Attachment 2

NON-PROPRIETARY

Duke Power Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX E

DPC-NE-2005

Duke Power Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX E

McGuire/Catawba Plant Specific Data

Advanced Mark-BW Fuel

BWU-Z CHF Correlation

Submitted: September 2001

This Appendix contains the plant specific data and statistical DNB limits for the McGuire and Catawba Nuclear Stations with the Advanced Mark-BW fuel design using the BWU-Z critical heat flux correlation. The thermal-hydraulic statistical core design analysis was performed as described in the main body of this report (DPC-NE-2005).

This appendix details the fuel assembly structural and thermal-hydraulic features unique to the Advanced Mark-BW fuel design. Two separate fuel pellet materials can be used in this structure. When used with uranium fuel pellets, the fuel assembly is called Advanced Mark-BW. If used with mixed oxide fuel pellets, the fuel assembly is called Mark-BW/MOX1. The fuel mechanical structure and grids are identical in each case, therefore the same critical heat flux correlation is applicable to both designs. The nuclear uncertainties used in this analysis bound both uranium and mixed oxide fuel rods. Therefore, the SCD analysis documented here is applicable to and bounds both the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs. For simplicity in this appendix, the term Advanced Mark-BW will be used.

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four loop Westinghouse PWR's) with the Advanced Mark-BW fuel. This fuel design incorporates a 17x17 fuel lattice with 0.374 inch outside diameter (OD) fuel rods, M5™ cladding, and three additional non-structural Mid-Span Mixing (MSM) grids in the upper fuel assembly spans to improve DNB performance. All the parameter uncertainties and statepoint ranges used in this analysis were selected to bound the unit and cycle specific system values at the McGuire and Catawba stations.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference E-3 and the McGuire/Catawba eight channel model approved in Reference E-1 are used in this analysis. The reference pin power distribution is the same as that used for the Westinghouse RFA fuel described in Reference E-4. The VIPRE-01 models, approved in Reference E-1 for the Mark-BW fuel, are used to analyze the Advanced Mark-BW fuel design with the following changes:

1) The Advanced Mark-BW fuel assembly geometry information is listed in Table E-1. Applicable form loss coefficients as per the vendor were used in the model.
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 LEVY to the EPRI model.

The Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 (void fractions of 0.85) and is discontinuous at a quality equal to 1.0 (Reference E-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 E-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 E-1, to the EPRI correlation. This change has been previously submitted and approved by the NRC for both the Westinghouse RFA fuel design (Reference E-4) and the Mark-B11 fuel design (DPC-NE-2005, Revision 2, Appendix D).

Critical Heat Flux Correlation

The BWU-Z critical heat flux correlation described in Reference E-2 is used for all statepoint analyses. This correlation was developed by Framatome Cogema Fuels and is applicable to the Advanced Mark-BW fuel design. The analysis in Reference E-2 was performed with the LYNXT thermal-hydraulic computer code. This correlation was programmed into the VIPRE-01 thermal-hydraulic computer code and the Advanced Mark-BW fuel database was analyzed in its entirety. The results of this analysis are shown in Table E-2. The resulting average measured to predicted (M/P) value and data standard deviation are within 1% and the CHF correlation limit with VIPRE-01 is 2% lower than the values in Reference E-2, page F-5 (also shown on Table E-2 under the LYNXT column).

Figures E-1 through E-4 graphically show the results of this evaluation. Figure E-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the database. Figures E-2 through E-4 show there is no bias with the VIPRE-01 calculated M/P ratios with respect to mass velocity, pressure, or thermodynamic quality. These figures compare closely with the same parameter representations in Reference E-2.

Based on the results shown in Table E-2 and Figures E-1 through E-4, the BWU-Z form of the BWU CHF application correlation can be used in DNBR calculations with VIPRE-01 for Advanced Mark-BW fuel.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table E-3. These statepoints represent the range of conditions to which the statistical DNB analysis limit will be applied.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table E-4. The uncertainties were selected to bound the values calculated for each parameter at McGuire and Catawba. As noted in Table E-4, the nuclear uncertainties used in this analyses bound both uranium and mixed oxide fuel. The resulting range of key parameter values generated in this analysis is listed on Table E-6.

Mixed Core Application

The mixed core model determines the impact of the geometric and hydraulic differences between the resident 17x17 Westinghouse RFA fuel described in Reference E-4 and the new Advanced Mark-BW design. The 8 channel model described in Reference E-1 is used to evaluate the impact of mixed cores containing Westinghouse RFA fuel and the Advanced Mark-BW fuel. In Figure 5 of Reference E-1, Advanced Mark-BW fuel is used instead of Mark-BW fuel. Therefore, the limiting assembly in Channels 1 through 7 are modeled as Advanced Mark-BW fuel and the remaining core, Channel 8, is modeled as Westinghouse RFA fuel. The mixed core analysis models each fuel type in those respective locations with the correct geometry. The form loss coefficients for each fuel design are input so the effect of crossflow between the different fuel types by elevation is calculated. This conservative mixed core model is used for all analyses since the equilibrium core reload cycles will contain both fuel types.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table E-5. Section 1 of Table E-5 contains the 500 case runs and Section 2 contains the 6000 case runs. The number of cases was increased from 5000 to 6000 as described in Attachment 1 of Revision 0 of DPC-NE-2005. The DNBR distributions for all statepoints in this analysis were normally distributed. It is seen from Section 2 of Table E-5 that the maximum statepoint statistical DNBR value is [ ]. Therefore, the statistical design limit using the BWU-Z CHF correlation for Advanced Mark-BW fuel at McGuire/Catawba was conservatively determined to be 1.36. This limit applies to mixed cores with Advanced Mark-BW and Westinghouse RFA fuel.

FIGURE E-1
Measured CHF versus Predicted CHF
Advanced Mark-BW Fuel Database

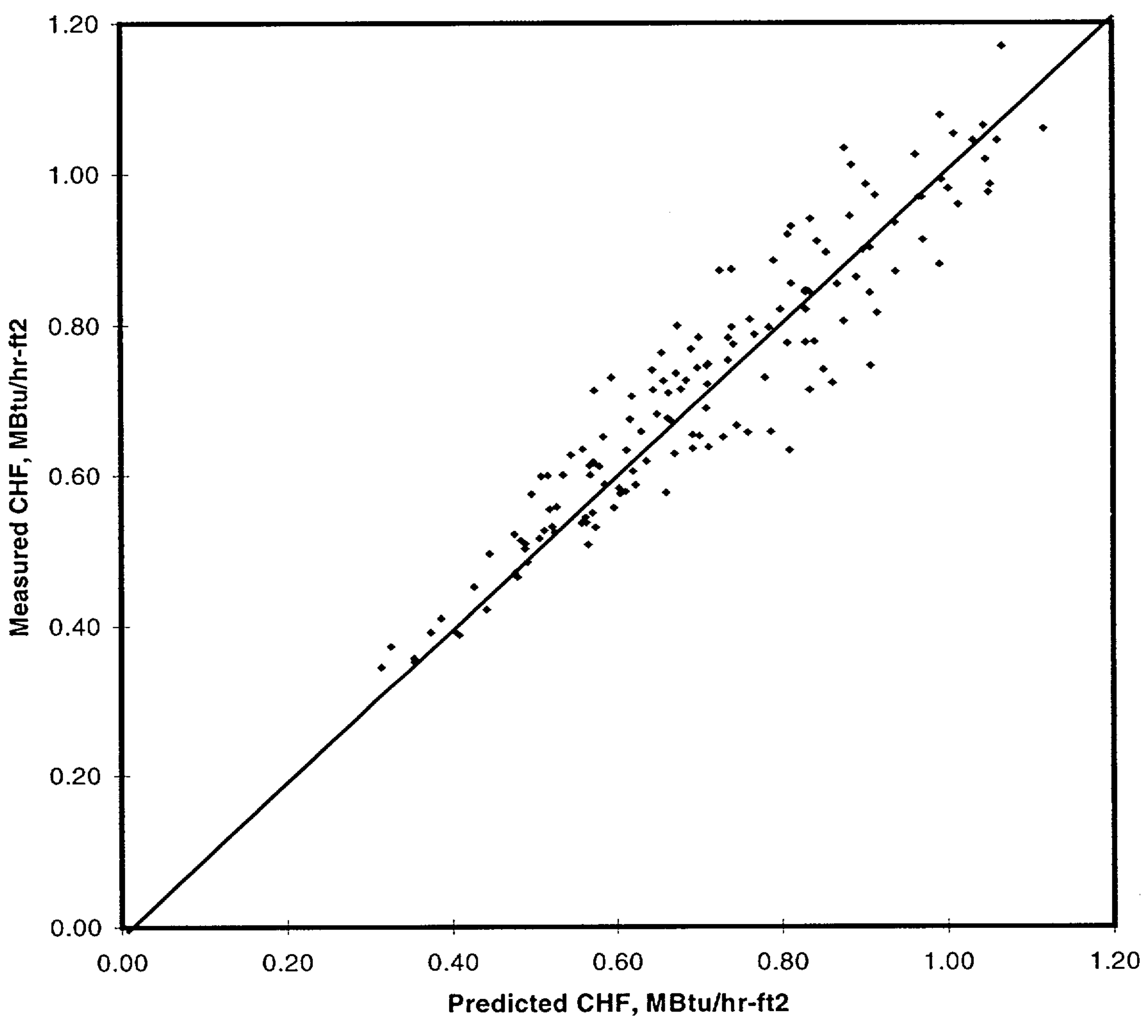

FIGURE E-2
Measured to Predicted CHF versus Mass Velocity
Advanced Mark-BW Fuel Data Base

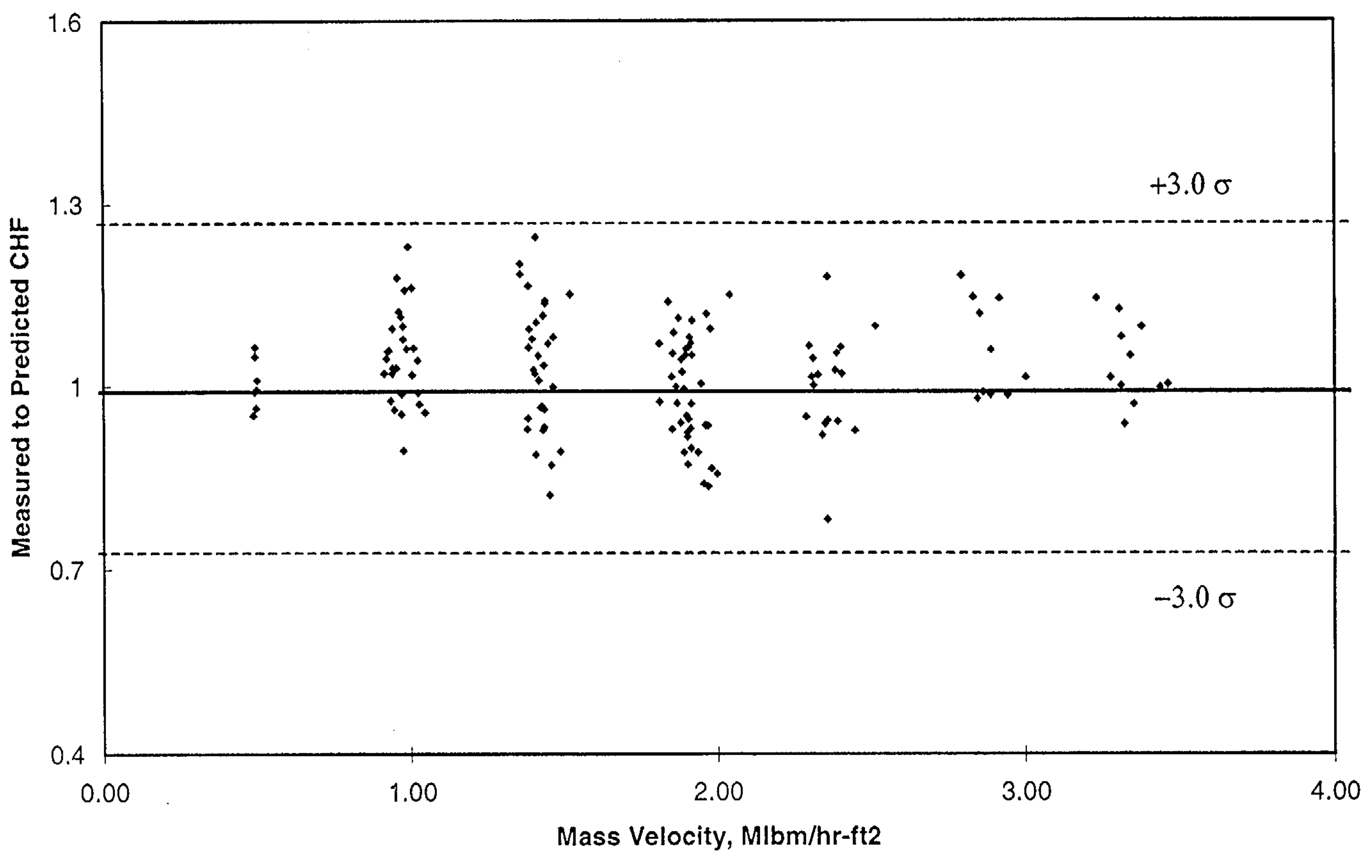

FIGURE E-3
Measured to Predicted CHF versus Pressure
Advanced Mark-BW Fuel Data Base

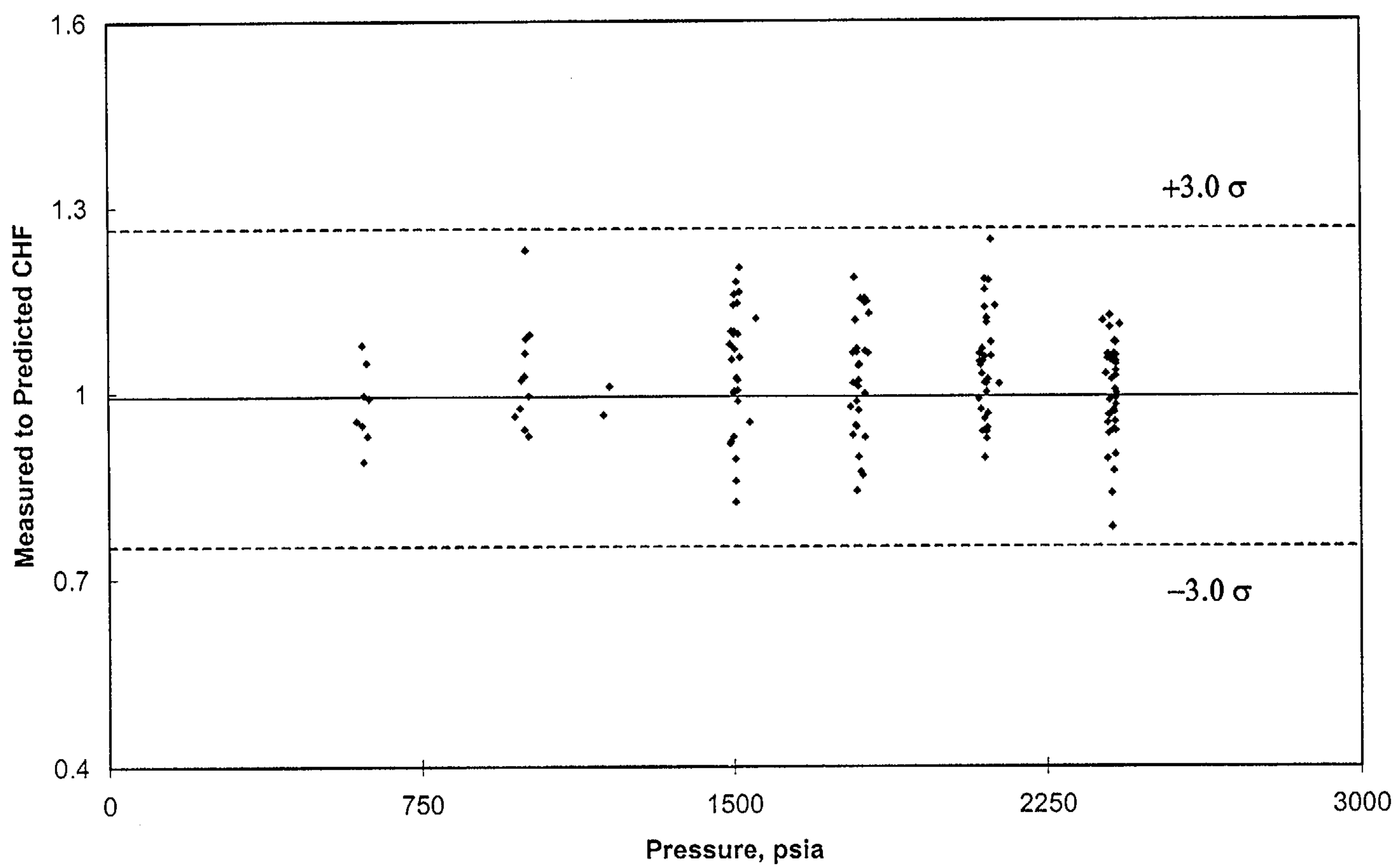

FIGURE E-4
Measured to Predicted CHF versus Quality
Advanced Mark-BW Fuel Data Base

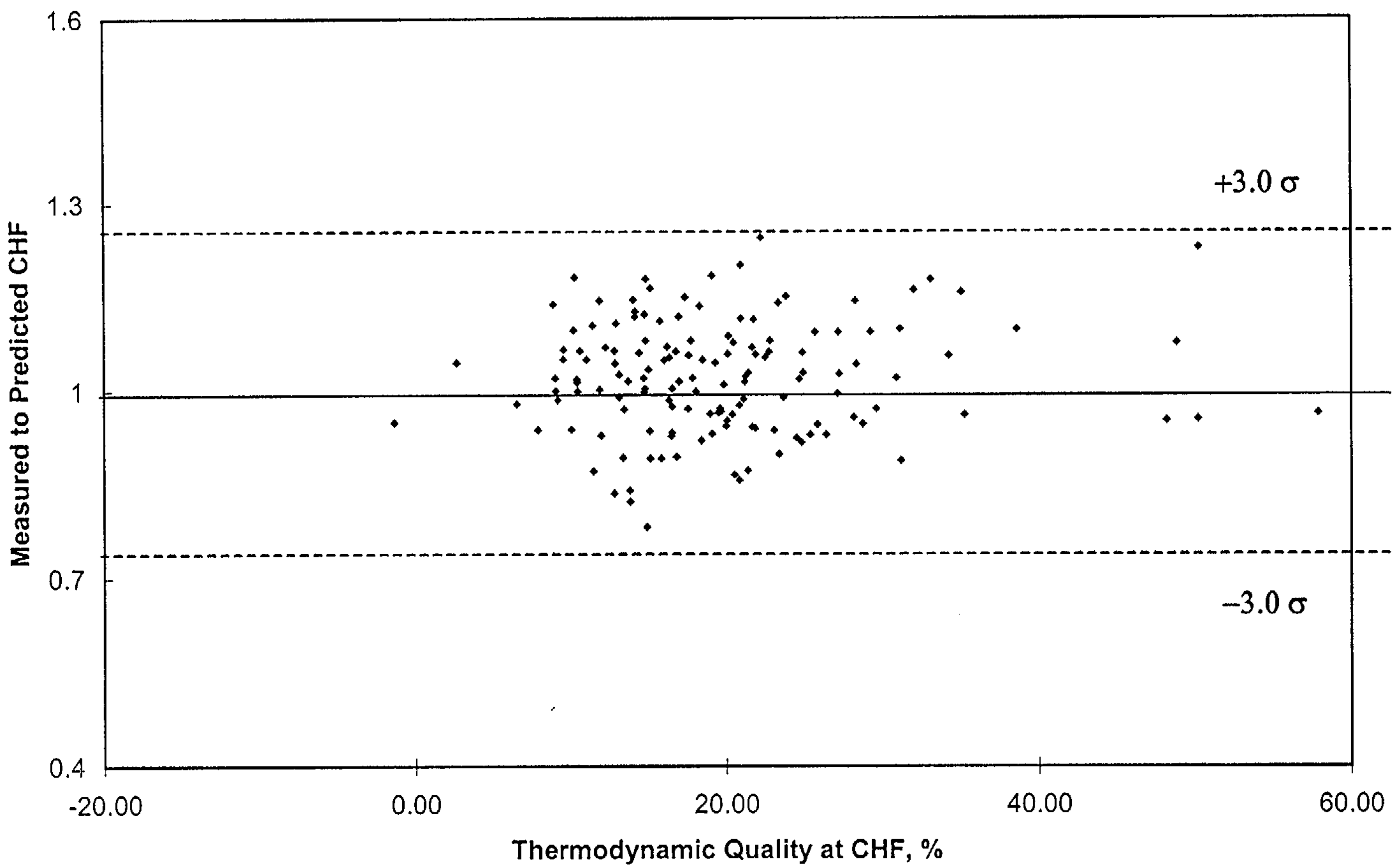

TABLE E-1 Advanced Mark-BW Fuel Assembly Data

(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	2	Upper and Lower	Non-Mixing Vane
	Zircaloy ($M5^{TM}$)	6	Intermediate	5 Vaned, 1 Vaneless
	Zircaloy ($M5^{TM}$)	3	Intermediate	Mid-Span Mixing (Non-structural)
Nozzles	304SS	1	Bottom	Fine Mesh
	304SS	1	Top	Quick Disconnect

TABLE E-2 CHF Test Database Analysis Results With VIPRE-01 Advanced Mark-BW Fuel, BWU-Z CHF Correlation

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # Of data	148	148
N, degrees of freedom (n-1)	147	147
M/P, Average measured to predicted CHF	1.0214	1.0138
σ (M/P/N)	0.0883	0.0920
K(147,0.95,0.95),one sided tolerance factor Ref. E-2)	1.872	1.872
DNBR(L)= 1 /(M/P - K * σ)	1.168	1.188

BWU-Z Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.36 to 3.55
Thermodynamic Quality at CHF	Less than 0.74
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Advanced Mark-BW, F_{MSM} = 1.18
Design Limit DNBR, VIPRE-01	1.19*

* The correlation design limit DNBR (1.19) applies only at or above the nominal pressure of 1000 psia (Reference E-2). In the low pressure region (below a pressure of 1000 psia) the design limit DNBR in the following table will be used (Reference E-2):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.20

TABLE E-3 McGuire/Catawba SCD Statepoints

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						
25						

* 100% RTP = 3411 Megawatts Thermal

\# FΔH is maximum pin peak

TABLE E-4 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
$F^E_{\Delta H}$	+/- 3.5% / 2.13%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.4% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	15.3% / +/- 9.27%	Normal
Code/Model	[ ]	Normal

* - Percentage of 100% RTP (68.22 MWth wherever applied).

TABLE E-4 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.

TABLE E-4 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^N_{\Delta H}$	
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. This uncertainty is bounding for both uranium and mixed oxide fuel.
$F^E_{\Delta H}$	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. This uncertainty bounds both uranium and mixed oxide fuel pellets.
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 physics code axial node length. The uncertainty distribution is conservatively applied as uniform.

TABLE E-4 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
DNBR	
Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The LYNXT value was used since the VIPRE-01 value was smaller. 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 E-5 McGuire/Catawba Statepoint Statistical Results

SECTION 1

BWU-Z Critical Heat Flux Correlation

500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
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TABLE E-5 Continued McGuire/Catawba Statepoint Statistical Results

SECTION 2

BWU-Z Critical Heat Flux Correlation

6000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
6				
7				
10				
11				
12				

TABLE E-6 McGuire/Catawba Key Parameter Ranges

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. Ranges are subject to change based on future statepoint conditions.

REFERENCES

E-1. DPC-NE-2004P-A, McGuire and Catawba Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, Revision 1, February 1997.

E-2. BAW-10199P, Addendum 2, Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids, submitted November 2000.

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

E-4. DPC-NE-2009P-A, Duke Power Company Westinghouse Fuel Transition Report, December 1999.