Duke Energy

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February 28, 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 (TAC Nos. MA2359, MA2361, MA2411, MA2412), Revision 2 - Updates to Chapters 2, 4, and 5

Reference: Duke Energy Corporation to NRC letter, dated, July 24, 2001, Topical Report DPC-NE-2009P, Westinghouse Fuel Transition Report, Revision 0, December 1999

Attached are revisions to Chapters 2, 4 and 5 of Topical Report DPC-NE-2009. The proposed changes are separated into two different categories:

1. Changes that require review and approval and
2. Administrative updates

The following information explains the type of change for each affected chapter.

The enclosed revisions to Chapter 5, Sections 5.2, 5.7, 5.8 and the addition of Figures 5-1 through 5-3 are modifications to increase the reference peaking values for Westinghouse RFA fuel at McGuire and Catawba Nuclear Stations. This increase is due to additional DNB performance margin inherent in the fuel design. These changes require review and approval by

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the NRC prior to implementation.

Several other updates have been identified as being necessary for accuracy. These updates are considered administrative in nature. The changed pages as attached will be incorporated in the final approved version of DPC-NE-2009, Revision 2 submitted to the NRC.

Sections 2.0 and 2.1
These sections are being revised to confirm batch implementation of the QRTN top nozzle design feature. The justification for this change is presented in "Reference 2-6".

Table 2.1
This section is being revised to correct a typographical error.

Section 4.0 (page 4-2)
This section is being revised to correct a typographical error.

Sections 4.0, 4.1, 4.2.6.1, 4.2.8.1 and 4.2
These sections are being revised to show implementation of PAD 4.0 computer code. The justification for this change is presented in the letter to the NRC, dated, July 24, 2001 (referenced above).

Section 5.3
This section is being revised to clarify the non-mixing vane span DNB correlation. The justification for this change is presented in a 10CFR50.59 evaluation.

Attachment A provides the updates to DPC-NE-2009 that contains information that Duke considers PROPRIETARY. In accordance with 10CFR 2.790, Duke requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of the applicable information is also included with this letter. Attachment B provides the non-proprietary version of the revised sections as a separate document.

NRC approval is requested by March 1, 2003 in order to support Catawba Nuclear Station Unit 2 Beginning-of-Cycle 13 Startup activities.

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

ATTACHMENTS

xc: with Proprietary and Non-Proprietary Versions

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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:

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

   b) 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.

   c) 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.

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

M. S. Tuckman

M. S. Tuckman

(Continued)

e) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the Duke Topical Report designated DPC-NE-2009P, Revision 0, *"Duke Power Company Westinghouse Fuel Transition Report,"* and omitted from the non-proprietary version. This information is being submitted to the NRC for review and approval as an ATTACHMENT to this Duke letter. The marked information enables Duke to:

(i) Respond to Generic Letter 83-11, Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions.
(ii) Perform core design, fuel rod design, and thermal-hydraulic analyses for the Westinghouse Robust Fuel Assembly design.
(iii) Simulate UFSAR Chapter 15 transients and accidents for McGuire and Catawba Nuclear Stations.
(iv) Perform safety evaluations per 10CFR50.59.
(v) Support Facility Operating Licenses/Technical Specifications amendments for McGuire and Catawba Nuclear Stations.

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

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

ii) 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 on this 28TH day of

February, 2002

Mary P. Nelms
Notary Public

My Commission Expires:

JAN 22, 2006

SEAL

ATTACHMENT B
NON-PROPRIETARY

Duke Power Company

DPC-NE-2009, Rev. 2

DUKE POWER COMPANY WESTINGHOUSE FUEL TRANSITION REPORT

Original Version: July 1998
Approved Version: December 1999

Revision 1 Submitted: August 2001

Revision 2 Submitted: February 2002

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

Duke Power Company Westinghouse Fuel Transition Report

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2

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		

2

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 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. 2

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

2

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,

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.

2

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.

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 to [ ] (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

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

2

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.

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.

4-3 S. L. Davidson & T. L. Ryan, "Vantage+ Fuel Assembly Reference Core Report", WCAP-12610-P-A, April 1995.

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.

4-6 Risher, D., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis", WCAP-8963-P-A, August 1978.

4-7 W. J. O'Donnell and B. F. Langer, "Fatigue Design Basis of Zircaloy Components", Nuclear Science and Engineering, 20, 1-12, 1964.

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.

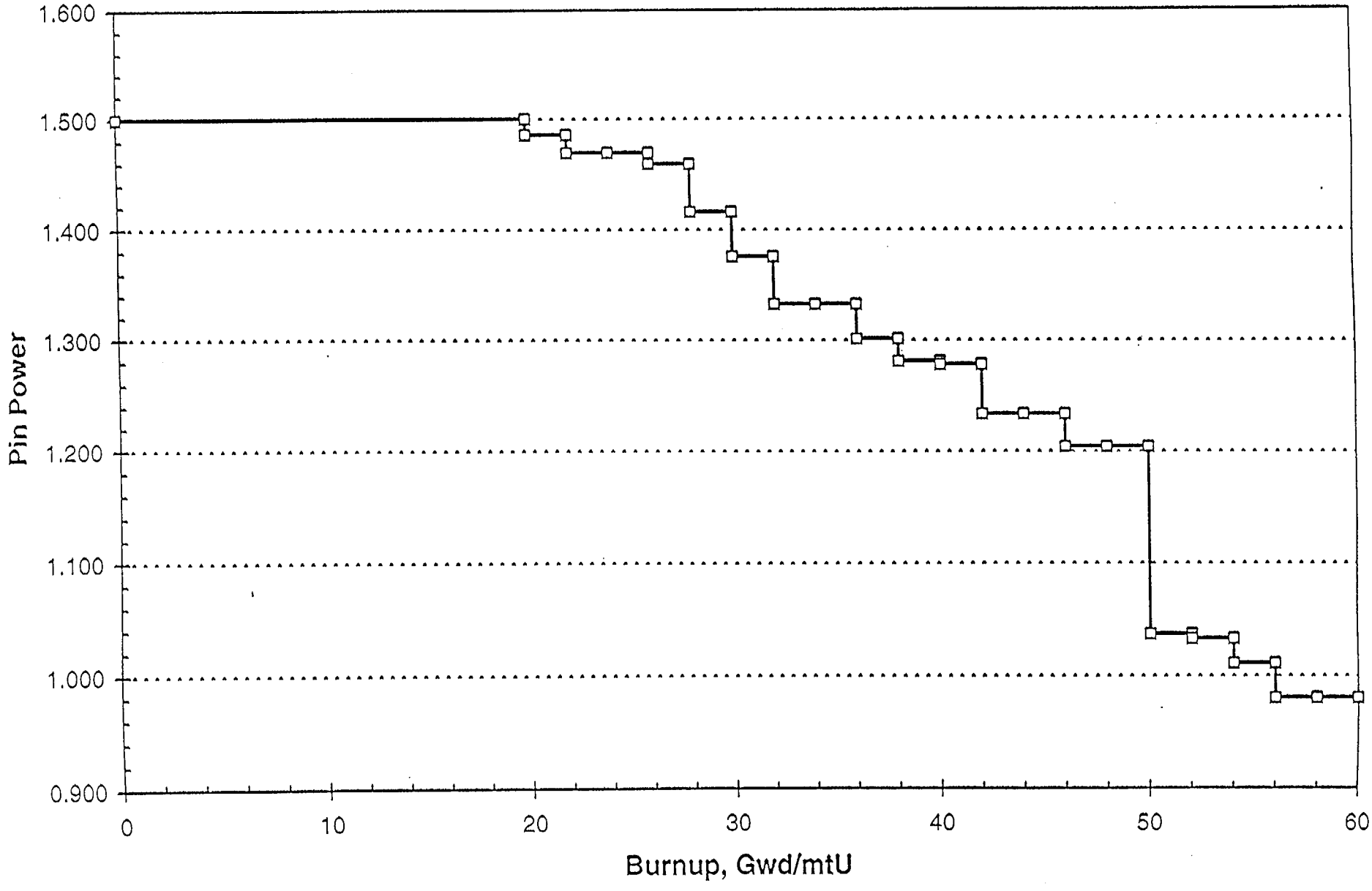
4-11 "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011P-A, March 1990.

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 VIPRE-01 models approved in Reference 5-1 for the Mark-BW fuel are used to analyze the RFA design with the following changes: 2

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 1.67. The associated pin power distribution was also updated based on this higher value. 2
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 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. 2

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

ATTACHMENT A
PROPRIETARY