explicitly calculates neutron leakage from the core by use of reflector nodes in the radial direction beyond the fuel region and in the axial direction above and below the fuel column stack. Required fuel and reflector cross sections are developed consistent with the methodology approved for SIMULATE-3P in topical report DPC-NE-1004A (Reference 6-8).

SIMULATE-3K is used to calculate the core power level and nodal power distribution versus time during the rod ejection transient. [

] This information is used by VIPRE to determine the fuel enthalpy, the percentage of the fuel pins exceeding the DNB limit, and the coolant expansion rate.

6.6.2.2.1 Initial Conditions

The SIMULATE-3K rod ejection analysis is analyzed at four statepoints; beginning-of-cycle (BOC) at hot zero power (HZP) and hot full power (HFP) and end-of-cycle (EOC) at HZP and HFP. The conservatisms applied to the rod ejection analysis as described in Reference 6-2 are implemented based on the methodology described in Reference 6-9 and are expected to bound future reload cycles Initial conditions for SIMULATE-3K different than those discussed in Reference 6-2 are described below.

[ ]

The moderator temperature coefficient (MTC) is also adjusted to conservative values at BOC or EOC which bounds the magnitude of the MTC expected in a reload core. The MTC is adjusted in SIMULATE-3K by [ ] This adjustment is made via the equation from SIMULATE-3K (Reference 6-9);

[

]

Similar adjustments are made to yield conservative rod worth for control rod withdrawal and rod worth for control rod insertion.

The Doppler (or fuel) temperature coefficient (DTC) is important to this transient because the negative reactivity from the increased fuel temperature is the only effect that limits the power excursion and starts to shut down the reactor. The DTC is adjusted to a conservative value which bounds the magnitude of the DTC expected in a reload core. The DTC is adjusted in SIMULATE-3K by [

]

The effective delayed neutron fraction (β) and the ejected rod worth both determine the transient power response of the reactor. The peak power level obtained during the transient will increase for small values of β and larger values of the ejected rod worth. The ejected rod worth and β are adjusted to conservative values which bound values expected for a reload core. The ejected rod worth is adjusted in SIMULATE-3K by [

] β can be adjusted in SIMULATE-3K by [

]

[ ]

or β can be adjusted by inputting a single set of delayed neutron parameters to be used for all fueled nodes.

The combined effect of all these changes to the SIMULATE-3K model is to produce a model that is expected to bound future reload cycles for both McGuire and Catawba Nuclear Stations.

6.6 2.2.2 Boundary Conditions

The fuel and core thermal-hydraulic boundary conditions are established using conservative assumptions. Boundary conditions for initial power, core flow, inlet temperature, reactor pressure, and fission power fraction in the coolant are selected to yield conservative results.

The reactor trip signal is generated when the third highest excore channel reaches either [ ] for the HZP cases or [ ] for the HFP cases. This modeling is based on a single failure of the highest channel and a two-out-of-the-remaining-three trip coincidence logic. [

] in SIMULATE-3K (Reference 6-9) can be used. The remaining control rods fall into the reactor assuming a conservative trip delay after the trip signal is generated.

During the reactor trip, the ejected rod and a second rod with the highest worth are assumed not to fall into the reactor. To conservatively model the reactor trip, not all of the control rod banks are allowed to drop, and some of the banks that are dropped have their worth reduced by a cross section adjustment. The rod worth adjustment is made in SIMULATE-3K by [

] based on Eq. 6.1. Also, negative reactivity inserted due to the reactor trip is not allowed to exceed the conservative trip reactivity curve. The integral worth of the falling control rods is computed for several different axial positions of the rods at the initial conditions. [

]

6.6.2.3 <u>Core Thermal-Hydraulic Analysis</u>

The core thermal-hydraulic analyses use the VIPRE-01 code for the calculation of peak fuel enthalpy, DNBR, and the coolant expansion rates for various initial and boundary conditions postulated for the REA transient. All input to the core thermal-hydraulic analyses once supplied by ARROTTA can now be supplied by SIMULATE-3K. The nuclear analysis input boundary conditions supplied by SIMULATE-3K for the thermal-hydraulic analyses are [

]

6.6.2.3.1 Fuel Temperature and Peak Fuel Enthalpy

The calculation of the transient maximum hot spot average fuel temperature and the maximum radial average fuel enthalpy requires the following input boundary conditions to be supplied by SIMULATE-3K: [

] This information is consistent with that provided by ARROTTA in Reference 6-2.

6.6.2.3.2 DNBR Evaluation

The percentage of the core experiencing DNBR is calculated as explained in Reference 6-2 except SIMULATE-3K results are used instead of ARROTTA results. For the HFP REA cases,

[

] For a given axial power profile, the maximum pin radial peak can be determined such that DNB would not occur during the transient. These DNB limits are referred to as maximum allowable radial peaks (MARP) limits. A fuel pin census is then performed to determine the number of fuel pins in the core that exceed the power peaking limit.

[

]

6.6.2.3.3 Coolant Expansion Rate

The calculation of the coolant expansion rate requires the following input boundary conditions to be supplied by SIMULATE-3K: [

] This SIMULATE-3K information is input to VIPRE to calculate the flow rate in each channel during the transient. Using the VIPRE channel flow rates, the total coolant expansion rate can be calculated. This total coolant expansion rate is input to the RETRAN plant transient model for simulating the resulting pressure response. This SIMULATE-3K information is consistent with that provided by ARROTTA in Reference 6-2.

6.6.2.4 Cycle-Specific Evaluation

Due to the conservative assumptions and modeling used in the SIMULATE-3K model, it is anticipated that for reload cores, no new SIMULATE-3K cases will be necessary. The determination as to whether the existing SIMULATE-3K cases remain bounding will be made by performing a cycle-specific reload check of the key physics input parameters as described in Reference 6-2. These parameters will be calculated using steady-state neutronics codes approved by the NRC for reload design. If the key physics parameters remain bounded then no new SIMULATE-3K analyses are necessary; otherwise, an evaluation, reanalysis, or re-design of the reload core will be performed.

For the HFP REA cases, a DNB pin census will be performed for the reload cycle, as described in Section 4.7 of Reference 6-2, with the radial power information being calculated with an NRC approved steady-state neutronics code. The HZP REA cases are bounded by the HFP cases in the offsite dose analyses, and therefore, a pin census is not required. The ejected rod worth shall be calculated with the fuel and moderator temperatures frozen in the pre-ejected condition or uniform throughout the core (either method will generate conservative results). [

] The power distribution with the ejected rod out will be used for the DNB pin census. The calculated percent fuel failure due to DNB will be compared for each cycle to the fuel failure limit assumed in the dose calculation. If the cycle specific value is less than the limit, then the existing safety analysis is still valid. Otherwise, an evaluation, a new dose calculation, reanalysis, or new reload design will be performed as appropriate.

6.6.2.5 Mixed Cores

The Westinghouse fuel is expected to behave neutronically similar to that of the Framatome Cogema Fuels Mark-BW fuel. The steady-state cycle-specific checks will verify that all key physics parameters remain valid and the DNB census will use the appropriate CHF correlations for the various fuel types present in the core.

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6-37 NTD-NRC-95-4520, Letter from N. J. Liparulo (Westinghouse) to W. T. Russel (USNRC), "Change in Methodology for Execution of BASH Evaluation Model", August 29, 1995.

6-38 WCAP-10484-P-A Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood", September 1993.

6-39 NSD-NRC-99-5839, "1998 Annual Notification of Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10CFR50.46 (a)(3)(ii)". 1

Table 6-1

Rod Ejection ARROTTA Results

Parameter	BOC HZP	BOC HFP	EOC HZP	EOC HFP
Time of peak power, sec	0 286	0.077	0.173	0 080
Peak power level, % of full power	1880	138	5139	155
Peak nodal power relative to core average	7.99	3.44	16.40	3.96
Time that trip setpoint reached, sec	0.246	0 061	0.155	0.057
Time for beginning of trip rod motion	0.746	0.561	0 655	0.557

Table 6-2

Rod Ejection SIMULATE-3K Results

Parameter	BOC HZP	BOC HFP	EOC HZP	EOC HFP
Time of peak power, sec	0 296	0.076	0 187	0 083
Peak power level, % of full power	1884	133	5280	154
Peak nodal power relative to core average	7.127	3 508	12.997	3 605
Time that trip setpoint reached, sec	0 246	0 061	0.155	0.057
Time for beginning of trip rod motion	0.746	0 561	0 655	0.557

Table 6-3

Rod Ejection Transient Kinetics Input Parameters

Parameter	Computer Code	BOC HZP	BOC HFP	EOC HZP	EOC HFP
Ejected Rod Worth, pcm	ARROTTA	720	201	900	196
MTC (pcm/°F)	ARROTTA	+7.06	+0.05	-9 45	-9.73
DTC (pcm/°F)	ARROTTA	-0.90	-0.90	-1.19	-1.19
Delayed Neutron Fraction, β	ARROTTA	0.0055	0.0055	0 0040	0.0040
Ejected Rod Worth, pcm	SIMULATE-3K	721	203	900	197
MTC (pcm/°F)	SIMULATE-3K	+7.00	+0.08	-10.09	-10.09
DTC (pcm/°F)	SIMULATE-3K	-0.90	-0.90	-1.20	-1.20
Delayed Neutron Fraction, β	SIMULATE-3K	0.0055	0.0055	0.0040	0.0040

Figure 6-1

Reference Core Loading Information

	H	G	F	E	D	C	B	A
8	14 4.10 24/2.5 P 43.165 58.124	17 4.40 24/2.5 0 21.143	15 4.15 24/3.0 P 33.53 51 68	17 4.15 24/3.0 0 22 606	15 4.15 24/3.0 P 33 452 51 893	16 4.40 0 15.245 36.447	16 4 40 24/2 5 P 21.428 40.26	17 4 40 0 0 15.204
9		14 4.40 0 36.418 53.838	17 4.15 24/3.0 0 22.308	16 4.15 24/3.0 P 22.295 43.375	17 4.15 24/3 0 0 22.364	16 4.15 24/2.5 P 22.495 42.515	17 4 40 24/3.0 0 19.812	16 4.15 24/3.0 P 22.41 33.733
10			16 4.15 24/3.0 P 22.583 43.611	17 4.15 24/3.0 0 22 605	15 4.15 24/3 0 P 32 304 51 037	17 4.15 24/3 0 0 21.883	16 4.40 12/2 0 P 18.423 36 579	16 4.15 24/3 0 P 22 588 32 353
11				15 4.15 24/3.0 P 35.265 53.466	17 4.15 24/2 5 0 22 485	16 4.40 24/2.5 P 19.774 40 535	17 4 40 12/2.0 0 18.483	15 4.40 24/3.0 P 38.018 44.405
12					16 4 15 24/3.0 P 21 824 41.811	17 4.40 24/2 5 0 19.727	16 4 40 24/3.0 P 19 812 32 032	
13						16 4.15 24/3 0 P 21.823 35.106	15 4.40 24/3 0 P 32.088 38.42	

Key:
Batch number
Enrichment
Number of BP fingers/ wt% of boron in BP (P means BPs pulled)
BOC exposure (GWD/MT)
EOC exposure (GWD/MT)

Batch 17 is fresh fuel
Batch 16 is starting its second burn
Batches 14 and 15 are starting their third burn

Figure 6-2

FSAR Section 15 4 8 - Control Rod Ejection

BOC HFP Core Power vs. Time

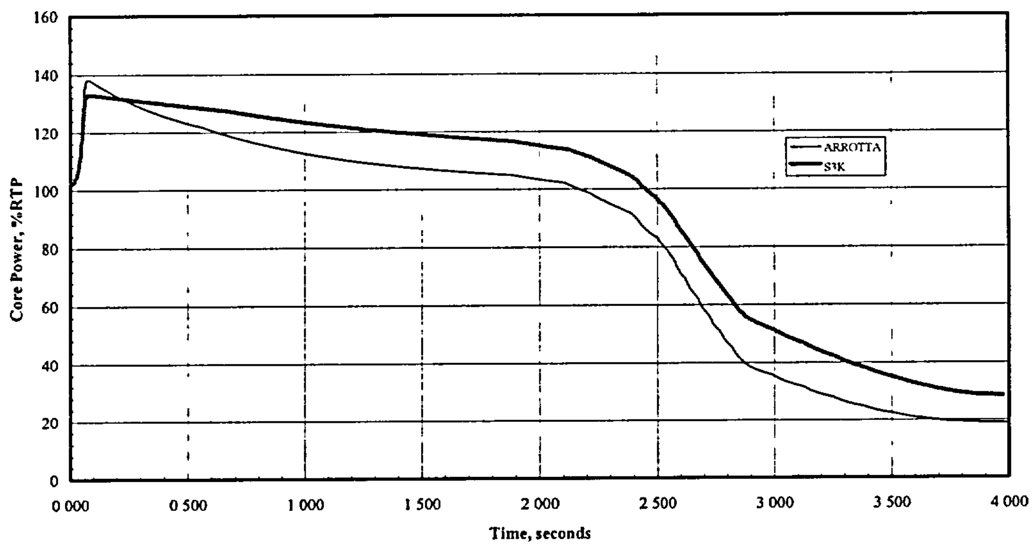


Figure 6-3

FSAR Section 15.4.8 - Control Rod Ejection

BOC HZP Core Power vs Time

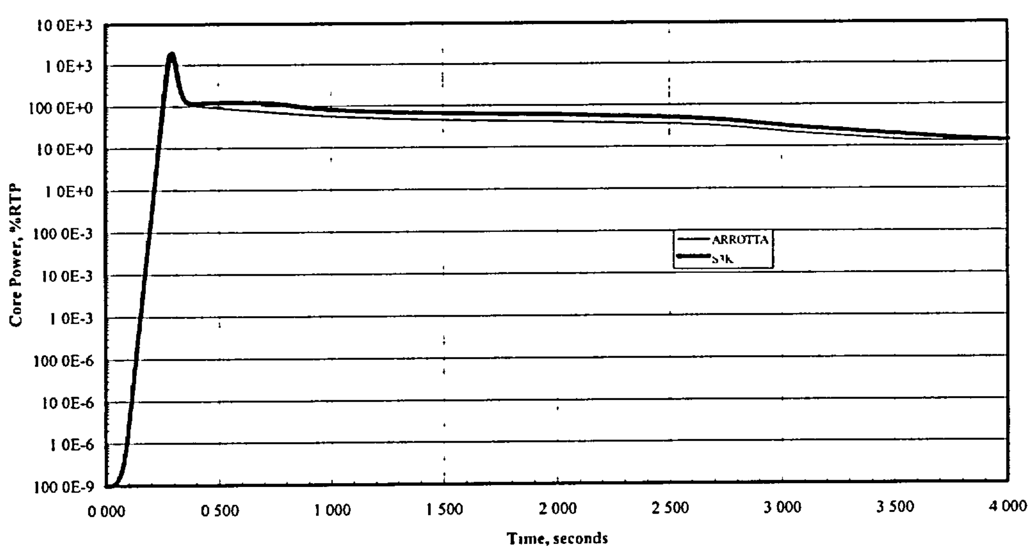


Figure 6-4

FSAR Section 15 4.8 - Control Rod Ejection

EOC HFP Core Power vs. Time

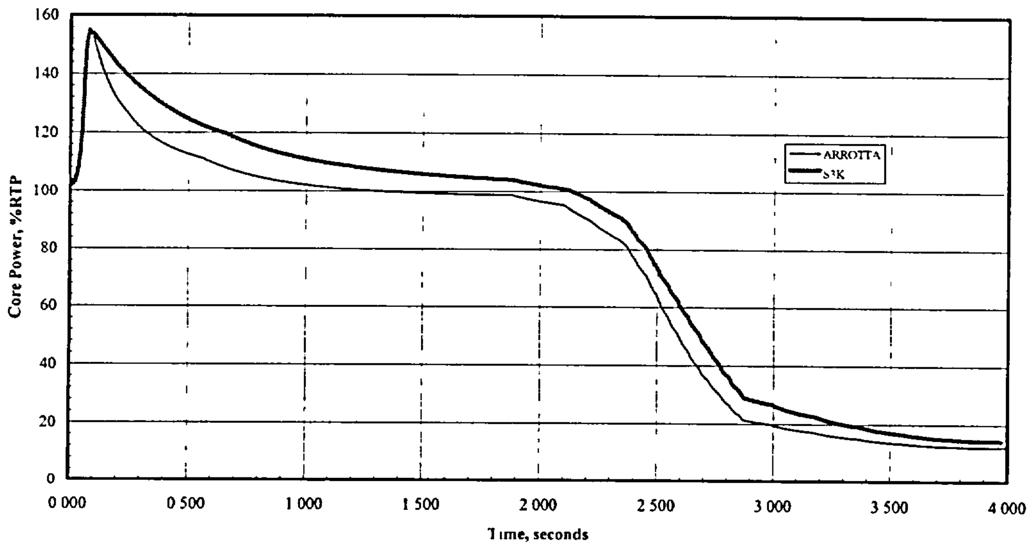
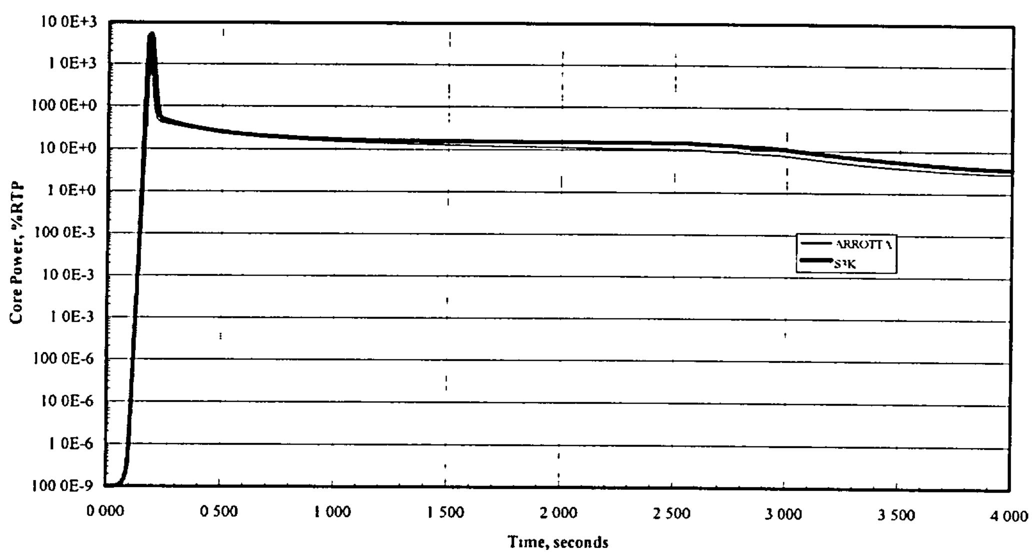


Figure 6-5

FSAR Section 15.4 8 - Control Rod Ejection

EOC HZP Core Power vs. Time



7.0 FUEL ASSEMBLY REPAIR AND RECONSTITUTION

The reconstitution of fuel assemblies is a routine occurrence during refueling outages in light water reactors. This is due to the concerted effort on the part of utilities to maintain zero fuel defects during cycle operation. This zero defect goal requires aggressive programs in two areas. First, all reasonable measures must be taken in the design and manufacturing of fuel assemblies to prevent any type of known failure mechanism. Secondly, failures that do occur during operation should be identified and the failed fuel rods removed before subsequent cycles.

Duke Power's primary replacement candidate for use in reconstitution is a fuel rod that contains pellets of natural uranium dioxide (UO_2). Aside from enrichment, this rod is the same in design and behavior as a standard fuel rod and is analyzed using standard approved methods. If local grid structural damage exists, the use of a natural UO_2 replacement rod is not the preferred alternative and solid filler rods made of stainless steel, zircaloy, or ZIRLO™ would be used.

The NRC-approved DPC-NE-2007 topical report, Reference 7-1, describes the methodology and guidelines Duke Power uses to support fuel assembly reconstitution with filler rods. The guidelines were developed to ensure acceptable nuclear, mechanical, and thermal-hydraulic performance of reconstituted fuel assemblies. Specific results were provided in the report for the Mark-B and Mark-BW fuel designs with licensed codes. As stated in DPC-NE-2007, the methodology would be applicable if different fuel designs or codes are licensed by Duke Power.

Duke Power will use the same licensing and analysis approach for reconstitution of the RFA design at McGuire and Catawba. The methodology described in Reference 7-1 will be used along with the licensed codes and correlations described in this report. These codes will be used to analyze reconstitution with filler rods for acceptable nuclear, mechanical, and thermal-hydraulic performance. For a reload core using reconstituted Westinghouse fuel, Westinghouse will evaluate the effects of the reconstitution on the LOCA analysis using the methodology given in Reference 7-2.

As discussed in Reference 7-2, Westinghouse has reviewed the criteria specified in Standard Review Plan 4.2 (Reference 7-3) and determined that the only fuel assembly mechanical criteria impacted by reconstitution are.

1) fuel assembly holddown force, and

2) fuel assembly structural response to Seismic/LOCA loads.

Westinghouse evaluated both of these criteria and concluded that the reconstituted fuel assembly designs are acceptable for both normal and faulted condition operations.

7.1 <u>References</u>

7-1 DPC-NE-2007P-A, Duke Power Company Fuel Reconstitution Analysis Methodology, October 1995.

7-2 W. H. Slagle (Ed.), "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology", WCAP-13060-P-A, July 1993.

7-3 "Section 4.2, Fuel System Design", Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, NUREG-0800, Rev. 2, US Nuclear Regulatory Commission, July 1981.

8.0 IMPROVED TECHNICAL SPECIFICATION CHANGES

Since the RFA design will be first implemented for Catawba 2 Cycle 11, changes to the current McGuire and Catawba Technical Specifications are not necessary. However, the following changes to the Improved Technical Specifications (ITS), originally submitted to the NRC on May 27, 1997 with numerous supplements submitted thereafter, are necessary to license the RFA design.

Figure 2.1.1-1 (Reactor Core Safety Limits - Four Loops in Operation) will be modified to delete the 2455 psia safety limit line. This line is the current upper bound pressure at which power operation is permitted and is dependent on the pressure range of the critical heat flux (CHF) correlation used in DNBR analyses. The critical heat flux correlation of the resident Mark-BW fuel is applicable up to a pressure of 2455 psia. Deleting the 2455 psia safety limit line is necessary due to implementation of the WRB-2M CHF correlation for the RFA design, which has an upper range of 2425 psia (Reference 8-1). The 2400 psia safety limit line will remain as the upper bound safety limit line because it is within the range of the CHF correlations for the RFA and Mark-BW fuel designs.

ITS 4.2.1 will be revised to add ZIRLO™ cladding to the fuel assembly description. ITS 5.6.5 will be revised to add this topical report to the list of approved methodologies for McGuire and Catawba.

The nuclear design related Technical Specification limits were reviewed for transition and full core reloads comprised of the Westinghouse RFA design. The power distribution Technical Specifications for Fq and FΔH have a 2% factor in each specification's surveillances which is used to account for the possible increase in Fq and FΔH between flux maps. This factor for IFBA cores will have to be burnup dependent because of the increased burnout rate of the integral burnable absorber relative to the lumped burnable absorbers. The technical justification for this proposed change is given in Sections 8.1 and 8.2.

8.1 Technical Justification for Surveillance Requirement SR 3.2.1.2 and SR 3.2 1.3

$F_q(x,y,z)$ is measured periodically using the incore detector system to ensure that the value of the total peaking factor, F_q-RTP, assumed in the accident analysis is bounding. The frequency requirement for this measurement is 31 effective full power days (EFPD). To account for the possibility that $F_q(x,y,z)$ may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If the extrapolation of the measurement indicates that the $F_q(x,y,z)$ measurement would exceed the $F_q(x,y,z)$ limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on available margin, or the $F_q(x,y,z)$ measurement would be increased by an appropriate penalty (currently 1.02) and compared against the $F_q(x,y,z)$ operational and RPS surveillance limits to ensure allowable total peaking limits are not exceeded.

Technical Specification surveillances SR 3.2.1.2 and SR 3.2.1.3 currently specify that the $F_q(x,y,z)$ measurement be increased by 1.02. This value was chosen because it bounded the maximum $F_q(x,y,z)$ increase in typical reload cores. However, for reactor cores containing integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. This penalty can be incorporated into either the $M_q(x,y,z)$ or $M_c(x,y,z)$ margin factors, or be provided in tabular form as a function of burnup.

It is proposed that this penalty factor be moved to the Core Operating Limits Report (COLR) in tabular form to facilitate cycle specific updates. Table 8-1 provides an example burnup dependent penalty factor that would replace the current 1.02 value. For burnup ranges where the increase in Fq over the 31 EFPD surveillance interval is less than 2.0%, the current 1.02 penalty factor will be maintained.

Relocation of this penalty factor to the Core Operating Limits Report (COLR) was included in TSTF-98 (Technical Specification Task Force), Revision 2. This generic change to NUREG-1431 was approved by the NRC in April 1998.

8.2 Technical Justification for Surveillance Requirement SR 3.2.2.2

The nuclear enthalpy rise hot channel factor, $F_{\Delta H}(x,y,)$, is measured periodically using the incore detector system to ensure that fuel design criteria are not violated and accident analysis assumptions are not violated. The frequency requirement for this measurement is 31 effective full power days (EFPD). To account for the possibility that $F_{\Delta H}(x,y)$ may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If the extrapolation of the measurement indicates that the $F_{\Delta H}(x,y)$ measurement would exceed the $F_{\Delta H}(x,y)$ surveillance limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on available margin, or the $F_{\Delta H}(x,y)$ measurement would be increased by an appropriate penalty (currently 1.02) and compared against the $F_{\Delta H}(x,y)$ surveillance limit to ensure allowable peaking limits are not exceeded.

Technical Specification surveillance SR 3.2.2.2 currently specifies that the $F_{\Delta H}(x,y)$ measurement be increased by 1.02. This value was chosen because it bounded the maximum $F_{\Delta H}(x,y)$ increase in typical reload cores. However, for reactor cores containing integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. This penalty can be incorporated into either the $F_{\Delta H}(x,y)$ surveillance limit or be provided in tabular form as a function of burnup.

It is proposed that this penalty factor be moved to the Core Operating Limits Report (COLR) in tabular form to facilitate cycle specific updates. Table 8-2 provides an example burnup dependent penalty factor that would replace the current 1.02 value. For burnup ranges where the increase in $F_{\Delta H}(x,y)$ over the 31 EFPD surveillance interval is less than 2.0%, the current 1.02 penalty factor will be maintained.

Relocation of this penalty factor to the Core Operating Limits Report (COLR) was included in TSTF-98 (Technical Specification Task Force), Revision 2. This generic change to NUREG-1431 was approved by the NRC in April 1998.

8.3 References

8-1 WCAP-15025-P, Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, Westinghouse Energy Systems, February 1998.

Table 8-1

$F_q(x,y,z)$ Margin Decrease Over 31 EFPD Surveillance Interval

(Typical Values)

Burnup (EFPD)	$F_q(x,y,z)$ Margin Decrease Penalty Factor
4	2.00 %
12	2.28 %
25	3.31 %
50	3.45 %
100	3.24 %
200	2.00 %
EOC	2.00 %

Note: Linear interpolation of the penalty factors is adequate for surveillances performed at intermediate burnups.

Table 8-2

$F_{\Delta H}(x,y)$ Margin Decrease Over 31 EFPD Surveillance Interval

(Typical Values)

Burnup (EFPD)	$F_{\Delta H}(x,y)$ Margin Decrease Penalty Factor
4	2.00 %
12	2.40 %
25	2.50 %
50	2.60 %
100	2.15 %
200	2.00 %
EOC	2.00 %

Note: Linear interpolation of the penalty factors is adequate for surviellances performed at intermediate burnups.

Section F

RAI Letters and Responses, Original Report

1. December 9, 1998, NRC RAI letter (Catawba), P. S. Tam to G. R. Peterson
2. January 5, 1999, NRC RAI letter (McGuire), F. Rinaldi to H. B. Barron
3. January 28, 1999, letter responding to NRC RAI, M. S. Tuckman to NRC
4. April 7, 1999, letter responding to NRC RAI (Question 11), M. S. Tuckman to NRC

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

December 9, 1998

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

SUBJECT: CATAWBA NUCLEAR STATION - REQUEST FOR ADDITIONAL INFORMATION ON YOUR AMENDMENT REQUEST OF JULY 22, 1998 (TAC NOS. MA2359 AND MA2361)

Dear Mr. Peterson:

By letter dated July 22, 1998, Duke Energy Corporation (DEC) proposed to amend the Catawba Nuclear Station, Units 1 and 2, Technical Specifications to permit use of Westinghouse fuel. Topical Report DPC-NE-2009P/ DPC-NE-2009, "Duke Power Company Westinghouse Fuel Transition Report" was part of DEC's submittal. The original submittal was supplemented by letter dated October 22, 1998.

The staff is reviewing DEC's submittals, and has found that additional information is needed to complete the review (enclosed). We have discussed this request for additional information with Mr. Steve Warren of your staff, and agreed that the response would be due on or before January 31, 1999. We will be glad to discuss the questions with you upon your request.

Sincerely,

Peter S. Tam

Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure: Request for Additional Information

cc w/encl: See next page

REQUEST FOR ADDITIONAL INFORMATION

DPC-NE-2009, "DUKE POWER COMPANY

WESTINGHOUSE FUEL TRANSITION REPORT"

(Reference: Letter, M. S. Tuckman to NRC, July 22, 1998)

1. Section 3.2 of DPC-NE-2009P states that conceptual transition core designs using the Robust Fuel Assembly (RFA) design have been evaluated and show that current reload limits remain bounding with respect to key physics parameters, and that in the event that one of the key parameters is exceeded, the evaluation process described in DPC-NE-3001-PA would be performed.

   (a) Describe the evaluation and the result of the conceptual transition core design.

   (b) Based on the statement, it appears that the evaluation process described in DPC-NE-3001-PA will not be performed unless one of the key parameters is exceeded. Without actual analysis of the RFA transitional or full cores, how is it determined that any of the key parameters is exceeded?

2. To demonstrate that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties in DPC-NE-1004-PA are applicable to the RFA design, Section 3.2 cites the analyses performed using Sequoyah Unit 2 Cycles 5, 6, and 7, as well as a 10 CFR 50.59 unreviewed safety question (USQ) evaluation. It is stated that the Sequoyah cores were chosen because they are similar to McGuire and Catawba and contained both Integral Fuel Burnable Absorber (IFBA) and Wet Annular Burnable Absorber fuel. Table 3-1 provides the statistical analysis results of nuclear uncertainty factors, which show they are bounded by the uncertainty factors of DPC-NE-1004A.

   (a) Describe any difference between the Catawba RFA cores and the Sequoyah cores analyzed. Describe why these differences would not affect the applicability of the analyses of the Sequoyah cores to Catawba.

   (b) Provide the comparison of the analysis results with measured data of boron concentrations, rod worths, and isothermal temperature coefficients.

   (c) Describe the details and results of the 10 CFR 50.59 USQ evaluation.

3. Section 3.2 states that (1) in all nuclear design analysis, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores, and (2) when establishing operating and reactor protection system limits (i.e., loss-of-coolant accident (LOCA) kw/ft, departure from nucleate boiling (DNB), containment failure mode, transient strain), the fuel specific limits or a conservative overlay of the limits are used. Please elaborate on the mixed core model for nuclear design analyses, and how fuel-specific limits are used.

Enclosure

4. Section 5.2 states that in using the VIPRE-01 code for the reactor core thermal-hydraulic analysis, the reference power distribution based on a 1.60 peak pin from DPC-NE-2004P-A, Revision 1, was used.

   (a) The report states that this reference pin power distribution "was" used. Will it be used for future RFA reload analyses?

   (b) Does the reference pin power distribution used in the core thermal-hydraulic analyses bound all power distribution for the RFA cores for future reload cycles?

5. Section 5.2 states that in the thermal-hydraulic analysis of the RFA design using VIPRE-01, the two-phase flow correlations will be changed from the Levy subcooled void correlation and the Zuber-Findlay bulk void correlation to the EPRI subcooled and bulk void correlations, respectively. While the sensitivity study provided in the report shows a minimal difference of 0.1 percent between the minimum DNB ratios (DNBRs) of 51 RFA critical heat flux (CHF) test data points calculated with both sets of correlations, it was stated in DPC-NE-2004 that the Levy/Zuber-Findlay combination compared most favorably with the Mark-BW test results as the DNBRs of the tests calculated with this combination yielded conservative results relative to the EPRI correlations.

   (a) Discuss whether the EPRI correlations will be used for the RFA design only, or if they will also be used for the Mark-BW design.

   (b) If the EPRI correlations will also be used for Mark-BW design, provide justification for their use.

   (c) If the Levy/Zuber-Findlay correlations will continue to be used for the Mark-BW fuel design, discuss how the VIPRE-01 code will be used to analyze transient mixed cores having both Mark-BW and RFA fuel designs.

6. Section 5.7 describes the use of a transition 8-channel RFA/Mark-BW core model to determine the impact of the geometric and hydraulic differences between the resident Mark-BW fuel and the RFA design, and determine a conservative DNBR penalty to be applied for the transition cores. Table 5-4 presented the statistical DNBRs for the 500 and 5000 case runs for various statepoints including the transition core case of the most limiting statepoint 12. The statistical design limit is chosen to bound both the full RFA cores and RFA/Mark-BW transition cores for the 5000 case runs.

   (a) Why is the statistical design limit value proprietary information?

   (b) With respect to the statistical core design methodology, describe how the uncertainties of the CHF correlation and the VIPRE code/model are propagated with the uncertainties of the selected parameters of each statepoint for the calculation of the statistical DNBR for each statepoint in Table 5-4.

   (c) With the statistical design limit specified in Section 5.7, is it your intention to use a full core of RFA in the thermal-hydraulic analysis for the transition core without the transition core DNBR penalty factor?

7. Section 2.0 states that the RFA is designed to be mechanically and hydraulically compatible with the Mark-BW fuel. Table 2.1 provides a comparison of the basic-design parameters of the two fuel designs, but does not provide a comparison of the hydraulic characteristics of spacer grids. Section 5.2 states that the VIPRE-01 core thermal-hydraulic analyses were performed with applicable form loss coefficients according to the vendor. Table 5.1 provides general RFA fuel specifications and characteristics without the hydraulic characteristics of the spacer grids.

   (a) Provide comparisons for the thickness, height, and form loss coefficients of the RFA and Mark-BW fuel spacer grids, including mixing-vane and nonmixing vane structural grids, and intermediate flow mixing grids.

   (b) Provide the form loss coefficients of the spacer grids used in the analyses and in the RFA CHF test assemblies if they are different from the values described in item (a).

   (c) Describe the procedures to ensure that the form loss coefficients of the RFA grids are comparable to those used in the statistical core design analysis and the CHF tests so that both the WRB-2M CHF correlation DNBR limit and the statistical core design limit are valid.

8. Section 6.1.3 states that the thermal-hydraulic methodology described in DPC-NE-3000-PA, Revision 1, with a simplified core model will be used for thermal-hydraulic analysis of the Updated Final Safety Analysis Report Chapter 15 non-LOCA transients and accidents for the RFA design. It also states that (1) no transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses, (2) the simplified core model of DPC-NE-3000-PA used for transient analyses was originally developed with additional conservatism over the 8-channel model used for steady-state analyses to specifically minimize the impact of changes in core reload design methods or fuel assembly design, and (3) should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly.

   (a) Explain what additional conservatism is provided in using the simplified core model of DPC-NE-3000-PA.

   (b) What is the criterion/criteria used to determine if transition core transient analyses are warranted? How would it be determined that the criteria have been exceeded without RFA transition core analyses?

9. Regarding rod ejection analysis using SIMULATE-3K, Section 6.6.2.2.1 states that the transient response is made more conservative by increasing the fission cross sections in the ejected rod location and in each assembly and by applying "factors of conservatism" in the moderator temperature coefficient, control rod worths for withdrawal and insertion, Doppler temperature coefficient, effective delay neutron fraction, and ejected rod worth, etc.

   (a) What are the values of the multiplication factors used for fission cross sections, and how are they determined?

(b) How are the input multipliers "VAL" in Equations 6.1 and 6.2 determined? Does "VAL" have a different value for different parameters, such as MTC or DTC? What are the values for these VALs?

(c) In Equation 6.1, the X's are described as "moderator temperatures." Should they be moderator temperature coefficients?

10. Regarding the SIMULATE-3K code, there is an optional "frequency transform" approach, under the "Temporal Integration Models," that can be chosen to separate the fluxes into exponential time varying and predominately spatial components, thus accelerating convergence of the transient neutronic solution and preserving accuracy on a coarser time mesh (see Page 5, Ref. 6-9).

    (a) What determines when the "frequency transform" approach should be used?

    (b) What are the consequences of exercising (or not exercising) this option? Please provide technical justification and comparisons of results.

11. The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current Catawba cores not having the RFA design. For example, DPC-NE-1004A, DPC-NE-2011-PA, DPC-NF-2010A, and DPC-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002-A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation models described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC safety evaluation reports in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions, and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.

12. Section 8.0 states that TS Figure 2.1.1-1 for the reactor core safety limits will be modified by deleting the 2455 psia safety limit line and making the 2400 psia safety limit line as the upper bound pressure allowed for power operation. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2400 psia safety limit line is within the range of the CHF correlations for the Mark-BW and RFA fuel designs.

    However, the safety limit lines in Figure 2.1.1-1 were based on the CHF correlation for the Mark-BW fuel design, in addition to the hot leg boiling limit. Has an analysis been performed to ensure these safety limit lines bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design?

13. TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor F_q (x,y,z) and the enthalpy rise hot channel factor $F_{\Delta h}$ (x,y) to be measured periodically using the incore detector system to ensure that the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limits are not violated. To avoid the possibility that these hot channel factors may increase beyond their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the F_q (x,y,z) or $F_{\Delta h}$ (x,y) has decreased since the previous surveillance. For the reactor core containing the RFA fuel design with integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. Section 8.1 proposes to remove the 2 percent penalty value from these surveillance requirements and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide "typical values" for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors.

(a) Provide the actual values of the margin-decrease penalty factors, as well as the bases, for these values.

(b) Provide references for the approved methodologies used to calculate these values, and to be included in TS 5.6.5 as a part of acceptability for COLR.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
January 5, 1999

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

SUBJECT: MCGUIRE NUCLEAR STATION - REQUEST FOR ADDITIONAL INFORMATION ON YOUR AMENDMENT REQUEST OF JULY 22, 1998 (TAC NOS. MA2411 AND MA2412)

Dear Mr. Barron:

By letter dated July 22, 1998, Duke Energy Corporation (DEC) proposed to amend the McGuire Nuclear Station, Units 1 and 2, Technical Specifications to permit use of Westinghouse fuel. Topical Report DPC-NE-2009P/DPC-NE-2009, "Duke Power Company Westinghouse Fuel Transition Report" was part of DEC's submittal. The original submittal was supplemented by letter dated October 22, 1998.

The staff is reviewing DEC's submittals, and has found that additional information is needed to complete the review (enclosed). We have discussed this request for additional information with Mr. Steve Warren of your staff, and agreed that the response would be due on or before January 31, 1999. We will be glad to discuss the questions with you upon your request.

Sincerely,

Frank Rinaldi

Frank Rinaldi, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosure: Request for Additional Information

cc w/encl: See next page

REQUEST FOR ADDITIONAL INFORMATION

DPC-NE-2009, "DUKE POWER COMPANY

WESTINGHOUSE FUEL TRANSITION REPORT"

(Reference: Letter, M. S. Tuckman to NRC, July 22, 1998)

1. Section 3.2 of DPC-NE-2009P states that conceptual transition core designs using the Robust Fuel Assembly (RFA) design have been evaluated and show that current reload limits remain bounding with respect to key physics parameters, and that in the event that one of the key parameters is exceeded, the evaluation process described in DPC-NE-3001-PA would be performed.

   (a) Describe the evaluation and the result of the conceptual transition core design.

   (b) Based on the statement, it appears that the evaluation process described in DPC-NE-3001-PA will not be performed unless one of the key parameters is exceeded. Without actual analysis of the RFA transitional or full cores, how is it determined that any of the key parameters is exceeded?

2. To demonstrate that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties in DPC-NE-1004-PA are applicable to the RFA design, Section 3.2 cites the analyses performed using Sequoyah Unit 2 Cycles 5, 6, and 7, as well as a 10 CFR 50.59 unreviewed safety question (USQ) evaluation. It is stated that the Sequoyah cores were chosen because they are similar to McGuire and Catawba and contained both Integral Fuel Burnable Absorber (IFBA) and Wet Annular Burnable Absorber fuel. Table 3-1 provides the statistical analysis results of nuclear uncertainty factors, which show they are bounded by the uncertainty factors of DPC-NE-1004A.

   (a) Describe any difference between the Catawba RFA cores and the Sequoyah cores analyzed. Describe why these differences would not affect the applicability of the analyses of the Sequoyah cores to Catawba.

   (b) Provide the comparison of the analysis results with measured data of boron concentrations, rod worths, and isothermal temperature coefficients.

   (c) Describe the details and results of the 10 CFR 50.59 USQ evaluation.

3. Section 3.2 states that (1) in all nuclear design analysis, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores, and (2) when establishing operating and reactor protection system limits (i.e., loss-of-coolant accident (LOCA) kw/ft, departure from nucleate boiling (DNB), containment failure mode, transient strain), the fuel specific limits or a conservative overlay of the limits are used. Please elaborate on the mixed core model for nuclear design analyses, and how fuel-specific limits are used.

Enclosure

4. Section 5.2 states that in using the VIPRE-01 code for the reactor core thermal-hydraulic analysis, the reference power distribution based on a 1.60 peak pin from DPC-NE-2004P-A, Revision 1, was used.

   (a) The report states that this reference pin power distribution "was" used. Will it be used for future RFA reload analyses?

   (b) Does the reference pin power distribution used in the core thermal-hydraulic analyses bound all power distribution for the RFA cores for future reload cycles?

5. Section 5.2 states that in the thermal-hydraulic analysis of the RFA design using VIPRE-01, the two-phase flow correlations will be changed from the Levy subcooled void correlation and the Zuber-Findlay bulk void correlation to the EPRI subcooled and bulk void correlations, respectively. While the sensitivity study provided in the report shows a minimal difference of 0.1 percent between the minimum DNB ratios (DNBRs) of 51 RFA critical heat flux (CHF) test data points calculated with both sets of correlations, it was stated in DPC-NE-2004 that the Levy/Zuber-Findlay combination compared most favorably with the Mark-BW test results as the DNBRs of the tests calculated with this combination yielded conservative results relative to the EPRI correlations.

   (a) Discuss whether the EPRI correlations will be used for the RFA design only, or if they will also be used for the Mark-BW design.

   (b) If the EPRI correlations will also be used for Mark-BW design, provide justification for their use.

   (c) If the Levy/Zuber-Findlay correlations will continue to be used for the Mark-BW fuel design, discuss how the VIPRE-01 code will be used to analyze transient mixed cores having both Mark-BW and RFA fuel designs.

6. Section 5.7 describes the use of a transition 8-channel RFA/Mark-BW core model to determine the impact of the geometric and hydraulic differences between the resident Mark-BW fuel and the RFA design, and determine a conservative DNBR penalty to be applied for the transition cores. Table 5-4 presented the statistical DNBRs for the 500 and 5000 case runs for various statepoints including the transition core case of the most limiting statepoint 12. The statistical design limit is chosen to bound both the full RFA cores and RFA/Mark-BW transition cores for the 5000 case runs.

   (a) Why is the statistical design limit value proprietary information?

   (b) With respect to the statistical core design methodology, describe how the uncertainties of the CHF correlation and the VIPRE code/model are propagated with the uncertainties of the selected parameters of each statepoint for the calculation of the statistical DNBR for each statepoint in Table 5-4.

   (c) With the statistical design limit specified in Section 5.7, is it your intention to use a full core of RFA in the thermal-hydraulic analysis for the transition core without the transition core DNBR penalty factor?

7. Section 2.0 states that the RFA is designed to be mechanically and hydraulically compatible with the Mark-BW fuel. Table 2.1 provides a comparison of the basic-design parameters of the two fuel designs, but does not provide a comparison of the hydraulic characteristics of spacer grids. Section 5.2 states that the VIPRE-01 core thermal-hydraulic analyses were performed with applicable form loss coefficients according to the vendor. Table 5.1 provides general RFA fuel specifications and characteristics without the hydraulic characteristics of the spacer grids.

   (a) Provide comparisons for the thickness, height, and form loss coefficients of the RFA and Mark-BW fuel spacer grids, including mixing-vane and nonmixing vane structural grids, and intermediate flow mixing grids.

   (b) Provide the form loss coefficients of the spacer grids used in the analyses and in the RFA CHF test assemblies if they are different from the values described in item (a).

   (c) Describe the procedures to ensure that the form loss coefficients of the RFA grids are comparable to those used in the statistical core design analysis and the CHF tests so that both the WRB-2M CHF correlation DNBR limit and the statistical core design limit are valid.

8. Section 6.1.3 states that the thermal-hydraulic methodology described in DPC-NE-3000-PA, Revision 1, with a simplified core model will be used for thermal-hydraulic analysis of the Updated Final Safety Analysis Report Chapter 15 non-LOCA transients and accidents for the RFA design. It also states that (1) no transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses, (2) the simplified core model of DPC-NE-3000-PA used for transient analyses was originally developed with additional conservatism over the 8-channel model used for steady-state analyses to specifically minimize the impact of changes in core reload design methods or fuel assembly design, and (3) should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly.

   (a) Explain what additional conservatism is provided in using the simplified core model of DPC-NE-3000-PA.

   (b) What is the criterion/criteria used to determine if transition core transient analyses are warranted? How would it be determined that the criteria have been exceeded without RFA transition core analyses?

9. Regarding rod ejection analysis using SIMULATE-3K, Section 6.6.2.2.1 states that the transient response is made more conservative by increasing the fission cross sections in the ejected rod location and in each assembly and by applying "factors of conservatism" in the moderator temperature coefficient, control rod worths for withdrawal and insertion, Doppler temperature coefficient, effective delay neutron fraction, and ejected rod worth, etc.

   (a) What are the values of the multiplication factors used for fission cross sections, and how are they determined?

(b) How are the input multipliers "VAL" in Equations 6.1 and 6.2 determined? Does "VAL" have a different value for different parameters, such as MTC or DTC? What are the values for these VALs?

(c) In Equation 6.1, the X's are described as "moderator temperatures." Should they be moderator temperature coefficients?

10. Regarding the SIMULATE-3K code, there is an optional "frequency transform" approach, under the "Temporal Integration Models," that can be chosen to separate the fluxes into exponential time varying and predominately spatial components, thus accelerating convergence of the transient neutronic solution and preserving accuracy on a coarser time mesh (see Page 5, Ref. 6-9).

    (a) What determines when the "frequency transform" approach should be used?

    (b) What are the consequences of exercising (or not exercising) this option? Please provide technical justification and comparisons of results.

11. The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current Catawba cores not having the RFA design. For example, DPC-NE-1004A, DPC-NE-2011-PA, DPC-NF-2010A, and DPC-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002-A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation models described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC safety evaluation reports in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions, and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.

12. Section 8.0 states that TS Figure 2.1.1-1 for the reactor core safety limits will be modified by deleting the 2455 psia safety limit line and making the 2400 psia safety limit line as the upper bound pressure allowed for power operation. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2400 psia safety limit line is within the range of the CHF correlations for the Mark-BW and RFA fuel designs.

    However, the safety limit lines in Figure 2.1.1-1 were based on the CHF correlation for the Mark-BW fuel design, in addition to the hot leg boiling limit. Has an analysis been performed to ensure these safety limit lines bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design?

13. TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor F_q (x,y,z) and the enthalpy rise hot channel factor $F_{\Delta h}$ (x,y) to be measured periodically using the incore detector system to ensure that the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limits are not violated. To avoid the possibility that these hot channel factors may increase beyond their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the F_q (x,y,z) or $F_{\Delta h}$ (x,y) has decreased since the previous surveillance. For the reactor core containing the RFA fuel design with integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. Section 8.1 proposes to remove the 2 percent penalty value from these surveillance requirements and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide "typical values" for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors.

(a) Provide the actual values of the margin-decrease penalty factors, as well as the bases, for these values.

(b) Provide references for the approved methodologies used to calculate these values, and to be included in TS 5.6.5 as a part of acceptability for COLR.

Duke Energy.

Duke Energy Corporation
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M. S. Tuckman
Executive Vice President
Nuclear Generation

January 28, 1999

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

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

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

Response to NRC Requests for Additional Information on License Amendment Requests for McGuire and Catawba Nuclear Stations

This submittal contains information that Duke Energy Corporation considers PROPRIETARY and is being made pursuant to 10CFR 2.790.

By letters dated December 9, 1998 and January 5, 1999 the NRC requested additional information on Duke Energy Corporation's July 22, 1998 license amendment requests (LARs) for the McGuire Nuclear Station, Units 1 & 2; and the Catawba Nuclear Station, Units 1 & 2 Technical Specifications. These LARs would permit use of Westinghouse fuel at McGuire and Catawba. Topical Report DPC-NE-2000P/DPC-NE-2009 was also included in the July 22, 1998 Duke submittal.

The thirteen questions contained in the December 9, 1998 NRC letter, and the corresponding Duke answers, are provided in the attachments to this letter. A proprietary version and a non-proprietary version of the Duke response are attached to this letter.

Some of the information contained in Attachment 1 is considered proprietary. In accordance with 10CFR 2.790, Duke Energy Corporation requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of the affected information is included with this letter. A non-proprietary version of the Duke response is included as Attachment 2 to this letter.

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc (w/o Attachment 1):

Mr. L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. F. Rinaldi, Senior Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-14H25
Washington, D. C. 20555-0001

Mr. P. S. Tam, Senior Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-14 H25
Washington, D. C. 20555-0001

Mr. S. M. Shaeffer
NRC Senior Resident Inspector
McGuire Nuclear Station

Mr. D. J. Roberts
NRC Senior Resident Inspector
Catawba Nuclear Station

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant ~~licensing;~~ and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

M. S. Tuckman

M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the Duke response to NRC requests for additional information dated December 9, 1998 and January 5, 1999. The subject of these requests for additional information is a Duke license amendment request dated July 22, 1998 and accompanying topical report designated DPC-NE-2009P, *Duke Power Company Westinghouse Fuel Transition Report*. The information of concern is omitted from the non-proprietary version of the Duke response. This information enables Duke to:

(a) Respond to Generic Letter 83-11, *Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions*.

(b) Perform core design, fuel rod design, and thermal-hydraulic analyses for the Westinghouse Robust Fuel Assembly design.

(c) Simulate UFSAR Chapter 15 transients and accidents for McGuire and Catawba Nuclear Stations.

(d) Perform safety evaluations per 10CFR50.59.

(e) Support Facility Operating Licenses/Technical Specifications amendments for McGuire and Catawba Nuclear Stations.

M. S. Tuckman

M. S. Tuckman

(Continued)

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(Continued)

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

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 28TH day of

JANUARY, 1999

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2001

SEAL

bxc (w/o Attachment 1):

L. A. Keller
M. T. Cash
G. D. Gilbert
K. L. Crane
K. E. Nicholson
R. H. Clark
G. B. Swindlehurst
D. E. Bortz
Catawba Owners: NCMPA-1, NCEMC, PMPA, SREC
Catawba Document Control File (T. K. Pasour)
Catawba RGC File 801.01 (T. K. Pasour)
ELL

Attachment 2

Response to NRC Requests for Additional Information Dated December 9, 1998 and January 5, 1999 Applicable to Duke Energy Corporation License Amendment Requests Dated July 22, 1998

*** Non-Proprietary Version ***

1. **Section 3.2 of DPC-NE-2009P states that conceptual transition core designs using the Robust Fuel Assembly (RFA) design have been evaluated and show that current reload limits remain bounding with respect to key physics parameters, and that in the event that one of the key parameters is exceeded, the evaluation process described in DPC-NE-3001-PA would be performed.**

   (a) Describe the evaluation and the result of the conceptual transition core design.

   (b) Based on the statement, it appears that the evaluation process described in DPC-NE-3001-PA will not be performed unless one of the key parameters is exceeded. Without actual analysis of the RFA transitional or full cores, how is it determined that any of the key parameters is exceeded?

Response 1a:

Conceptual Westinghouse RFA transition core designs were setup and evaluated using NRC approved codes and methods. The evaluation performed considered the effects of partial and full RFA cores. The purpose of the evaluation was to determine the acceptability of the current licensing bases transient analyses. Key safety parameters were calculated for the conceptual core designs and compared against reference values assumed in the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses. Examples of some of the key parameters calculated include Doppler temperature coefficients, moderator temperature coefficients, control bank worth, individual rod worths, boron concentrations, differential boron worths and kinetics data. A summary of the key parameters important to the licensing bases transient analyses are provided in Table 2-1 of DPC-NE-3001. The evaluation demonstrated the expected neutronic similarities between reactor cores loaded with Westinghouse RFA fuel and with Mk-BW fuel, and the acceptability of key safety parameters assumed in the UFSAR Chapter 15 accident analyses.

Response 1b:

Key physics parameters important to the UFSAR Chapter 15 accident analyses are calculated for each reload core using NRC approved methodology to confirm that these parameters are bounded by values assumed in the reference UFSAR Chapter 15 accident evaluations. This check is always performed for each new core design. If the key safety parameters assumed in the reference safety analysis are determined to bound the reload core values, then no additional analyses are required. However, if a key physics parameter is not bounded by the reference value, then the affected accidents will be re-analyzed using the new key physics parameter, or the core will be re-designed to produce an acceptable result.

2. **To demonstrate that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties in DPC-NE-1004-PA are applicable to the RFA design, Section 3.2 cites the analyses performed using Sequoyah Unit 2 Cycles 5, 6 and 7, as well as a 10 CFR 50.59 unreviewed safety question (USQ) evaluation. It is stated that the Sequoyah cores were chosen because they are similar to McGuire and Catawba and contained both Integral Fuel Burnable Absorber (IFBA) and Wet Annular Burnable Absorber fuel. Table 3-1 provides the statistical analysis results of nuclear uncertainty factors, which show they are bounded by the uncertainty factors of DPC-NE-1004A.**

   (a) **Describe any difference between the McGuire/Catawba RFA cores and the Sequoyah cores analyzed. Describe why these differences would not affect the applicability of the analyses of the Sequoyah cores to McGuire and Catawba.**

   (b) **Provide the comparison of the analysis results with measured data of boron concentrations, rod worths, and isothermal temperature coefficients.**

   (c) **Describe the details and results of the 10 CFR 50.59 USQ evaluation.**

Response 2a:

The primary reason for benchmarking the Sequoyah Unit 2 cores was to confirm the fidelity of the CASMO-3/SIMULATE-3 code suite for analyzing reactor cores containing integral fuel burnable absorbers (IFBA). While the introduction of the IFBA product is not considered a major design change, and therefore the benchmarking of this product is not required by the SER requirements of DPC-NE-1004-PA, a conservative approach was adopted to perform benchmark calculations to confirm the acceptability of the current nuclear uncertainty factors. Benchmark calculations were performed using measured data from Sequoyah Unit 2 Cycles 5, 6 and 7.

The Westinghouse Nuclear Design Reports for Sequoyah Unit 2 Cycles 5, 6, and 7, the McGuire and Catawba Updated Final Safety Analysis Report (UFSAR) and Section 2.0 of DPC-NE-2009 were reviewed to determine the differences between the Sequoyah cores analyzed in the benchmark calculations, and McGuire/Catawba RFA core designs A list of differences is provided below.

a. The Sequoyah cores modeled and analyzed in the Sequoyah benchmark calculations contained the Westinghouse Vantage-5H (V5H) fuel design. The V5H fuel design is geometrically (ie. pellet diameter, gap and clad thickness and assembly envelope) equivalent to the RFA fuel design to be used in the McGuire and Catawba cores. Differences between the V5H and RFA fuel design are primarily mechanical and, as a result, do not impact the nuclear characteristics of the fuel assemblies. Specific differences between the V5H and RFA fuel design are summarized below.

   - Zirlo™ is used for the fuel rod clad, guide tubes, instrument tubes and mixing grids in the RFA fuel design. The V5H design uses Zr-4 for these components.
   - The RFA fuel design has thicker instrument and guide tubes than the V5H design in order to improve structural stability.
   - The grid design for the RFA design has been modified (optimized vane angles and window size) to improve thermal performance.

- The RFA design Duke intends to use has a pre-oxide coating on the bottom of the fuel rods, longer fuel rod end plugs and a protective bottom grid. The V5H fuel design used at Sequoyah did not have these features.
- The RFA design employs intermediate mid span mixing grids. The V5H design used at Sequoyah did not use mid span mixing grids.

Neutronically, Zirlo™ and Zr-4 are equivalent. The changes in instrument tube and guide tube thickness does not impact core modeling as long as they are accounted for in the generation of cross sections and few group constants. The pre-oxide coating does not impact the modeling of the fuel rod or the neutronic properties of Zirlo™. The fuel rod end plugs are neutronically unimportant because they are located outside of the active fuel region. The mixing grids are specifically accounted for in the neutronics models, therefore, the use of a modified grid design, the addition of the protective bottom grid and mid span mixing grids should not impact model performance. In summary, the differences in the RFA and V5H fuel designs are primarily mechanical and do not impact the nuclear performance of the fuel assembly. Design features that do impact the neutronics (ie. mid span mixing grids) are specifically accounted for in the nuclear models. Therefore, the results and conclusions reached based on the analysis of the Sequoyah core designs are applicable to the RFA fuel design.

b. The Sequoyah cores that were benchmarked contained both 1.0x and 1.5x IFBAs with rod patterns containing between 48 and 128 IFBA rods. The IFBA loadings (1.0x and 1.5x) and the number of IFBA rods per assembly are representative of the IFBA loadings and the number of IFBA rods expected to be used in McGuire and Catawba core designs. However, the IFBA rod patterns used in the Sequoyah core designs and the IFBA rod patterns that will be used in the McGuire and Catawba core designs are different. The changes in IFBA rod patterns are the result of Westinghouse optimizations that were performed [ ] The optimized IFBA rod patterns will be used in the McGuire and Catawba RFA core designs. In addition, all combinations of IFBA loading and rod patterns are explicitly modeled to account for the impact of any design change in the analysis of each reload core design.

The Sequoyah benchmark calculations that were performed demonstrate the acceptability of the CASMO-3/SIMULATE-3 model to accurately calculate core reactivity, reactivity parameters and power distributions for representative IFBA rod loadings and rod configurations. Changes in the IFBA rod configurations primarily affect intra-assembly peaking and not integral and local nodal power distributions which are the parameters that are measured. Consequently, the results from the benchmark analysis are not expected to change as the result of changing the IFBA rod pattern design.

c. The fuel management strategy (low leakage – ring of fire core designs), the number of fuel assemblies in the reactor core and the core power rating are the same between McGuire, Catawba and Sequoyah. However, there are differences in the reactor coolant flow rate and core inlet temperature. The reactor coolant flow rate at Sequoyah is approximately 3.0% less than at McGuire or Catawba. The core inlet temperature at Sequoyah is ~547°F versus ~555°F at McGuire and Catawba. Core inlet flow and temperature are input variables to the nuclear model and are therefore specifically accounted for. As a result, the performance of the nuclear model and the applicability of the benchmark results are not expected to change due to the aforementioned core inlet flow and temperature differences.

d. The Control Bank (Bank D) for Sequoyah Unit 2 is comprised of 9 control rods versus 5 control rods for McGuire and Catawba. Since control bank locations are specifically modeled, and because during normal operation control banks are positioned near all rods out (ARO), the impact of this difference on the results of the benchmark analysis is negligible.

Response 2b:

Comparisons between Duke predicted and measured zero power physics testing (ZPPT) results are shown below for Sequoyah Unit 2 Cycles 5, 6 and 7. The ZPPT results included comparisons of critical boron concentrations, control rod worths and isothermal temperature coefficients. Excellent agreement between predicted and measured results is generally observed. The large percent differences between predicted and measured control rod worths for Control Bank A in cycles 5 and 6 is primarily the result of the low worth of these banks and to a lessor extent a slight mis-prediction (~1.0%) in the local power distribution. The observed difference in the worth for Control Bank B in cycle 6 is also the result of a slight mis-prediction in the local power distribution and possibly measurement error. However, the observed differences are well within the test acceptance criteria for individual bank worths of +/-30% or 200 pcm, whichever is greater.

[ ]

Response 2c:

A 10CFR 50.59 evaluation was performed to determine if any Unreviewed Safety Questions (USQs) exists when the current methodology is applied to a fuel design that differs from those previously benchmarked and documented in topical report DPC-NE-1004A. For the Duke Power Westinghouse designed nuclear plants, DPC-NE-1004A is considered applicable to Westinghouse OFA, Standard, and FCF Mark-BW (similar to Westinghouse Standard) fuel. The November 1992 SER to this topical stipulated that "*the application of CASMO-3 and SIMULATE-3P to fuel designs that differ significantly from those included in the topical data base should be supported by additional code validation to ensure that the DPC-NE-1004 methodology and uncertainties apply.*" The fuel type evaluated in this 10CFR 50.59 evaluation was the Westinghouse

Performance Plus fuel type (similar to Westinghouse Standard and RFA fuel) with integral fuel burnable absorber (IFBA). The integral fuel burnable absorber consists of a thin coating of ZrB_2 applied directly to fuel pellets of selected fuel rods. The analysis is applicable to the Westinghouse RFA fuel design as discussed in the answer to question 2b.

The results of the evaluation concluded that the methodology described in DPC-NE-1004A is applicable to fuel containing IFBA coated fuel pins. This conclusion is based on the results of benchmark calculations that showed code performance commensurate with that described in DPC-NE-1004A. Power distribution uncertainty factors calculated for fuel containing IFBA coated fuel rods, based on a 95% probability and confidence level, were bounded by uncertainty factors approved by the NRC in DPC-NE-1004A. Consequently, the introduction of IFBA fuel will not change the power peaking uncertainties assumed in the analysis of Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents. Therefore, it can be concluded from a nuclear design perspective that the consequences of UFSAR accidents previously evaluated are not increased and the margin to safety as defined in the bases to Technical Specifications is not decreased. In addition, safety margin will be maintained in future analyses through the application of a conservative combination of uncertainty factors. There are no USQs associated with this change.

3. **Section 3.2 states that (1) in all nuclear design analysis, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores, and (2) when establishing Operating and reactor protection system limits (i.e., loss of coolant accident (LOCA) kw/ft, departure from nucleate boiling (DNB), centerline fuel melt (CFM), transient strain), the fuel specific limits or a conservative overlay of the limits are used. Please elaborate on the mixed core model for nuclear design analyses, and how fuel-specific limits are used.**

Response:

The mixed core model used in the evaluation of transition cores containing RFA and Mark-BW fuel is based on the same methodology that is used to setup a nuclear model for a reactor cores containing a single fuel type. A SIMULATE-3 model is developed for each reload core design in accordance with the methodology described in DPC-NE-1004A. For mixed cores, this model contains cross sections and few group constants for each unique combination of fuel type (ie. RFA or Mark-BW), enrichment and burnable poison loading and geometry. Cross sections and few group constants are derived from [ ] CASMO-3 calculations. The SIMULATE-3 model is used to confirm the acceptability of key physics parameters assumed in UFSAR Chapter 15 accident analyses and to develop core power distributions used in the evaluation of LOCA, DNB, transient strain and centerline fuel melt limits.

The generation of core power distributions for the development of core operational axial flux difference (AFD) limits and the f(ΔI) portion of the over-power delta-T and over-temperature delta-T trip functions (i.e. RPS limits) are conservatively performed using SIMULATE-3 based on the methodology described in DPC-NE-2011PA. The power distributions developed during this process are compared against fuel specific Mark-BW and RFA LOCA, DNB, CFM and transient strain limits by assigning specific Mark-BW and RFA limits to each fuel type. Mark-BW and RFA fuel limits are developed using NRC approved methodologies. If positive margin exists to all limits, then no changes are made to operational AFD, or the RPS limits used in the development of the f(ΔI) trip functions. If any of the limits are exceeded, then either (1) the AFD or RPS limits are reduced to produce positive margin to all limits, (2) a specific analysis is performed on the out-of-limit parameter, or (3) the core is redesigned.

In some instances it may be desirable to develop a single composite set of limits that can be used to evaluate both fuel types. For this scenario, a conservative overlay of Mark-BW and RFA limits would be performed to develop a single set of limits that would be applicable to both Mark-BW and RFA fuel. Either of the above mentioned approaches is equally valid.

4. **Section 5.2 states that in using the VIPRE-01 code for the reactor core thermal-hydraulic analysis, the reference power distribution based on a 1.60 peak pin from DPC-NE-2004P-A, Rev.1, was used.**

   (a) **The report states that this reference pin power distribution "was" used. Will it be used for future RFA reload analyses?**

   (b) **Does the reference pin power distribution used in the core thermal-hydraulic analyses bound all power distribution for the RFA cores for future reload cycles?**

Response 4a:

The reference power distribution given in DPC-NE-2004P-A, Rev. 1 will be used in all future RFA analyses. This radial pin power distribution (the relationship of the peak pin to the remaining fuel pins in the highest power fuel assembly) used in DPC-NE-2009P and previous topical reports will not be modified. This maintains the relative radial power distribution the same as previously approved. There are no plans to change this distribution.

The peak pin value, however, could be increased in the future to utilize the increased thermal performance available in the RFA design. For DNB analyses using the Maximum Allowable Peaking (MAP) methodology described in DPC-NE-2004P-A, Rev. 1, the key DNB parameter is the reference power distribution, not the peak pin power. The peak pin power is only meaningful when all other DNB parameters are specified (axial peak location and magnitude, core power level, RCS pressure, flow rate, and temperature). The reference power distribution is used to create the Maximum Allowable Peaking (MAP) limits that ensure the required level of DNBR protection is provided. The MAP limits define the maximum allowable peak pin as a function of axial peak. The reference power distribution is used consistently in all DNB analyses (core DNB limit lines, transient analyses, SCD statepoint determinations, etc.). Any change in the peak pin value will be evaluated in all DNB analyses and will be reflected in the Maximum Allowable Peaking limits provided in the COLR for each reload cycle.

The ability to increase the peak pin value is a result of a new fuel design, additional design features, a new or modified CHF correlation, or changes to the analysis conditions. If the performance improvement is related to fuel hardware or correlation change, a submittal is made to the NRC and approval required prior to use. If the change is to the analysis conditions and no methodology is modified, the change can be implemented through the 10CFR50.59 process. In either case, any increase in the peak pin value is not made unless all analyses and related licensing limits are verified to be conservatively satisfied.

Response 4b:

The reference power distribution used to create the Maximum Allowable Peaking (MAP) limits is used in all steady state generic analyses. This distribution is verified each reload by performing DNB calculations with cycle specific predicted radial pin power distributions. This specific pin distribution comparison between what is predicted for a particular cycle and the generic analysis reference power distribution verifies the conservatism of the reference distribution.

5. **Section 5.2 states that in the thermal-hydraulic analysis of the RFA design using VIPRE-01, the two-phase flow correlations will be changed from the Levy subcooled void correlation and the Zuber-Findlay bulk void correlation to the EPRI subcooled and bulk void correlations, respectively. While the sensitivity study provided in the report shows a minimal difference of 0.1% between the minimum DNBRs of 51 RFA CHF test data points calculated with both set of correlations, it was stated in DPC-NE-2004 that the Levy/Zuber-Findlay combination compared most favorably with the Mark-BW test results as the DNBRs of the tests calculated with this combination yielded conservative results relative to the EPRI correlations.**

   (a) Discuss whether the EPRI correlations will be used for the RFA design only, or they will also be used for the Mark-BW design.

   (b) If the EPRI correlations will also be used for Mark-BW design, provide justification for their use.

   (c) If the Levy/Zuber-Findlay correlations will continue to be used for Mark-BW fuel design, discuss how the VIPRE-01 code will be used to analyze transient mixed cores having both Mark-BW and RFA fuel designs.

Response 5a:

The EPRI correlations will only be used in the RFA models in VIPRE-01. The Levy/ Zuber-Findlay combination will be used when modeling Mark-BW fuel.

Duke considers the selection of the two-phase flow correlations to be a very minor effect on DNBR analyses. Mark-BW CHF test data was analyzed with both Levy/Zuber-Findlay and EPRI/EPRI in the same manner as the RFA with comparable results.

Response 5b:

See 5(a) above.

Response 5c:

The transition core models use the simplified (8 Channel) models to maximize the impact of different fuel types. In the transition core model, the limiting assembly is modeled as an RFA and the rest of the core is modeled as Mark-BW fuel. Since the MDNBR occurs in the limiting assembly, the void correlations are input for the fuel type modeled as the limiting assembly. For the RFA/Mark-BW transition core model, the RFA design is the limiting assembly; thus the EPRI set of correlations are used.

The transition analyses covered a wide range of statepoint fluid conditions and 3-dimensional core power distributions. This matrix of conditions were analyzed using both the EPRI and Levy/Zuber-Findlay correlations with minimal difference in transition core results using either set of void correlations (average difference of <1% in peaking).

6. **Section 5.7 describes the use of a transition 8-channel RFA/Mark-BW core model to determine the impact of the geometric and hydraulic differences between the resident Mark-BW fuel and the RFA design, and determine a conservative DNBR penalty to be applied for the transition cores. Table 5-4 presented the statistical DNBRs for the 500 and 5000 case runs for various statepoints including the transition core case of the most limiting statepoint 12. The statistical design limit is chosen to bound both the full RFA cores and RFA/Mark-BW transition cores for the 5000 case runs.**

   (a) Why is the statistical design limit value proprietary information?

   (b) With respect to the statistical core design methodology, describe how the uncertainties of the CHF correlation and the VIPRE code/model are propagated with the uncertainties of the selected parameters of each statepoint for the calculation of the statistical DNBR for each statepoint in Table 5-4.

   (c) With the statistical design limit specified in Section 5.7, is it your intention to use a full core of RFA in the thermal hydraulic analysis for the transition core without the transition core DNBR penalty factor?

Response 6a:

The Statistical Design Limit (SDL) will be changed to non-proprietary. This change will be included when the approved versions of the report are issued.

Response 6b:

When a statepoint is selected, all key parameters, including CHF correlation and code/model uncertainties, are randomly varied based on the uncertainty distribution and magnitude. The resulting values of power, pressure, temperature, flow, and 3-D power distribution are used to create the VIPRE-01 input for the cases. After the code is executed and the DNBR calculated for each case, the DNBR value is multiplied by the propagated values for the CHF correlation uncertainty and the VIPRE code/model uncertainty. This final DNBR value for each case (500 or 5000 cases are run for each statepoint) is used to determine the statepoint's statistical DNBR value.

Response 6c:

The analysis discussed in the last paragraph of Section 5.7 verified that the statistical DNB limit developed with a full core RFA model is valid for transition RFA/Mark-BW cores. The limiting statepoint (12TR) was evaluated using the RFA/Mark-BW transition core model, confirming that the same statistical design limit can be used for transition and full core analyses.

The transition core DNB penalty factor is determined separately using the RFA/Mark-BW transition core model described in Section 5.7. The DNB penalty is determined by evaluating the effect of the transition core hydraulic behavior on the Maximum Allowable Peaking (MAP) limits calculated for a full RFA core. The resulting DNB penalty is then accounted for in all RFA/Mark-BW transition core DNB analyses.

7. **Section 2.0 states that the RFA is designed to be mechanically and hydraulically compatible with the Mark-BW fuel. Table 2.1 provides a comparison of the basic design parameters of the two fuel designs, but does not provide a comparison of the hydraulic characteristics of spacer grids. Section 5.2 states that the VIPRE-01 core thermal-hydraulic analyses were performed with applicable form loss coefficients according to the vendor. Table 5.1 provides general RFA fuel specifications and characteristics without the hydraulic characteristics of the spacer grids.**

   (a) Provide comparisons for the thickness, height, and form loss coefficients of the RFA and Mark-BW fuel spacer grids, including mixing-vane and non-mixing vane structural grids, and intermediate flow mixing grids.

   (b) Provide the form loss coefficients of the spacer grids used in the analyses and in the RFA CHF test assemblies if they are different from the values described in item (a).

   (c) Describe the procedures to ensure that the form loss coefficients of the RFA grids are comparable to those used in the statistical core design analysis and the CHF tests so that both the WRB-2M CHF correlation DNBR limit and the statistical core design limit are valid.

Response 7a:

The grid data is shown in the following table:

[ ]

Response 7b:

The RFA CHF tests used Mixing Vane (MV) and intermediate flow mixing (IFM) grids representative of the production RFA design fuel assembly. The CHF test sections are a 5x5 rod bundle with either all typical (unit) cells or typical cells with a thimble (guide tube) cell in the center. The form loss coefficients for the CHF test section are calculated for these subchannels and are based on the total 5x5 bundle flow area. Likewise, the fuel assembly subchannel form loss coefficients are calculated based on the fuel assembly flow area. The ratio of thimble/typical cell form loss coefficients, to which DNBR is sensitive, is equivalent for the CHF test section and the production grid (for both MV and IFM grids). Therefore, the CHF test section and production RFA grids are identical with respect to DNBR analyses.

In comparing the test versus production geometry, the vanes and strap features of the respective grid types are consistent. There is one slight difference between one of the CHF test sections and the production fuel assemblies. The thimble OD was 0.474 inches for the thimble CHF rod bundle section tested. The production assembly will have thimbles with an OD of 0.482 inches. The difference in thimble tube OD has negligible impact on the correlation's predictive capability. This difference was addressed in WCAP-15025 and determined to be acceptable.

Response 7c:

The RFA analysis was completed with the form loss coefficients supplied in response to Question 7a. The transition core analysis used the RFA and Mark-BW values listed in the table in the respective model locations to accurately capture the hydraulic differences between the fuel types side-by-side incore.

For each batch of fuel manufactured, critical RFA grid dimensions and form loss coefficients are supplied by the vendor to Duke Power. This data, along with other critical reload analysis parameters, are transmitted to Duke, on a batch basis, in a QA document known as the Databook. Upon receipt of the Databook, the fuel design is frozen and may not be changed without Duke Power concurrence. This design notification process, including the process for changes occurring after the batch is frozen, is described in Duke Power Nuclear Engineering Workplace Procedure XSTP-101. The batch specific design information, transmitted in the Databook, will be used to ensure the validity of the Duke VIPRE-01 RFA models and associated SCD limit.

Any changes in the design data will be evaluated to verify that the generic analyses remain valid or the analyses will be revised using the new design data.

8. Section 6.1.3 states that the thermal-hydraulic methodology described in DPC-NE-3000-PA Revision 1, with a simplified core model will be used for thermal-hydraulic analysis for the Updated Final Safety Analysis Report Chapter 15 non-LOCA transients and accidents for the RFA design. It also states that (1) no transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses, (2) the simplified core model of DPC-NE-3000-PA used for transient analyses was originally developed with additional conservatism over the 8-channel model used for steady-state analyses to specifically minimize the impact of changes in core reload design methods or fuel assembly design, and (3) should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly.

(a) Explain what additional conservatism is provided in using the simplified core model of DPC-NE-3000-PA.

(b) What is the criterion/criteria used to determine if transition core transient analyses are warranted? How would it be determined that the criteria have been exceeded without RFA transition core analyses?

Response 8a:

The additional conservatism provided in using the simplified core model of DPC-NE-3000-PA is described in detail in Section 3.3.4 of DPC-NE-3000-PA (Reference 6-1 of DPC-NE-2009-P)

Response 8b:

Section 6.1.3 states the following. "No transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses. . . . . . Should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly." These statements summarize the results of an evaluation that has concluded that based on current information there is no need for performing transition core analyses for transients. The transition core effects on core thermal-hydraulic analyses for transients are adequately assessed by the steady-state core thermal-hydraulic transition core analysis in Chapter 5. The purpose of the second sentence quoted above was to state Duke's intent to evaluate any emerging issues or information, and, if necessary, to re-evaluate the current conclusion that no transient analyses of transition core effects are necessary. Duke does not expect any emerging information to change this conclusion, but Duke will address any such situations in the future.

9. Regarding rod ejection analysis using SIMULATE-3K, Section 6.6.2.2.1 states that the transient response is made more conservative by increasing the fission cross sections in the ejected rod location and in each assembly and by applying "factors of conservatism" in the moderator temperature coefficient, control rod worths for withdrawal and insertion, Doppler temperature coefficient, effective delayed neutron fraction, and ejected rod worth, etc.

(a) What are the values of the multiplication factors used for fission cross sections, and how are they determined?

(b) How are the input multipliers "VAL" in Equations 6.1 and 6.2 determined? Does "VAL" have a different value for different parameters, such as MTC or DTC? What are the values for these VALs?

(c) In Equation 6.1, the X's are described as "moderator temperatures." Should they be moderator temperature coefficients?

Response 9a:

An iterative process is used to determine the [

] Note

that these multipliers are [ ]

The methodology used to determine the [ ] adjustments is consistent with the power distribution adjustment methodology described in DPC-NE-3001 with one exception. [

]

The BOC and EOC [ ] multipliers are shown in Figures 9-1 and 9-2.

Figure 9-1

BOC [ ] Multipliers

	H	G	F	E	D	C	B	A
8	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
9	1.0	1.0	1 0	1 0	1.0	1.0	1.0	1.0
10	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
11	1.0	1.0	1.0	1.1450	0.9810	1.0445	1.0	1.0
12	1.0	1.0	1.0	0.9810	1.0904	0.9435	1.0	
13	1.0	1.0	1.0	1.0445	0.9435	1.1358	1.0	
14	1.0	1.0	1.0	1.0	1.0	1.0		
15	1.0	1.0	1.0	1.0				

Figure 9-2

EOC [ ] Multipliers

	H	G	F	E	D	C	B	A
8	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
9	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
10	1.0	1.0	1.0	1.0	1.0	1 0	1.0	1.0
11	1.0	1.0	1.0	1.1955	0.9760	1.0960	1.0	1.0
12	1.0	1.0	1.0	0.9760	1.1230	0.9530	1.0	
13	1.0	1.0	1.0	1.0960	0.9530	1.1240	1.0	
14	1.0	1.0	1.0	1.0	1.0	1.0		
15	1.0	1.0	1.0	1.0				

Response 9b:

The input multiplier "VAL" in equation 6.1 is determined through an iterative process until bounding control rod worths (ejected and trip rod worths), Doppler temperature coefficients and moderator temperature coefficients are determined. Unique multipliers are required for each of the parameters adjusted. As a result, unique sets of multipliers are calculated for each of the four rod ejection accident cases evaluated (ie. BOC HFP and HZP and EOC HFP and HZP).

Conservative Doppler temperature coefficients are calculated [ ] For this case, the "X" variable in equation 6.1 is fuel temperature. The parameter "VAL" is adjusted until a conservative Doppler temperature coefficient is determined. Conservative moderator temperature coefficients are developed by [ ] Iterations are performed until a multiplier is determined that yields the desired moderator temperature coefficient. For this calculation, the X variable in equation 6 1 is moderator temperature.

A similar process is used to develop limiting control rod worths. Ejected rod worths are conservatively calculated by [ ] Trip rod worths are minimized to conservatively limit the amount of negative reactivity inserted into the core post trip assuming the highest worth control rod and ejected control rod are fully withdrawn. For these cases, the X variable in equation 6.1 is the [ ] Iterations are performed until a conservative ejected rod worth and trip rod worth are calculated.

The multiplier required to produce a conservative beta-effective is determined by re-arranging equation 6.2 to the following.

[ ]

The multipliers calculated for each of the key physics parameters assumed in each of the four rod ejection accidents (ie. BOC HFP and HZP and EOC HFP and HZP) are shown below. These multipliers were developed to produce bounding key physics parameters to ensure a conservative transient response. The multipliers presented are unique to each of the accidents presented in this report [ ]

	BOC HFP	BOC HFP	BOC HZP	BOC HZP
Parameter	Multiplier (VAL)	Target Value	Multiplier (VAL)	Target Value
DTC	0.689	-0.90 pcm/°F	0.555	-0.90 pcm/°F
MTC	-0.005	0.0 pcm/°F	-1.247	0.0 pcm/°F
Ejected Rod Worth	1.168	200 pcm	1.029	720 pcm
Trip Worth	0.510	250 pcm	1.650	250 pcm
Beta-effective	0.882	0.0050	0.878	0.0050

	EOC HFP	EOC HFP	EOC HZP	EOC HZP
Parameter	Multiplier (VAL)	Target Value	Multiplier (VAL)	Target Value
DTC	0.810	-1.20 pcm/°F	0.666	-1.20 pcm/°F
MTC	0.283	-10.0 pcm/°F	0.478	-10.0 pcm/°F
Ejected Rod Worth	1.055	200 pcm	0.868	900 pcm
Trip Worth	1.073	250 pcm	1.650	250 pcm
Beta-effective	0.768	0.0040	0.763	0.0040

Response 9c:

No. The X's in equation 6.1 are moderator temperature. Refer to answer "9b" for additional information.

10. Regarding the SIMULATE-3K code, there is an optional "frequency transform" approach, under the "Temporal Integration Models," that can be chosen to separate the fluxes into exponential time varying and predominately spatial components, thus accelerating convergence of the transient neutronic solution and preserving accuracy on a coarser time mesh (see Page 5, Ref. 6-9).

(a) What determines when the "frequency transform" approach should be used?

(b) What are the consequences of exercising (or not exercising) this option? Please provide technical justification and comparisons of results.

Response 10a:

The frequency transform method is SIMULATE-3K's default transient neutronics solution option and was used in all of the transient evaluations presented in DPC-NE-2009. This approach was used because it is computationally more efficient and reproduces the results of finite difference methods, which require smaller times step to achieve the same accuracy as the frequency transform method.

The method used to solve the transient neutronics equations is determined by the code user and used throughout the transient. There is no switching of solution methods during the transient.

Response 10b:

There are no physical consequences from using either the frequency transform or finite difference methods to solve the transient neutronic equations since both methods are equally accurate. From theory, the flux variation from one time step to the next is exponential. The frequency transform method takes credit for this behavior, instead of an assumed linear variation in simple finite difference methods. By taking credit for the exponential flux variation, computational efficiency is increased because larger time steps can be taken without loss in accuracy as is the case in traditional methods. Therefore, the frequency transform method is the preferred solution technique because it produces the same answers as finite difference methods, but with reduced code execution time.

Sensitivity studies performed showed no difference in the peak core power or the time of the peak core power for cases where the frequency transform method was turned on and off. Therefore, it can be concluded that there is no consequence of using the frequency transform approach.

11. **The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current Catawba cores not having the RFA design. For example, DPC-NE-1004A, DPC-NE-2011-PA, DPC-NF-2010A and DPE-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002-A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation model described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC safety evaluation reports in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.**

Response:

DPC-NE-1004A, Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, Rev. 1, April 1998.

The SER states that this methodology is acceptable for performing reload analyses for B&W 177 and Westinghouse 193 assembly reactor cores. subject to the following restrictions:

a. The application of CASMO-3 and SIMULATE-3 to fuel designs that differ significantly from those included in the topical data base should be supported by additional code validation to ensure that the DPC-NE-1004A methodology and uncertainties apply.

   Resolution: While Duke does not consider the introduction of the Integral Fuel Burnable Absorber (IFBA) in the Westinghouse RFA design to be a significant fuel design change, a conservative approach was adopted to confirm the acceptability of current nuclear uncertainty factors because of the availability of IFBA benchmark data. The uncertainty analysis (described in Section 3.2 and in the answer to question 2) confirmed the acceptability of the currently licensed nuclear uncertainty factors for $F\Delta H$, Fq and FZ for Westinghouse fuel containing IFBA and WABA burnable absorbers.

b. The system of codes represented in the topical report must be protected with appropriate quality assurance procedures, subject to auditing by the NRC staff.

   Resolution: The codes represented in the topical report DPC-NE-1004A are procedurally controlled and are in compliance with the Duke Energy Corporation Quality Assurance Topical Report which is in compliance with the requirements of 10CFR 50, Appendix B and other approved industry standards such as ANSI N45.2-1971 and ANSI N18.7-1976.

DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990.

The SER for this methodology imposes the following restrictions:

a. The application of this methodology is limited to the McGuire and Catawba Nuclear Stations.

   Resolution: Duke is only using this methodology for McGuire and Catawba

b. The application of this methodology to other Westinghouse plants would be acceptable provided that plant-specific differences be considered and justified.

   Resolution: Duke is only using this methodology for McGuire and Catawba. The use of this methodology for application to another Westinghouse unit (or units) would require the submittal of license amendments and NRC approval.

c. Application of this methodology is contingent upon NRC approval of the Reload Design Thermal-Hydraulic Methodology DPC-NE-2004P-A using the VIPRE-01 code. (Topical Approved)

   Resolution: The Thermal-Hydraulic Methodology described in Topical Report DPC-NE-2004P-A has been approved.

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

   Resolution: The approval of the Topical Report DPC-NE-1004A allowed the substitution of either the CASMO-3/SIMULATE-3P or the CASMO-3/NODE-P codes in place of the EPRI-NODE-P and PDQ-07.

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

The SER for this methodology imposes the following restrictions:

None, with the exception that the methodology in sections 6.3, 7.1, 7.2, 7.3, 7.4 and 7.4.1 were excluded from this report. The replacement methodology is described below:

a. Section 6.3: Comparison of Cycle Specific Safety Related Physics Parameters

   Resolution: This methodology was replaced by the methodology described in Topical Report DPC-NE-3001PA.

b. Section 7.1 – 7.4.1: Three-dimensional peaking analysis

   Resolution: This methodology was replaced by the methodology described in Topical Report DPC-NE-2011PA.

DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," Revision 1, February 1997.

The limitations, conditions or restrictions identified in the SER and TER for DPC-NE-2004, Revision 1, are:

a. The DPC developed statistical core design methodology, as described in the submittal, is a generic methodology and is conceptually acceptable and generally applicable to other PWR plants; however, the approval we recommend at this time is only for McGuire and Catawba Nuclear Stations due to DPC's use of the specific uncertainties and distributions based upon plant data and its selection of statepoints used for generating the statistical design limit.

   Resolution: The RFA SCD analysis presented in DPC-NE-2009 is only for McGuire and Catawba.

b. Either the response surface model (RSM) must be re-evaluated or the "simplified method" for determining an SDL using VIPRE-01 directly must be used whenever any of the following occur:

   - a significant change is made in the fuel assembly design
   - a new or revised CHF correlation is developed
   - operating conditions outside the range of conditions considered in the development of the RSM

   The licensee is further required to make a submittal to the NRC for review if a new SDL is calculated as a result of conditions outside the range of conditions considered in the development of the RSM.

   Resolution: The RSM was not used to calculate the SDL in DPC-NE-2009 The RFA analysis presented in DPC-NE-2009 calculates the SDL for the RFA fuel with the WRB-2M CHF correlation as per the "simplified method" referenced in DPC-NE-2004, Rev 1 which is the SCD calculation methodology subsequently approved in DPC-NE-2005, Rev 1.

c. Whenever DPC intends to use other CHF correlations, power distribution, fuel pin conduction model, or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, DPC must submit its justification for NRC review and approval.

   Resolution: DPC-NE-2009 identifies the VIPRE-01 modeling requirements as well as the CHF correlation and statistical analysis limit.

d. Core bypass flow is cycle dependent. DPC will verify, in future applications, that its use of a particular core flowrate resulting from a bypass flowrate for that cycle is bounded by the range of values used in the subject topical report. Otherwise DPC will reassess the need for regeneration of a new response surface model.

Resolution: The value of core bypass flow used in the generic RFA SCD analysis is expected to bound the values for all reload cores. The core bypass flow will be verified on a cycle by cycle basis to ensure conservatism

DPC-NE-2005P-A, Thermal-Hydraulic Statistical Core Design Methodology", Revision 1, November 1996.

The limitations, conditions or restrictions identified in the SER and TER for DPC-NE-2005P-A are:

a. The statistical core design (SCD) methodology developed by DPC, as described in the submittal (DPC-NE-2005), is direct and general enough to be widely applicable to any pressurized-water reactor (PWR) fuel or reactor, provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the critical heat flux (CHF) correlation subject to the conditions in the VIPRE safety evaluation report (SER). DPC committed in their topical report that its use of specific uncertainties and distributions will be justified on a plant specific basis, and also that its selection of statepoints used for generating the statistical design limit will be justified to be appropriate. The methodology is approved only for use in DPC plants.

   Resolution: Addressed in Chapter 5 of DPC-NE-2009. The RFA analysis presented in DPC-NE-2009 is only for McGuire and Catawba.

b. Of the two DNBR limits, only the use of the single, most-conservative DNBR limit is approved.

   Resolution: Use of two DNBR limits was not requested in this submittal. The single DNBR limit stated for use for RFA fuel in full cores or transition cores will be used for all statepoints within the conditions listed in Table 5-5.

WCAP-15025, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids".

The limitations, conditions or restrictions identified in the SER and for WCAP-15025 are:

a. Since WRB-2M was developed from test assemblies designed to simulate Modified 17x17 Vantage 5H fuel the correlation may only be used to perform evaluations for fuel of that type without further justification. Modified Vantage 5H fuel with or without modified intermediate flow mixer grids may be evaluated with WRB-2M.

   Resolution: The SCD analysis presented in DPC-NE-2009 is for the RFA design, which includes the modified low pressure drop structural mid-grids and modified intermediate flow mixing grids (see Chapter 2 of DPC-NE-2009). The WRB-2M CHF correlation is used for all DNBR calculations on the RFA.

b. Since WRB-2M is dependent on calculated local fluid properties these should be calculated by a computer code that has been reviewed and approved by the NRC staff for that purpose. Currently WRB-2M with a DNBR limit of 1.14 may be used with the THINC-IV computer code. The use of VIPRE-01 by Westinghouse with WRB-2M is currently under separate review.

   Resolution: The DNB analyses in DPC-NE-2009 are performed with VIPRE-01. As stated in Section 2.3 and 3.3 of WCAP-15025, both VIPRE-01 and THINC-IV were used to analyze the CHF test data. Tables A-1 to A-4 in the Appendix to WCAP-15025 show the local fluid conditions calculated with VIPRE-01. Also, the SER for WCAP-15025 states that the "results of the THINC-IV analyses agreed with those from VIPRE-01." Additionally, VIPRE-01 was approved for use in thermal/ hydraulic analyses at McGuire and Catawba in DPC-NE-2004, Revision 1. Based on this, Duke has used and will continue to use VIPRE-01 to perform all RFA analyses.

c. WRB-2M may be used for PWR plant analyses of steady state and reactor transients other than loss of coolant accidents. Use of WRB-2M for loss of coolant accident analysis will require additional justification that the applicable NRC regulations are met and the computer code used to calculate local fuel element thermal/hydraulic properties has been approved for that purpose

   Resolution. The RFA LOCA analysis is not described in DPC-NE-2009. The LOCA analysis is performed by the fuel vendor with the approved correlations specified by the vendor's methodology. The CHF correlation used in the LOCA analysis is listed in WCAP-8301.

d. The correlation should not be used outside the range of applicability defined by the range of the test data from which it was developed. This range is listed in Table 1.

   Resolution· Table 1 is listed below for reference.

Parameter	Range
Pressure (psia)	$1495 \leq P \leq 2425$
Local Mass Velocity (Mlbm/hr-ft^2)	$0.97 \leq G \leq 3.1$
Local Quality	$-0.1 \leq X \leq 0.29$
Heated length, inlet to CHF location (ft)	$Lh \leq 14$
Grid Spacing (in)	$10 \leq gsp \leq 20\ 6$
Equivalent hydraulic diameter (in)	$0.37 \leq De \leq 0.46$
Equivalent heated diameter (in)	$0.46 \leq Dh \leq 0.54$

The WRB-2M CHF correlation was used for all RFA DNBR calculations. The fluid parameter ranges (first two items on the parameter list) are confirmed by the statepoint selection listed in Chapter 5 of DPC-NE-2009. The fuel design related parameters (last four items on the parameter list) are confirmed implicitly by the fuel model. The local quality limit is verified for each analysis.

Attachment 2
(Non-Proprietary)

DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology," Revision 2, December 1997.

The original SER dated November 15, 1991 lists the following conditions (Section 3.0). The SER for Revision 1 dated August 8, 1994 does not have any limitations or conditions for McGuire and Catawba. The SER for Revision 2 dated October 14, 1998 does not have any new limitations or conditions.

a. With respect to analyzing transients which result in a reduction in steam generator secondary water inventory, use of the RETRAN-02 steam generator modeling is acceptable, only for transients in that category for which the secondary side inventory for the effective steam generator(s) relied upon for heat removal never decreases below an amount which would cover enough tube height to remove decay heat.

   Resolution: By letter dated September 25, 1998 (G. R. Peterson to NRC Document Control Desk), Duke notified the NRC of a new RETRAN-02 steam generator model which addresses this SER condition for the Catawba Unit 2 UFSAR Section 15.2.7 loss of normal feedwater analysis. This submittal is currently under NRC review. The subject of this condition is not applicable for all other UFSAR Chapter 15 transients and accidents for McGuire and Catawba.

b. All generic limitations specified in the RETRAN-02 SER.

   Resolution: By letter dated June 3, 1991 (M.S. Tuckman to NRC Document Control Desk), Duke responded to the generic limitations specified in the RETRAN-02 SER in the response to Question #29. This response along with subsequent methodology revisions (including the revisions in DPC-NE-2009-P) have all been submitted to the NRC. Later RETRAN-02 SERs were reviewed and it was determined that three new SER conditions exist for the RETRAN-02 MOD005.0 code version (SER dated November 1, 1991). The responses to these conditions are as follows:

   1) The user must justify, for each transient in which the general transport model, the selected degree of mixing with considerations as discussed in Section 2.1 of this SER.

      Response: Topical report DPC-NE-3001-P described the application of the general transport model in the Duke methodology. The topical report was reviewed and approved by the NRC.

   2) The user must justify, for each use of the ANS 1979 standard decay heat model, the associated parameter inputs, as discussed in Section 2.2 of this SER.

      Response: The Duke modeling of decay heat as described in topical report DPC-NE-3002-A is based on the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty. This decay heat modeling approach is standard in the industry for non-LOCA analyses, and meets the intent of this condition. The NRC has reviewed and approved DPC-NE-3002-A.

   3) Because of the inexactness of the new reactivity edit feature, use of values in the edit either directly or as constituent factors in calculations of parameters for comparison to formal performance criteria must be justified.

Response: The Duke methodology does not use the reactivity edit feature in the manner that is the subject of this condition. Therefore this condition is not applicable.

c. Determination of acceptability is based upon review of selection of models/correlations for transients involving symmetric core neutronic and thermal-hydraulic conditions only. Thus, VIPRE-01 models are approved for use in analyzing symmetric transients only.

Resolution: The DPC-NE-3001-PA topical report submitted VIPRE-01 models for transients involving asymmetric core neutronic and thermal-hydraulic conditions. NRC approval of DPC-NE-3001-PA closed out this condition.

d. When using the DPC developed SCD method, the licensee must satisfy the conditions set forth in the staff's safety evaluation of DPC-NE-2004.

Resolution: Duke responded to this question by letter dated August 29, 1991 (M. S. Tuckman to NRC Document Control Desk). Attachment 2 to this letter addresses the applicable conditions and how these conditions are met, and is summarized as follows. The first condition requiring submittal of models for asymmetric transients was met with the submittal of DPC-NE-3001. The second condition required a transition core penalty to be applied. The details of the transition core penalty modeling were presented in the response. The third condition required modeling to avoid errors related to the use of the subcooled boiling models. A commitment to properly apply this model was made. The fourth condition required submittal of the BWCMV correlation prior to use, which was done. The fifth condition required future submittal of any methodology changes to important inputs and models such as different CHF correlations, power distributions, input options, etc. Duke observes this condition and has and will submit such methodology changes prior to implementation. The sixth condition requires that the core bypass flow be determined and justified on a cycle-by-cycle basis. Duke commits to confirming that the core bypass flow for each reload cycle will be bounded by the core bypass flow assumed in the analyses.

e. Whenever DPC intends to use other CHF correlations, power distribution, fuel pin conduction model or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, DPC must submit its justification for NRC review and approval.

Resolution: Duke recognizes the requirements of this condition and continues to meet this condition. For example, Revision 2 to DPC-NE-3000-P (Letter, M. S. Tuckman to NRC Document Control Desk, December 23, 1997) submitted revised VIPRE-01 methodology to include the Mk-B11 fuel assembly design and the BWU-Z CHF correlation, along with other minor changes. Future revisions to Duke topical reports will be submitted as necessary per the requirements of this condition.

DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," November 1991.

The SER dated November 15, 1991 lists the following limitations (Section 4.0).

a. The licensing application of the SIMULATE-3P static methods for determining the key safety parameters requires NRC approval of the reference topical report, DPC-NE-1004 (Section 3.1)

   Resolution: Topical report DPC-NE-1004-A was reviewed and approved by the NRC. The latest NRC SER for DPC-NE-1004-A, Revision 1, is dated April 26, 1996

b. The licensing application of the DPC-NE-3001-P transient analysis methods requires NRC approval of MOD005 of RETRAN-02 for boron transport calculations (Section 3.5)

   Resolution: The NRC SER for RETRAN-02 MOD005.0 was dated November 1, 1991.

c. The licensing application of the DPC-NE-3001-P transient analysis methods requires NRC approval of the thermal-hydraulics topical report DPC-NE-3000 (Section 3.5)

   Resolution: The NRC SER (McGuire/Catawba scope) for DPC-NE-3000-PA was dated November 15, 1991.

DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology," Revision 2, December 1997.

The original SER dated November 15, 1991 lists the following conditions (Section 3.0).

a. DPC's Statistical Core Design methodology treats seven state variables as key parameters. Four of these variables were accounted for in this topical report. Of the remaining parameters, the power factors are also input items for systems analysis, which was not presented in the topical report. Similarly, reactivity feedback was not discussed in this report. Both of these parameters can significantly influence the course of the transient. Therefore, when application of the philosophical approach reported in this topical report is made and submitted for NRC review and approval, review should be made of the modeling of power and reactivity feedback, and to assure that such modeling has no adverse impact on the other modeling described herein.

   Resolution: The power factors used in the models are described in the DPC-NE-3000 topical report, which has been reviewed and approved by the NRC. The reactivity feedback modeling is described in the DPC-NE-3001 topical report, which has been reviewed and approved by the NRC. The application of the integrated methodologies, including UFSAR Chapter 15 revisions, was submitted on June 26, 1991 for the McGuire 1 Cycle 8 reload license amendment application. The SER for this submittal was dated November 27, 1991. The above DPC-NE-3002-A SER condition appears to be directed at the NRC review of the other topical reports and to the application of the methodology. It is inferred via NRC review and approval of all of the related topical reports and of the McGuire 1 Cycle 8 reload that the intent of this condition has been met.

b. Validity of DPC's assumption of 120% of design pressure as part of the acceptance criteria for Reactor Coolant Pump Locked Rotor should be determined by the NRC staff.

Resolution: Duke has adopted an acceptance criterion of 110% of design pressure for the locked rotor accident analysis as stated in Section 4.3 of DPC-NE-3002, Revision 2.

c. No justification was presented for trip and actuation times assumed in the Feedwater System Pipe Break event analysis. Such justifications must be presented when this methodology is applied.

Resolution: The NRC SER, Section 2.2.1, dated November 15, 1991 specifically states that this TER condition is outside of the scope of DPC-NE-3002 and this review. Therefore, Duke has not prepared a response to this TER condition.

d. DPC documented intent to perform parametric studies in order to select conservative scenarios or assumptions throughout the subject topical report. Therefore, such parametric studies must be presented when this methodology is applied.

Resolution: The DPC-NE-3002-A topical report states that parametric studies are necessary to determine the conservative modeling approach for a limited number of assumptions for some of the transients. These parametric studies were performed and are documented in the engineering calculations. The results of the analyses using the conservative modeling approach and assumptions were submitted for NRC review with the McGuire 1 Cycle 8 reload license amendment request dated June 26, 1991. These results were in the form of UFSAR revisions. Since it is not typical to include results of parametric studies in the UFSAR, only the results of the limiting cases were presented in the submittal of the application of the methodology. The engineering calculations which document the parametric studies are available for audit. It is concluded that this condition has been adequately addressed.

The SER for Revision 1 dated December 28, 1995 has the following conditions in Section 4.0. The SER for Revision 2 dated April 26, 1996 does not have any new limitations or conditions.

a. The acceptability of the use of DPC's approach to FSAR analysis is subject to the conditions of SERs on all aspects of transient analysis and methodologies (DPC-NE-3000, DPC-NE-3001, DPC-NE-3002, DPC-NE-2004, DPC-NE-2005) as well as the SERs on RETRAN and VIPRE computer codes.

Resolution: This condition has been addressed in this submittal.

b. There are scenarios in which an SGTR event may result in loss of subcooling and the consequent two-phase flow conditions in the primary system. In such instances, the use of RETRAN is not acceptable without a detailed review of the analysis.

Resolution: The McGuire and Catawba UFSAR SGTR analyses do not result in a loss of subcooling.

c. In the future if hardware or methodology changes, selection of limiting transients needs to be reconsidered, and DPC is required to perform sensitivity studies to identify the initial conditions in such a way to avoid conflict between transient objective, such as DNB and worst primary pressure.

   Resolution: Duke's methodology, as described in DPC-NE-3002-A, does select initial and boundary conditions with consideration of the possibility that different selections and possibly separate analyses may be necessary depending on the acceptance criteria and the margin to the acceptance criteria. This approach will be continued for future re-analyses due to hardware or methodology changes.

d. It is emphasized that, when using the SCD methodology to determine DNBR, the range of applicability of the selected CHF correlation must not be violated.

   Resolution: Duke recognizes the need to restrict the use of CHF correlations to within their ranges of applicability. Any deviations from this approach will be submitted for review and approval.

e. DPC's assumption of 120% of design pressure as part of the acceptance criteria for Reactor Coolant Pump Locked Rotor is not acceptable. DPC is required to use 110% of design pressure for that limit.

   Resolution: Duke has revised DPC-NE-3002-A to use 110% of design pressure as an acceptance criterion for the locked rotor accident (See Section 4.3 of DPC-NE-3002-A, Revision 2).

Westinghouse LOCA Topical Reports

Westinghouse will provide the requested information regarding SER limitations, conditions, and restrictions for the LOCA-related topical reports referenced by DPC-NE-2009-P and to be used for the RFA and transitional RFA/Mark-BW cores. This information will be submitted to the NRC by April 1, 1999.

12. Section 8.0 states that TS Figure 2.1.1-1 for the reactor core safety limits will be modified by deleting the 2455 psia safety limit line and making the 2400 psia safety limit line as the upper bound pressure allowed for power operation. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2400 psia safety limit line is within the range of the CHF correlations for the Mark-BW and RFA fuel designs.

However, the safety limit lines in Figure 2.1.1-1 were based on the CHF correlation for the Mark-BW fuel design, in addition to the hot leg boiling limit. Has an analysis been performed to ensure these safety limit lines bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design?

Response :

Yes. As stated, the 2400 psia line was selected since it was already defined for the Mark-BW fuel. Using the reference power distribution and the reactor inlet conditions defined by the hot leg boiling and DNB portions of the 2400 psia Safety Limit Line, the MDNBR was calculated using the full RFA core VIPRE-01 model and the WRB-2M CHF correlation. Additionally, the transition RFA/Mark-BW cores were also evaluated to ensure the established limits were conservative. The MDNBR values were greater than the design DNBR limit for all of the cases in both evaluations.

13. **TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor $F_q(x,y,z)$ and the enthalpy rise hot channel factor $F_{\Delta H}(x,y)$ to be measured periodically using the incore detector system to ensure the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limits are not violated. To avoid the possibility that these hot channel factors may increase beyond their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the $F_q(x,y,z)$ or $F_{\Delta H}(x,y)$ has decreased since the previous surveillance. For the reactor core containing the RFA fuel design with integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. Section 8.1 proposes to remove the 2% penalty value from these surveillance requirements and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide "typical values" for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors.**

    (a) **Provide the actual values of the margin-decrease penalty factors, as well as the bases for these values.**

    (b) **Provide references for the approved methodologies used to calculate these values, and to be included in TS 5.6.5 as a part of acceptability for COLR.**

Response 13a:

Margin decrease penalty factors will be calculated for each reload core. The actual margin decrease penalty factors for the initial transition core can not be provided until the final design for this core is complete. The cycle-specific factors for each core design will be included in each units' cycle-specific Core Operating Limits Report (COLR).

The methodology used to calculate the $F_q(x,y,z)$ and $F_{\Delta H}(x,y)$ margin decrease penalty factors is described below. The peaking factors used to calculate the margin decrease penalty factors are obtained from the analysis performed to establish operational axial flux difference limits as described in DPC-NE-2011PA ("Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors).

Nuclear Heat Flux Hot Channel Factor, Fq:

$F_q(x,y,z)$ is measured periodically using the incore detector system to ensure that the value of the total peaking factor, F_q-RTP, assumed in the accident analysis is bounding. The frequency requirement for this measurement is 31 effective full power days (EFPD). In order to account for the possibility that $F_q(x,y,z)$ may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If extrapolation of the measurement indicates that the $F_q(x,y,z)$ measurement would exceed the $F_q(x,y,z)$ limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on the available margin, or the $F_q(x,y,z)$ measurement would be increased by an appropriate penalty and compared against the $F_q(x,y,z)$ operational and RPS surveillance limits to ensure allowable total peaking limits are not exceeded.

The $F_q(x,y,z)$ penalty factor is calculated by projecting the change in the [

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[

] The Fq margin decrease factor may be applied directly to the measured Fq or may be incorporated into the $M_q(x,y,z)$ and $M_c(x,y,z)$ margin factors as described in DPC-NE-2011PA. For burnup ranges where the Fq margin decrease factor is less than 1.02, a value of 1 02 will be maintained.

Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}$.

The nuclear enthalpy rise hot channel factor, $F_{\Delta H}(x,y)$, is measured periodically using the incore detector system to ensure that fuel design criteria are not violated and accident analysis assumptions are not violated. The frequency requirement for this measurement is 31 effective full power days (EFPD). In order to account for the possibility that $F_{\Delta H}(x,y)$ may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If extrapolation of the measurement indicates that the $F_{\Delta H}(x,y)$ measurement would exceed the $F_{\Delta H}(x,y)$ surveillance limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on the available margin, or the $F_{\Delta H}(x,y)$ measurement would be increased by an appropriate penalty and compared against the $F_{\Delta H}(x,y)$ surveillance limit to ensure allowable peaking limits are not exceeded.

The $F_{\Delta H}(x,y)$ penalty factor is calculated by projecting the change in the [

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[ ]

[

] The $F_{\Delta H}$ margin decrease factor may be applied directly to the measured $F_{\Delta H}$ or may be incorporated into the $M_{\Delta H}(x,y)$ margin factors. For burnup ranges where the $F_{\Delta H}$ margin decrease factor is less than 1.02, a value of 1.02 will be maintained.

Response 13b:

The methodology used to calculate the $F_{\Delta H}(x,y)$ and $F_q(x,y,z)$ margin-decrease peaking penalty factors was described in answer 13a. Duke intends to reference this topical report (DPC-NE-2009) in Technical Specification 5.6.5 for the approved methodology used to calculate these parameters.

Duke Energy.

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April 7, 1999

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

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

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

Response to NRC Requests for Additional Information on License Amendment Requests for McGuire and Catawba Nuclear Stations

By letters dated December 9, 1998 and January 5, 1999 the NRC requested additional information on Duke Energy Corporation's July 22, 1998 license amendment requests (LARs) for the McGuire Nuclear Station, Units 1 & 2; and the Catawba Nuclear Station, Units 1 & 2 Technical Specifications. These LARs would permit use of Westinghouse fuel at McGuire and Catawba. Topical Report DPC-NE-2009P/DPC-NE-2009 was also included in the July 22, 1998 Duke submittal.

By letter dated January 28, 1999, Duke Energy Corporation responded to the thirteen questions contained in the December 9, 1998 and January 5, 1999 NRC letters. However, the response to Question No. 11 was incomplete, pending Duke's receipt of additional information from Westinghouse Electric Company. Duke has now received this information from Westinghouse and hereby submits this to the NRC. This information is contained in a Westinghouse letter dated March 31, 1999 which is included as the attachment to this letter.

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc (w/Attachment):

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Catawba Owners: NCMPA-1, NCEMC, PMPA, SREC
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Catawba RGC File 801.01 (T. K. Pasour)
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Westinghouse Electric Company

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March 31, 1999

DPC-99-016

Mr. G. Swindlehurst, Section Manager
Safety Analysis, Nuclear Generation
Duke Power Company
P. O. Box 1006
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DUKE POWER COMPANY
SER Restriction Evaluations for SB and LB LOCA

Dear Mr. Swindlehurst:

In response to your request for assistance in responding to NRC Question 11 (included as Attachment 1), Westinghouse has prepared a response which addresses Westinghouse's compliance with restrictions imposed by NRC Safety Evaluation Reports (SERs) related to the Westinghouse 1985 SBLOCA Evaluation Model with NOTRUMP (References 1-4) and the 1981 Evaluation Model with BASH (References 5-13). Attachment 2 provides the NOTRUMP SER Restriction Compliance Summary. Attachment 3 contains the large break LOCA SER compliance information.

References:

1. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", N. Lee, et al., August 1985.
2. WCAP-1054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model", C. M. Thompson, et al., July 1997.
3. WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code", S. D. Rupprecht, et al., 1986.
4. WCAP-14710-P-A, "1-D Heat conduction Model for Annular Fuel Pellets", D. J. Shimeck, May 1988.
5. WCAP-10484-P-A, "Spacer Grid Heat Transfer Effects During Reflood", J. S. Chiou, et al., March 1991.
6. WCAP-10484-P-A, Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood", D. J. Shimeck, December 1992.
7. WCAP-10266-P-A, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using BASH", M. Y. Young, et al., March 1987.

8. WCAP-10266-P-A, Addendum 1, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 1: Power Shape Sensitivity Studies", M. Y. Young, et al., January 1987.
9. NTD-NRC-95-4518, "Withdrawal of WCAP-12909-P on Power Shape Sensitivity Model (PSSM)", August 1995. [copy attached to DPC-95-224, "LOCA Axial Power Shape Sensitivity Model", K. B. Hanahan, August 1995.]
10. WCAP-9220-P-A, Revision 1, "Westinghouse ECCS Evaluation Model - 1981 Version", February 1982.
11. WCAP-8471-P-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information", April 1975.
12. WCAP-8354-P-A, Supplement 1, "Long-Term Ice Condenser containment LOTIC Code Supplement 1", T. Hsieh, et al., July 1974.
13. ET-NRC-92-3746, "Extension of NUREG-0630 Fuel Rod Burst Strain and Assembly Blockage Models to High Fuel Rod Burst Temperatures", N. J. Liparulo, September 1992.

If you have any questions, please call Mr. John Besspiata at 412-374-4524 or me at 412-374-5651.

Sincerely,

Dwain W. Alexander
Customer Projects Manager

cc: J. J. Besspiata, W
S. P. Shaver, W Charlotte

bcc: J. M. Leonelli
D. W. Alexander
M. J. Boyles

NRC Question 11, as received:

11. The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current McGuire/Catawba cores not having the RFA design. For example, DPC-NE-1004A, PDC-NE-2011-PA, DPC-NF-2010A, and DPC-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation models described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC SERs in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.

NOTRUMP SER Restriction Compliance Summary

The following document contains a synopsis of the NRC imposed Safety Evaluation Report (SER) restrictions/requirements and the Westinghouse compliance status related to these issues. Not all the items identified are clearly SER restrictions, but sometimes state the NRC's interpretation of the Westinghouse Evaluation Methodology utilized for a particular aspect of the Small Break Loss Of Coolant (LOCA) Evaluation Model.

WCAP-10054-P-A

WCAP-10054-P-A is titled "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," and is dated August, 1985. The following summarizes the SER restrictions and requirements associated with this WCAP:

SER Wording (Page 6)

"The use of a single momentum equation implies that the inertias of the separate phases can not be treated. The model therefore would not be appropriate for situations when separate inertial effects are significant. For the small break transients, these effects are not significant."

SER Compliance

Inherent compliance due to the use of a single momentum equation.

SER Wording (Page 8)

"To assure the validity of this application, the bubble diameter should be on the order of 10^{-1}-2 cm. As long as steam generator tube uncovery (concurrent with a severe depressurization rate) does not occur, this option is acceptable."

SER Compliance

Westinghouse complies with this restriction for all Appendix-K licensing basis calculations. Typical Appendix-K calculations do not undergo a significant secondary side system depressurization in conjunction with steam generator tube uncovery due to the modeling methodology utilized.

SER Wording (Page 14)

"The two phase multiplier used is the Thom modification of the Martinelli-Nelson correlation. This model is acceptable per 10 CFR Part 50 Appendix K for LOCA analysis at pressure above 250 psia"

SER Compliance

The original NOTRUMP model was limited to no less than 250 psia since the model, as contained in the NOTRUMP code, did not contain information below this range. Westinghouse extended the model to below 250 psia, as allowed by Appendix K paragraph I-C-2, and reported these modifications to the NRC via the 1995 annual reporting period (NSD-NRC-96-4639).

SER Wording (Page 16)

"Westinghouse, however, has stated that the separator models are not used in their SBLOCA analyses."

SER Compliance

Westinghouse does not model the separators in the secondary side of the steam generators for Appendix-K Small Break LOCA analyses; therefore, compliance exists.

SER Wording (Pages 16-17)

"Axial heat conduction is not modeled." and "Deletion of clad axial heat conduction maximizes the peak clad temperature."

SER Compliance

The Westinghouse Small Break LOCA is comprised of two computer codes, the NOTRUMP code which performs the detailed system wide thermal hydraulic calculations and the LOCTA code which performs the detailed fuel rod heatup calculations. The NOTRUMP code does not model axial conduction in the fuel rod and therefore complies. The LOCTA code has always accounted for axial conduction as is clearly stated in WCAP-14710-P-A which supplements the original NOTRUMP documentation.

SER Wording (Page 17)

"...; critical heat flux, W-2, W-3, or Macbeth, or GE transient CHF (the W-2 and W-3 correlations are used for licensing evaluations);..."

SER Compliance

The information presented here indicates that the NRC apparently misstated that Westinghouse was utilizing the W-2,W-3 correlations for Critical Heat Flux (CHF) in the fuel rod heat transfer model. A review of the analyses performed by Westinghouse, including those in WCAP-11145-P- A, indicates that the Macbeth CHF correlation has been utilized for all Appendix-K analyses performed by Westinghouse. This is consistent with the slab heat transfer map as described in WCAP-10054-P-A. In addition, the Macbeth correlation is specifically called out in Appendix K I-C-4-4 as an acceptable CHF model.

In a supplemental response to NRC questions (Specifically question 440.1 found in Appendix-A of WCAP-10054-P-A, Page A-10), a description of the core model describes the Macbeth as being utilized as the CHF correlation in the NOTRUMP Small Break LOCA model.

SER Wording (Page 21)

"The standard continuous contact model is not appropriate for vertical flow,..."

SER Compliance

The standard continuous contact flow links are not utilized when modeling vertical flow in the Appendix-K NOTRUMP Evaluation Model analyses; therefore, compliance is demonstrated.

SER Wording (Page 27)

"..., the hardwired choice of one fuel pin time step per coolant time step should result in sufficient accuracy."

SER Compliance

The NOTRUMP code continues to utilize only one fuel pin time step per coolant time step and therefore complies with this requirement.

SER Wording (Page 47)

"The code options available to the user but not applied in licensing evaluations were not reviewed."

SER Compliance

Westinghouse complies with this requirement.

SER Wording (Page 53)

"4. Steam Interaction with ECCS Water, a. Zero Steam Flow in the Intact Loops While Accumulators Discharge Water."

SER Compliance

Per paragraph I-D-4 Appendix-K, the following is stated:

> "During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption."

As can be seen, the specific Appendix-K wording can be considered applicable to Large Break LOCAs only since Small Break LOCAs do not undergo a true refill/reflood period. However, the Westinghouse Small Break LOCA Evaluation Model methodology is such that for break sizes in which the intact loop seal restriction is not removed (WCAP-11145-P-A Page 2-11), steam flow through the intact loop(s) is automatically (artificially) restricted via the loop seal model. While not specifically limited to zero, the flow is drastically reduced via the application of the artificial loop seal restriction model.

For breaks sizes above which the loop seal restriction is removed (typically >= 6 inch diameter breaks), this criterion is not explicitly adhered to. The implementation of the COSI condensation model into NOTRUMP (As approved by the NRC in WCAP-10054-P-A, Addendum 2, Revision 1), which is based on additional experimental documentation and improved modeling techniques, more accurately models the interaction of steam with Emergency Core Cooling Water in the cold leg region. This experimental documentation supports the more accurate modeling of steam/water interaction in the cold leg region as allowed by Appendix-K. Note however that even with the COSI condensation model active, the accumulator injection condensation model still utilizes the conservative model as originally licensed in the NOTRUMP code.

SER Wording (Page 7 of enclosure 2)

"Per generic letter 83-35, compliance with Action Item II.K.3.31 may be submitted generically. We require that the generic submittal include validation that the limiting break location has not shifted away from the cold legs to the hot or pump suction legs."

SER Compliance

Westinghouse submitted WCAP-11145-P-A in support of generic letter 83-35 Action Item II.K.3.31. As part of this effort, verification was provided which documented that the cold leg break location remains limiting.

WCAP-10054-P-A, Addendum 2, Revision 1

WCAP-10054-P-A, Addendum 2, Revision 1 is titled "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," and is dated July 1997. The following summarizes the SER restrictions and requirements associated with this WCAP:

SER Wording (Page 3)

"It is stated in Ref. 5 that the range of injection jet velocities used in the experiments brackets the corresponding rates in small break LOCAs for Westinghouse plants and that the model will be used within the experimental range. Also in References 1 and 5 Westinghouse submitted analyses demonstrating that the condensation efficiency is virtually independent of RCS pressure and state that the COSI model will be applied within the pressure range of 550 to 1200 psia."

SER Compliance

The coding implementation of the COSI model correlation in the NOTRUMP model restricts the application of the COSI condensation model to a default pressure range of 550 to 1200 psia and limits the injection flow rate to a default value of 40 lbm/sec-loop. The value of 40 lbm/sec-loop corresponds to the 30 ft./sec velocity utilized in the COSI experiments. As such, the default NOTRUMP implementation of the COSI condensation model complies with the applicable SER restrictions.

WCAP-11145-P-A

WCAP-11145-P-A, is titled "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code," and is dated 1986. No specific SER restrictions were provided by the NRC as part of this WCAP review; however, the SER contains verification that the requirements of Item II.K.3.31 have been satisfied (i.e. break location study).

SER Wording (Page 5)

"We therefore, find that the requirements of NUREG-0737, Item II.K3.31, as clarified by Generic Letter 83-35, have been satisfied.

We find that a condition of the safety evaluation for NOTRUMP as applied to Item II.K.3.30 has been satisfied. The limiting cold leg break size for a 4-loop plant was reanalyzed at pump suction and at hot leg locations. The results confirmed that the cold leg break was limiting."

WCAP-14710-P-A

WCAP-14710-P-A, is titled "1-D Heat Conduction Model for Annular Fuel Pellets," and is dated May 1998. No specific SER restrictions are provided by the NRC in this document; however, a conclusion was reached regarding the modeling of annular pellets during Small Break LOCA event.

SER Wording

"Based on its conclusions that the explicit modeling of annular pellets, as described in WCAP-14710(P), provides a more realistic representation in W Appendix K ECCS evaluation models of the annular pellets, while retaining conservatism in those evaluation models, the staff finds that the explicit modeling of annular pellets, as described in WCAP-14710(P), in W Appendix K LOCA evaluation models permits those models to continue to satisfy the regulations to which they were approved, and is, therefore, acceptable for incorporation into those models."

SER Compliance

Westinghouse performs sensitivity studies to assess the impact of modeling annular pellets on plant specific analyses.

LARGE BREAK LOCA SER COMPLIANCE

Over the years a number of SERs have been issued with specified restrictions on model applications. The individual WCAP titles which have been published with an SER included that have conditions or limitations relevant to the BASH Evaluation Model have been reviewed in the context of the Catawba 2 Cycle 11 large break LOCA analysis. The relevant restrictions, limitations, and conditions specific to the BASH Evaluation Model as imposed by the NRC are listed, together with the means by which they are resolved in the Westinghouse Catawba Unit 2 Cycle 11 large break LOCA analysis.

WCAP-10484-P-A

WCAP-10484-P-A is titled "Spacer Grid Heat Transfer Effects during Reflood" and is dated March, 1991. The following summarizes the SER restrictions and requirements associated with this WCAP:

SER Wording - "Acceptance of the droplet breakup model is premature due to the limited..information..." (page 16 of the WCAP-10484 SER)

SER Compliance - The droplet breakup model has been deleted from the LOCBART computer code and is not used in any BASH Evaluation Model (EM) analysis.

SER Wording - "The length average heat transfer coefficient h(Z/L) in the node should be used in applying the Y-H-L (Yao-Hochreiter-Leech) correlation." (page 16 of the WCAP-10484 SER)

SER Compliance - A review of the LOCBART computer code, Version 17.0 logic to compute grid single phase heat transfer enhancement identified the Y-H-L correlation value was not being averaged over the length of the node in some situations. This discovery led to the preparation of a Nonconformance Report concerning this error. LOCBART Version 18.0, which corrects the subject error, has been created and documented for use in the Catawba 2 Cycle 11 large break LOCA BASH EM analysis.

SER Wording - "The use of BART with grid rewet models should be restricted to the range of conditions consistent with the data base tested as indicated in Table 1 of this SER." (page 16 of the WCAP-10484 SER)

SER Compliance - WCAP-10484-P-A Addendum 1 supersedes the "BART with grid rewet" simulations presented here and is the reference for validation of these LOCBART models within the current BASH EM. The Catawba BASH EM analyses are "under the limitations delineated in (that) report," quoting the SER letter for the Addendum 1 WCAP.

WCAP-10266-P-A, Revision 2

WCAP-10266-P-A is titled "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code " and is dated March, 1987. The following is a review of the SER restrictions and requirements found in this WCAP:

SER Wording - "The EM has no downward quench capability and therefore cannot be used for the analysis of either upper head injection plants or upper plenum injection plants." (page 10 of the WCAP-10266 SER)

SER Compliance - The BASH EM has not been applied in the large break LOCA analysis of any plant equipped with upper head injection or upper plenum injection. The upper head injection system has been removed from service at the Catawba Units.

SER Wording - "Westinghouse has committed to continue to analyze the large break LOCA with both minimum and maximum safety injection to confirm which produces the limiting large break scenario for each plant." (page 11 of the WCAP-10266 SER)

SER Compliance - The maximum SI scenario will be analyzed for the limiting discharge coefficient DECLG break as identified in the Catawba Unit 2 Cycle 11 BASH EM analysis.

SER Wording - "Westinghouse has committed to submit confirmatory analyses with the first BASH plant calculation of each type (2, 3 and 4 loop) to demonstrate that the cosine power shape is limiting and is the appropriate power shape to use for licensing calculations." (page 11 of the WCAP-10266 SER)

SER Compliance - This SER requirement was originally fulfilled for 3 and 4 loop plants via sensitivity studies presented in WCAP-10266-P-A, Addendum 1, Revision 2. However, the current BASH EM power shape methodology is to use an explicit approach introduced in 1995 for top-skewed power distributions, which was noticed to the NRC in NTD-NRC-95-4518. The disposition of this issue for the Catawba Units has been included in the PCT Margin Utilization Sheets, beginning with the 1995 10CFR50.46 Annual Reporting. The Catawba Unit 2 Cycle 11 BASH EM analysis will consider power shape effects consistent with the current methodology.

WCAP-9220-P-A, Revision 1

WCAP-9220-P-A, Revision 1 is titled "Westinghouse ECCS Evaluation Model - 1981 Version" and is dated February, 1982. This WCAP documents a number of changes to the existing large break LOCA EM which were implemented to correct errors and/or to obtain more favorable PCT results. Two separate Safety Evaluation Reports, dated August 29, 1978 and December 1, 1981 are included in the WCAP. The following is a review of the SER restrictions and requirements found in this WCAP which remain applicable to the BASH Evaluation Model. First, from the August 29, 1978 SER:

SER Wording - "Westinghouse has recently decided to cancel requests for using (Dougall-Rohsenow post-CHF heat transfer) correlation in place of the Westinghouse transition boiling correlation." (page xiv of WCAP-9220 Revision 1)

SER Compliance - The Westinghouse transition boiling correlation continues to be used in BASH EM calculations.

The restrictions and requirements identified in the December 1981 SER are discussed below:

SER Wording - "Based on the data and analyses contained in NUREG-0630....we find the (algorithm for computing heatup rates; rupture, strain, and blockage models; the prerupture strain model and the artificial limit on the degree of swelling), and their proposed applications to be acceptable." (page B-15 of WCAP-9220 Revision 1)

SER Compliance - The NUREG-0630 fuel rod burst and blockage models are programmed into the SATAN and LOCBART codes and are applied in BASH EM computations, including Catawba Unit 2 Cycle 11. Note that the SER restricted the usage of the NUREG-0630 model to calculated burst temperatures less than 950°C. In letter ET-NRC-92-3746 dated September 16, 1992 Westinghouse described the extension of the NUREG-0630 modeling to conservatively consider burst temperatures greater than 950°C.

WCAP-8471-P-A

WCAP-8471-P-A is titled "The Westinghouse ECCS Evaluation Model: Supplementary Information" and is dated April, 1975. The SER for this WCAP covers the entire set of WCAP reports that documented the Westinghouse model originally created and submitted when 10CFR50 and its Appendix K first became effective in 1974; it contains no restrictions and requirements as such other than for LOTIC as noted below.

SER Wording - "Until such time that LOTIC is modified to resolve the staff concerns.... a conservative minimum containment pressure of zero psig must be assumed in ECCS analyses of plants using an ice condenser containment." (page 3 of the WCAP-8471 SER)

SER Compliance - Approval of the LOTIC-2 computer code for ECCS minimum containment pressure analysis of ice condenser containments was obtained in the SER of WCAP-8354-P-A, Supplement 1. This code version has the capability of modeling additional ice condenser phenomena as required by the NRC. LOTIC-2 will be employed in the Catawba Unit 2 Cycle 11 BASH EM analysis.

Section G

Revision 1 RAI Letters and Responses

1. August 7, 2001, letter from M. S. Tuckman to NRC, Response for NRC Request for Additional Information – TAC nos. MB3222, MB3223, MB3343, and MB3344) and License Amendment Request Supplement, Attachment 5

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Michael S. Tuckman
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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:

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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. Nelms, Notary Public

My commission expires: JAN 22, 2006

SEAL

bxc w/Attachments:

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Attachment 5
Responses to NRC Concern on
Topical Report Numbered DPC-NE-2009-P, Revision 1 Westinghouse Fuel Transition Report

NRC Concern: The proposed Revision 1 for Topical Report DPC-NE-2009 added a new reference document designated as Reference 6-39. This document, WCAP-15085, *Model Changes to the Westinghouse Appendix K Small Break LOCA NOTRUMP Evaluation Model: 1988 – 1997*, has not been approved by the NRC. In an NRC/Duke telephone conference held on July 24, 2002, NRC officials expressed a concern with the Duke proposal to reference an unapproved topical report in DPC-NE-2009-P, Revision 1.

Response

WCAP-15085 is a compilation of 10 CFR 50.46 reports related to the Westinghouse SBLOCA evaluation model previously reported individually to the NRC by Westinghouse pursuant to 10 CFR 50.46. The reference to WCAP-15085 has been deleted from Duke's proposed Revision 1 to DPC-NE-2009-P, since it is not totally applicable to McGuire and Catawba. Only those 10 CFR 50.46 reports applicable to McGuire and Catawba (identified as References 6-22, 6-28, and 6-39) are now referenced in the proposed Revision 1 to DPC-NE-2009-P. This is consistent with the current NRC-approved Revision 0 of DPC-NE-2009-P. Appropriate changes have been made to the affected pages of DPC-NE-2009-P and are included within Attachment 5 in both marked and reprinted versions.

the RELAP5 model, which is used to model the mass and energy release from LOCAs, are also anticipated. The RETRAN and RELAP5 model changes for the RFA design are not significant enough to require reanalyses. Future reanalyses will incorporate the RFA design model revisions.

6.5 LOCA Analyses

Large and small break LOCA analyses will be performed by Westinghouse using approved versions of the Westinghouse Appendix K LOCA evaluation models. All features employed have been approved by the NRC as required and annual model reports for the evaluation models have been supplied to the NRC, the most recent of which is found in Reference 6-22. Therefore, no NRC review of the evaluation model features is necessary, and only methodology with respect to analyzing McGuire/Catawba will be presented in this section. New LOCA analyses will be performed to support the licensing of McGuire/Catawba during the transition and full core operation of the RFA design.

6.5.1 Small Break LOCA

For small break LOCAs (SBLOCAs) due to breaks less than 1 ft^2, Westinghouse developed the NOTRUMP computer code (Reference 6-23) to calculate the transient depressurization of the reactor coolant system (RCS) as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 6-24, 6-25, 6-26, and 6-27) was developed and licensed by Westinghouse to determine the RCS response to design basis SBLOCAs, and to address NRC concerns expressed in NUREG-0737, Item II.K.3.30. ^ Insert A

The NRC approved noding scheme for the NOTRUMP Evaluation Model is shown in Reference 6-24, although minor noding changes to facilitate the modeling of broken loop ECCS were instituted and reported to the NRC in Reference 6-28. Peak cladding temperature (PCT) calculations are performed with the LOCTA-IV code (Reference 6-29) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. Additional modifications to the LOCTA-IV code to allow the modeling

Insert A

In addition, several model enhancements have been made to the evaluation model and implemented via the 10 CFR 50.46 process. These enhancements or changes were determined to be non-significant as defined by 10 CFR 50.46. Westinghouse reported these enhancements to the NRC in annual notification reports (References 6-22, 6-28 and 6-39) and implemented them on a forward fit basis. Duke did not report these changes in their annual 10 CFR 50.46 reports since the Westinghouse SBLOCA analysis using these enhancements had not been implemented for McGuire and Catawba during this time period. The purpose of identifying these enhancements in this report is to clearly identify the SBLOCA analysis method to be used to support McGuire and Catawba.

6-38 WCAP-10484-P-A Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood", September 1993.

6-39 NSD-NRC-99-5839, "1998 Annual Notification of ~~[illegible]~~ Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46 (a)(3)(ii).

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Duke Power Company Westinghouse Fuel Transition Report, Revision 1

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the RELAP5 model, which is used to model the mass and energy release from LOCAs, are also anticipated. The RETRAN and RELAP5 model changes for the RFA design are not significant enough to require reanalyses. Future reanalyses will incorporate the RFA design model revisions.

6.5 LOCA Analyses

Large and small break LOCA analyses will be performed by Westinghouse using approved versions of the Westinghouse Appendix K LOCA evaluation models. All features employed have been approved by the NRC as required and annual model reports for the evaluation models have been supplied to the NRC, the most recent of which is found in Reference 6-22. Therefore, no NRC review of the evaluation model features is necessary, and only methodology with respect to analyzing McGuire/Catawba will be presented in this section. New LOCA analyses will be performed to support the licensing of McGuire/Catawba during the transition and full core operation of the RFA design.

6.5.1 Small Break LOCA

For small break LOCAs (SBLOCAs) due to breaks less than 1 ft^2, Westinghouse developed the NOTRUMP computer code (Reference 6-23) to calculate the transient depressurization of the reactor coolant system (RCS) as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 6-24, 6-25, 6-26, and 6-27) was developed and licensed by Westinghouse to determine the RCS response to design basis SBLOCAs, and to address NRC concerns expressed in NUREG-0737, Item II.K.3.30.

In addition, several model enhancements have been made to the evaluation model and implemented via the 10 CFR 50.46 process. These enhancements or changes were determined to be non-significant as defined by 10 CFR 50.46. Westinghouse reported these enhancements to the NRC in annual notification reports (References 6-22, 6-28 and 6-39) and implemented them on a forward fit basis. Duke did not report these changes in their annual 10 CFR 50.46 reports since the Westinghouse SBLOCA analysis using these enhancements had not been implemented for McGuire and Catawba during this time period. The purpose of identifying these enhancements in this report is to clearly identify the SBLOCA analysis method to be used to support McGuire and Catawba.

The NRC approved noding scheme for the NOTRUMP Evaluation Model is shown in Reference 6-24, although minor noding changes to facilitate the modeling of broken loop ECCS were instituted and reported to the NRC in Reference 6-28. Peak cladding temperature (PCT) calculations are performed with the LOCTA-IV code (Reference 6-29) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. Additional modifications to the LOCTA-IV code to allow the modeling

6-38 WCAP-10484-P-A Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood", September 1993.

6-39 NSD-NRC-99-5839, "1998 Annual Notification of Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10CFR50.46 (a)(3)(ii)".

Section H

Revision 2 RAI Letters and Responses

1. July 26, 2002, letter from NRC to M. S. Tuckman, Review of Duke Topical Report DPC-NE-2009, Revision 2
2. September 9, 2002, letter from M. S. Tuckman to NRC, Response to NRC Request for Additional Information

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

July 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 - REVIEW OF DUKE TOPICAL REPORT DPC-NE-2009, REVISION 2 (TAC NOS. MB4502, MB4503, MB4504 AND MB4505)

Dear Mr. Tuckman:

The Nuclear Regulatory Commission is reviewing your application dated February 28, 2002, entitled "Topical Report DPC-NE-2009, Revision 2 - Updates to Chapters 2, 4, and 5" and has identified a need for additional information as identified in the Enclosure. These issues were discussed with your staff on July 24, 2002. Please provide a response to this request within 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 II
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

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST APPLICABLE TO

REVISIONS TO TOPICAL REPORT DPC-NE-2009, REVISION 2

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DUKE ENERGY CORPORATION

The staff has reviewed Duke Energy Corporations's submittal dated Febuary 28, 2002, "Topical Report DPC-NE-2009, Revision 2 - Updates to Chapters 2, 4, and 5" and has identified a need for the following additional information.

1. Section 5.3 of DPC-NE-2009, Revision 2, states that the WRB-2M critical heat flux (CHF) correlation will be used for the robust fuel assembly (RFA) design, whereas the BWU-N CHF correlation will be applied for the non-mixing vane span of the RFA fuel.

   A. Discuss the applicability of the BWU-N correlation to the RFA non-mixing vane span. The discussion should include whether the RFA fuel design is within the range of the test assemblies data base used to develop the BWU-N correlation. The test assemblies data base parameters include the fuel diameter, pitch, hydraulic diameters, grid design (grid thickness, height, and vane design), grid spacing, and heated length.

   B. The WRB-2M correlation described in WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," is applicable to the 17 x 17 fuel with 0.374 inch outer diameter rods and modified low pressure drop grids, with or without modified intermediate flow mixing grids. Is the WRB-2M correlation not applicable to the RFA non-mixing vane span? Why is the BWU-N correlation used?

   C. Discuss how two different correlations are applied to the different spans of a fuel assembly. Is the VIPRE-01 code programmed to automatically perform the switch in the correlations? Has verification and validation been done to ensure correctness of VIPRE-01 in the correlation switch?

2. For the transition cores with co-existence of the RFA and Mark-BW fuel designs, Section 5.7 of Revision 2 of the report, states that a transition core departure from nucleate boiling ratio penalty for the RFA design is determined using the 8 channel RFA/Mark-BW transition core model for initial transition reload cycles, and using the 75 channel model for subsequent cycles where the RFA fuel composes greater than 80 percent of the assemblies in the core.

   Explain why it is necessary to use different core models depending on whether the RFA fuel composes greater than 80 percent of the assemblies.

Duke Energy

M. S. Tuckman
Executive Vice President
Nuclear Generation

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September 9, 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 50-370
Catawba Nuclear Station - Units 1 and 2
Docket Nos. 50-413 and 50-414
Topical Report DPC-NE-2009, Revision 2 - Updates to Chapters 2, 4, and 5 (TAC Nos. MB4502, MB4503, MB4504, MB4505)

Response to NRC Request for Additional Information

By letter dated July 26, 2002, the NRC requested additional information regarding Topical Report DPC-NE-2009, Revision 2, "Updates to Chapters 2, 4, and 5." The questions contained in the July 26, 2002 NRC letter, and the corresponding Duke answers, are provided in the attachment to this letter.

If there are any questions or additional information is needed on this matter, please call A. Jones-Young at (704) 382-3154.

Very truly yours,

M S Tuckman

M.S. Tuckman

ATTACHMENT

xc:

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A. D. Jones-Young
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ATTACHMENT

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST APPLICABLE TO

REVISIONS TO TOPICAL REPORT DPC-NE-2009, REVISION 2

CATAWBA NUCKEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DUKE ENERGY CORPORATION

The staff has reviewed Duke Energy Corporation's submittal dated February 28, 2002, "Topical Report DPC-NE-2009, Revision 2 - Updates to Chapters 2, 4, and 5" and has identified a need for the following information.

1. Section 5.3 of DPC-NE-2009, Revision 2, states that the WRB2-M critical heat flux (CHF) correlation will be used for the robust fuel assembly (RFA) design, whereas the BWU-N CHF correlation will be applied for the non-mixing vane span of the RFA fuel.

    A. Discuss the applicability of the BWU-N correlation to the RFA non-mixing vane span. The discussion should include whether the RFA fuel design is within the range of test assemblies data base used to develop the BWU-N correlation. The test assemblies data base parameters include the fuel diameter, pitch, hydraulic diameters, grid design (grid thickness, height, and vane design), grid spacing, and heated length.

    B. The WRB-2M correlation described in WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux In 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," is applicable to the 17x17 fuel with 0.374 inch outer diameter rods and modified low pressure drop grids, with or without modified intermediate flow mixing grids. Is the WRB-2M correlation not applicable to the RFA non-mixing vane span? Why is the BWU-N correlation used?

C. Discuss how two different correlations are applied to the different spans of a fuel assembly. Is the VIPRE-01 code programmed to automatically perform the switch in the correlations? Has verification and validation been done to ensure correctness of the VIPRE-01 in the correlation switch?

2. For the transition cores with co-existence of the RFA and Mark-BW fuel designs, Section 5.7 of Revision 2 of the report, states that a transition core departure from nucleate boiling ratio penalty for the RFA design is determined using the 8 channel RFA/Mark-BW transition core model for the initial transition reload cycles, and using the 75 channel model for subsequent cycles where RFA fuel composes greater than 80 percent of the assemblies in the core.

   Explain why it is necessary to use different core models depending on whether the RFA fuel composes greater than 80 percent of the assemblies.

1. *Section 5.3 of DPC-NE-2009, Revision 2, states that the WRB2-M critical heat flux (CHF) correlation will be used for the robust fuel assembly (RFA) design, whereas the BWU-N CHF correlation will be applied for the non-mixing vane span of the RFA fuel.*
   A. *Discuss the applicability of the BWU-N correlation to the RFA non-mixing vane span. The discussion should include whether the RFA fuel design is within the range of test assemblies data base used to develop the BWU-N correlation. The test assemblies data base parameters include the fuel diameter, pitch, hydraulic diameters, grid design (grid thickness, height, and vane design), grid spacing, and heated length.*

The BWU-N correlation is based on local conditions (pressure, mass flux, local quality) that bound the operation of the RFA fuel at McGuire and Catawba. The following table compares the geometry parameters for the RFA design against the BWU-N correlation:

Parameter	RFA Fuel	BWU-N Database
Fuel Diameter	0.374	0.379 - 0.430
Rod Pitch	0.496	0.501 - 0.590
Hydraulic Diameter	0.375 - 0.464	0.39 - 0.60
*Grid Spacing (inches)	20.5	21.0
Heated Length (feet)	12	6 - 12

* - In the span of interest

BWU-N is one of a series of CHF correlations developed to apply to PWR cores with mixing or non-mixing vane spacer grids. In each of the approved correlations, the correlated independent variables were the thermal-hydraulic local conditions (pressure, mass velocity and equilibrium thermodynamic quality at CHF), axial flux shape (via the F factor), heated length, and the grid axial spacing. The geometric independent variables such as rod diameter, pitch to diameter ratio, hydraulic or heated diameters were found to be non-correlated (that is, there was no sensitivity in CHF level for geometric independent variables) and thus these parameters were not needed as part of the correlation.

Even though the geometric variables were found to be non-correlated, it would be improper to make large extrapolations of these geometric variables. Only very small extrapolations are necessary to apply BWU-N to RFA fuel. This is shown in the following table:

Geometric Variable	**RFA Application**	**BWU-N Data Base**	**Difference, %**
Pin Pitch, in.	0.496	0.501	1.0
Rod Diameter, in.	0.374	0.379	1.3
Pitch to Diameter Ratio	0.496/0.374 = 1.326	0.501/0.379 = 1.322	0.3
Unit Hydraulic Diameter, in.	0.4635	0.4642	0.2

The grid design is the same in that BWU-N is being applied to the RFA fuel only above a non-mixing vane grid. There are no vanes present on the grid in question. The grid heights and thickness are within 0.026 and 0.003 inches, respectively. As explained above, these parameters have no significant impact on the CHF performance in a non-mixing vane span.

Table 4-3 of Reference 1 limits BWU-N to Non-Mixing Grids. Thus, the use of BWU-N is based on:

1. the geometric similarity of the designs
2. the fact that the geometric variables are not included (needed) in the base BWU correlations and
3. the fact that BWU-N results in conservative levels of CHF compared to the mixing vane correlations.

In summary, CHF performance is influenced by the presence or absence of mixing vanes and the local conditions. There are no specific grid features to enhance thermal performance in the span of interest and the local conditions are bounded. Therefore, BWU-N can be applied to the non-mixing vane span of the RFA assembly and will predict lower CHF (conservative) than the mixing vane grid correlations.

B. The WRB-2M correlation described in WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux In 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," is *applicable to the 17x17 fuel with 0.374 inch outer diameter rods and modified low pressure drop grids, with or without modified intermediate flow mixing grids. Is the WRB-2M correlation not applicable to the RFA non-mixing vane span? Why is the BWU-N correlation used?*

The WRB-2M correlation was developed from fuel with mixing vane modified LPD mid-grids, modified LDP IFM grids, and non-vaned end grids. All the CHF data from the test program documented in WCAP-15025-P-A was in a region above one of the mixing vane grid types. Therefore, the WRB-2M correlation is directly applicable to regions of the fuel above a modified LPD mixing vane grid of either type. The very bottom span of the RFA fuel assembly (lower ~21 inches of the heated length) is above an Inconel grid without any type of mixing vane. For this region of the fuel assembly, Duke considers the use of the BWU-N non-mixing vane grid correlation to be appropriate and conservative as discussed in the answer to question 1 (A).

C. *Discuss how two different correlations are applied to the different spans of a fuel assembly. Is the VIPRE-01 code programmed to automatically perform the switch in the correlations? Has verification and validation been done to ensure correctness of the VIPRE-01 in the correlation switch?*

The VIPRE-01 computer code solves the sets of equations for the geometry modeled and the boundary conditions specified to determine a converged fluid solution. This converged fluid solution yields the local conditions at each node and elevation modeled. After the fluid solution is converged, all the inputs for the CHF correlation (local pressure, mass flux, enthalpy, etc.) are fixed and the DNBR calculation is performed. Therefore, the calculation of CHF and DNBR has no effect on the converged fluid solution.

Due to this, VIPRE-01 has the built-in capability to calculate DNBR with multiple CHF correlations. Each correlation is applied to all channels at all elevations. Since the switch in this case is based solely on grid type and elevation, two options are available to apply the BWU-N correlation:

- manually overlay the output of the code after selecting both correlations
- program the code to automatically switch based on grid elevation inputs

For the current application of BWU-N on the RFA fuel, the manual process was used. The automatic switching from WRB-2M to BWU-N was not programmed into VIPRE-01. However, the VIPRE-01 code has been programmed by Duke to automatically perform the switch in other applications such as the Mark-BW (BWU-N to BWU-Z) and the Advanced Mark-BW (BWU-N to BWU-Z/MSM) and may be added to this application (RFA) in the future.

The verification and validation of the manual overlay process is performed by the independent review of the calculation results in the standard quality assurance process. The verification and validation of an automatic switchover by elevation is performed in the code revision process by performing independent calculations of the correct critical heat flux value from the local fluid conditions in the channel. This independent calculation by elevation of the critical heat flux is compared against the code output for cases to confirm the switch is being performed correctly.

2. *For the transition cores with co-existence of the RFA and Mark-BW fuel designs, Section 5.7 of Revision 2 of the report, states that a transition core departure from nucleate boiling ratio penalty for the RFA design is determined using the 8 channel RFA/Mark-BW transition core model for the initial transition reload cycles, and using the 75 channel model for subsequent cycles where RFA fuel composes greater than 80 percent of the assemblies in the core.*

   Explain why it is necessary to use different core models depending on whether the RFA fuel composes greater than 80 percent of the assemblies.

The RFA fuel assembly contains 3 extra grids, the IFM grids, compared to the Mark-BW assembly. These extra grids in the upper span force flow out of the RFA assemblies and into the surrounding Mark-BW assemblies. In the 8 channel model, the single hot assembly (RFA) is modeled by the first 7 channels and the remainder of the core (Mark-BW fuel) is lumped into one single channel. Therefore, in the 8 channel transition model, there is one RFA assembly surrounded by 192 Mark-BW assemblies. This maximizes the hydraulic difference in transition cores and creates a very bounding penalty for the RFAs.

The loss of flow in the upper spans of the RFA is the major element of the DNB penalty. This hydraulic effect of flow reduction in the RFA is a direct function of the number of RFA and Mark-BW assemblies incore. As subsequent cores of RFA fuel are loaded, only a few Mark-BW assemblies remain. As fewer Mark-BWs are present, the simple 8 channel model becomes overly conservative for the RFAs in transition. The only option to better reflect the physical effects of the last transition cycles is to increase the detail in VIPRE-01 to the 75 channel model. This more detailed model better represents the hydraulic effects of cores where most of the fuel is RFA where a small fraction (less than 20% or fewer than 38 assemblies) of the core is Mark-BW. The 80% value was selected because it corresponds to approximately two batches of RFA fuel residing incore. With this more detailed 75 channel model, a conservative penalty is still determined in the same manner as with the 8 channel model.

This approach of using a more detailed transition core model was discussed previously in Reference 2 [response to Question 2] for Mark-BW/OFA transition at McGuire/Catawba and Reference 3 [response to Question 2(d)] for the Mark-B11/Mark-B10 transition at Oconee.

References

1) BAW-10199P-A, August, 1996, "The BWU CHF Correlations", D. A. Farnsworth and G. A. Meyer.
2) Letter from M.S. Tuckman to USNRC, Supplemental Information to Assist in Review of Topical Reports DPC-NE-3000 and DPC-NE-2004, August 29, 1991 (included in Attachment D of DPC-NE-2004P-A, Revision 1)
3) Letter from M.S Tuckman to USNRC, Response to NRC Request for Additional Information on Appendix D to Topical Report DPC-NE-2005-P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, September 21, 1998 (included in Appendix D of DPC-NE-2005P-A, Revision 2)