DPC-NE-2005P-A

Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX C

McGuire/Catawba Plant Specific Data

Mark-BW Fuel

BWU-Z CHF Correlation

April 1996

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

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four loop Westinghouse PWR's) with Mark-BW fuel assemblies as described in Reference C-1. The parameter uncertainties and statepoint ranges were selected to bound the unit and cycle specific values of the McGuire and Catawba stations.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference C-3 and the McGuire/Catawba eight channel code model approved in Reference C-1 are used in this analysis.

Critical Heat Flux Correlation

The BWU-Z form of the BWU critical heat flux correlation described in Reference C-2 is used for all statepoint analyses. This correlation was developed by BWFC for application to the Mark-BW fuel design. Reference C-2 was performed with the LYNXT thermal-hydraulic computer codes. The correlation was programmed into the VIPRE-01 thermal-hydraulic computer code by Duke Power Company and the BWU-Z CHF data base analyzed in its entirety. The results of this analysis are shown in Table C-1. The resulting Average M/P value and data standard deviation are within 1% of the values reported in Reference C-2.

Figures C-1 through C-5 graphically show the results of this evaluation. Figure C-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the data base. Figure C-2 shows a histogram of the VIPRE-01 M/P ratios for the 530 point data base. Figures C-3 through C-5 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 C-2.

Based on the results shown in Table C-1 and Figures C-1 through C-5, the BWU-Z form of the BWU CHF correlation licensed in Reference C-2 can be used in DNBR calculations with VIPRE-01 for Mark-BW fuel.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table C-2. These statepoints represent the range of conditions to which the statistical DNB analyses 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 C-2. The uncertainties were selected to bound the values calculated for each parameter at McGuire and Catawba. The resulting range of key parameter values generated in this analyses is listed on Table C-5.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table C-4. Section 1 of Table C-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 the report. All statepoint SDL values listed in this analysis are normally distributed. The maximum statepoint statistical DNBR value in Table C-4 for the 5000 case propagations was 1.364.

Therefore, the statistical design limit using the BWU-Z form of the BWU CHF correlation for Mark-BW fuel at McGuire/Catawba was conservatively determined to be [ ].

FIGURE C-1

Measured CHF Versus Predicted CHF

Mark-BW Data Base

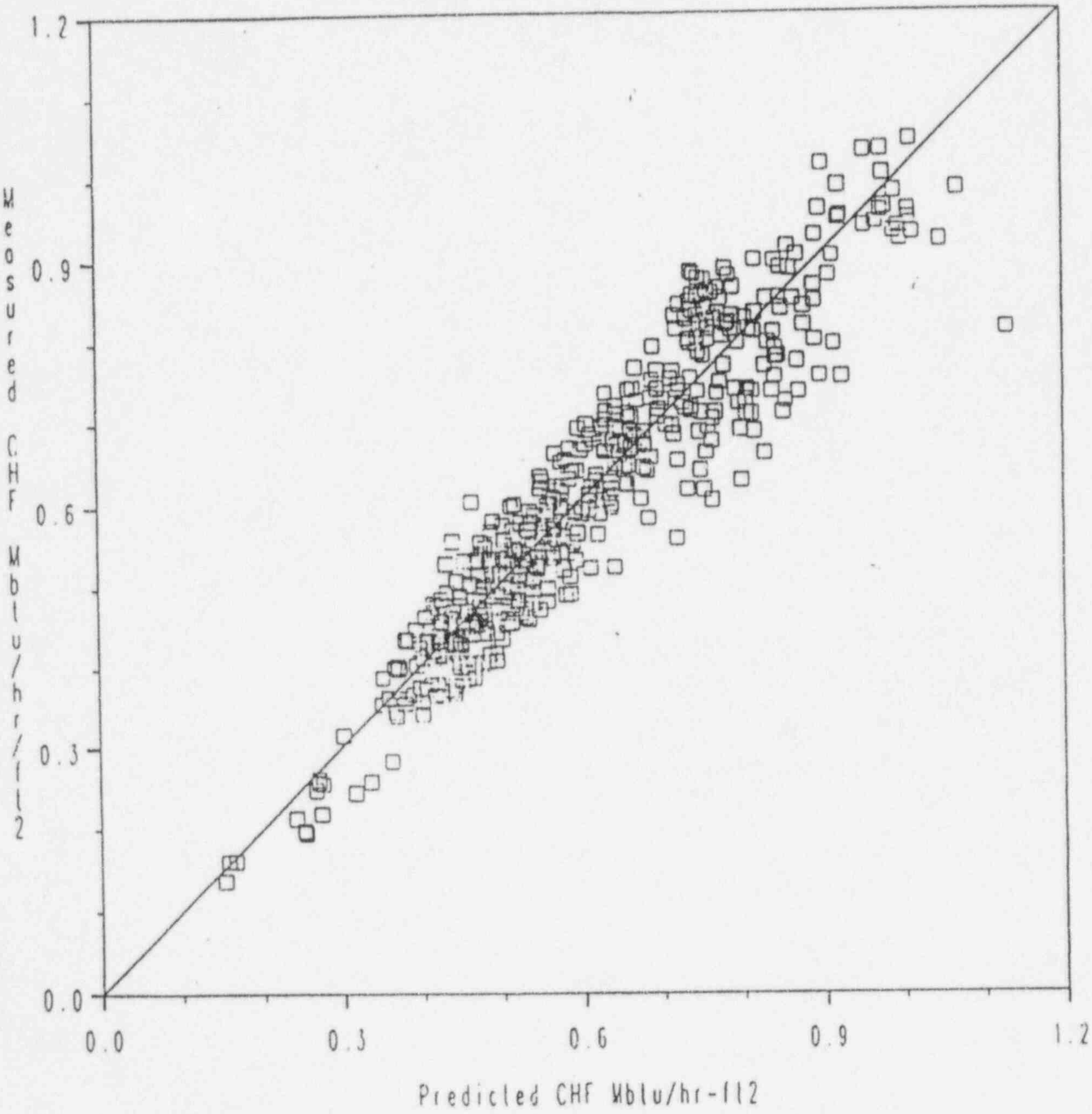

FIGURE C-2

Distribution of CHF Ratios

Mark-BW Data Base

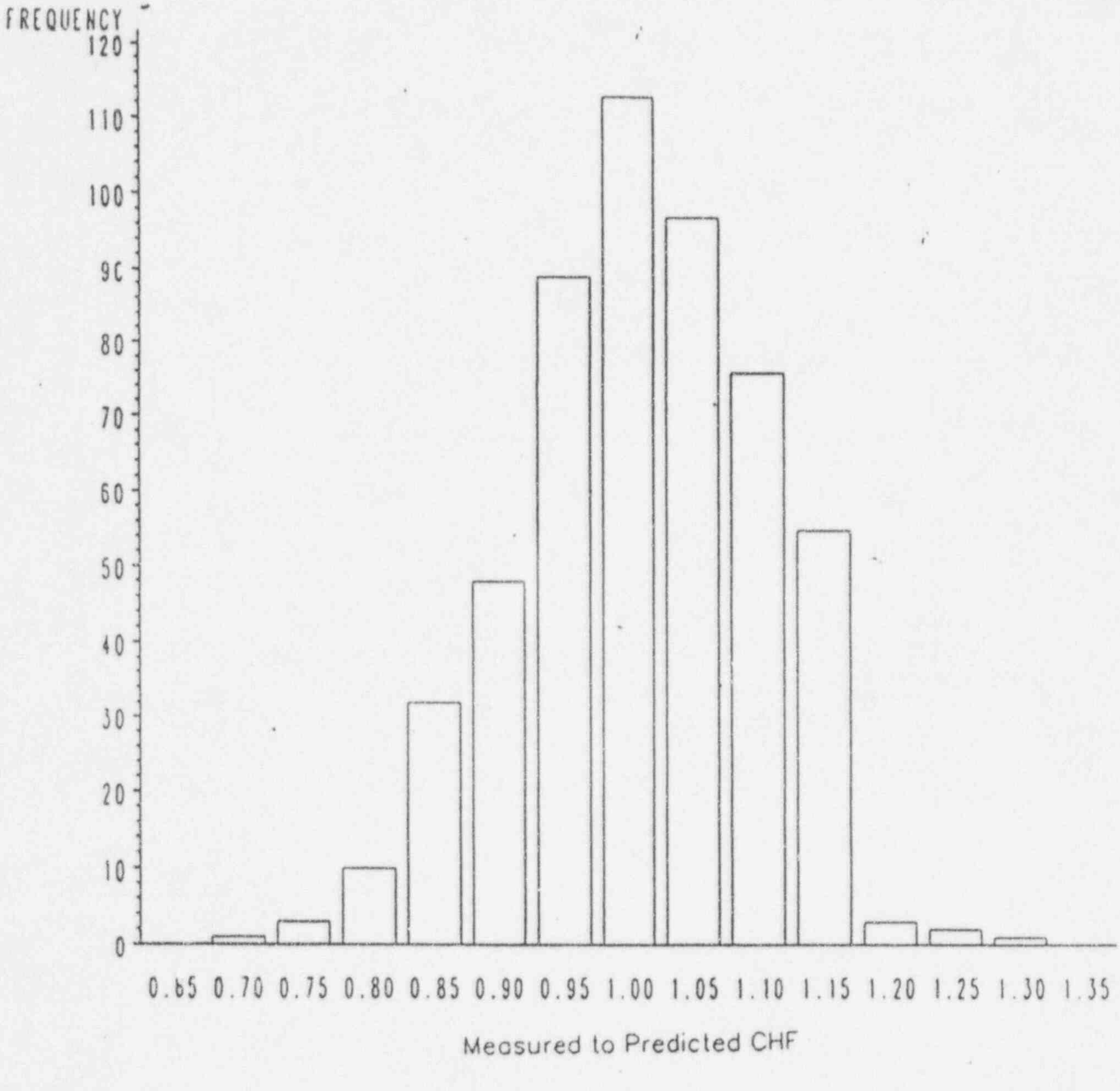

FIGURE C-3

Measured to Predicted CHF Versus Mass Velocity

Mark-BW Data Base

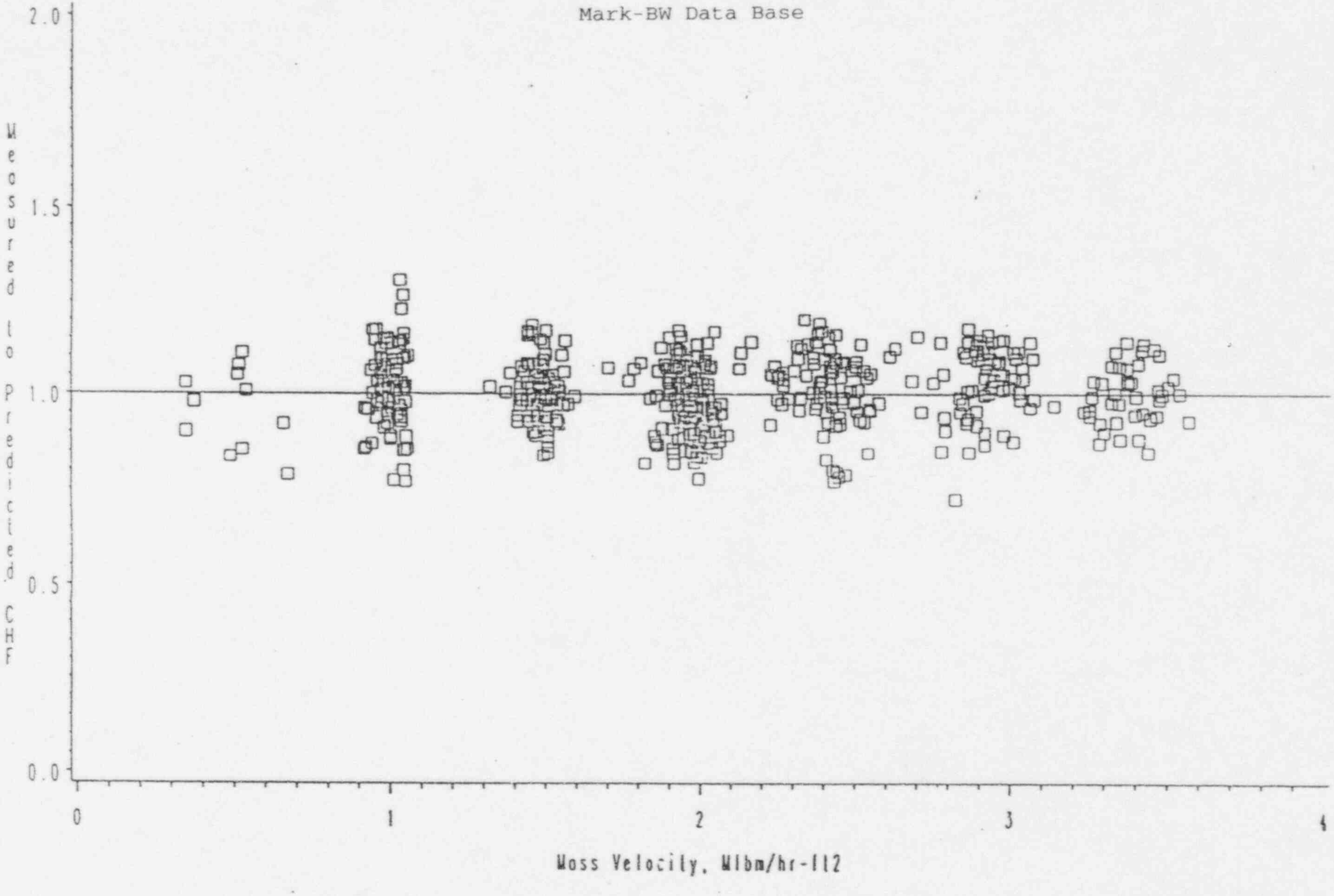

FIGURE C-4

Measured to Predicted CHF Versus Pressure

Mark-BW Data Base

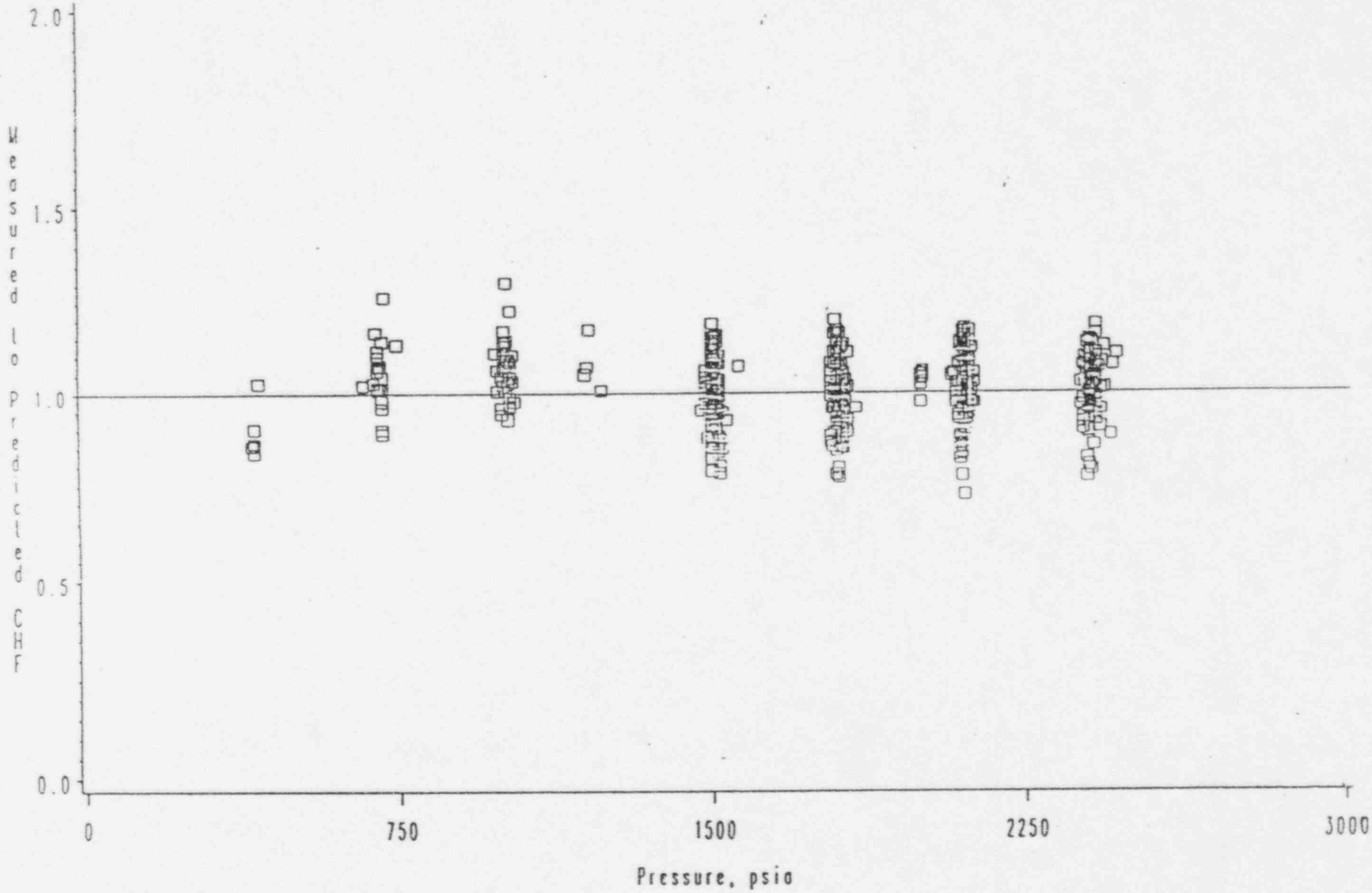

FIGURE C-5

Measured to Predicted CHF Versus Quality

Mark-BW Data Base

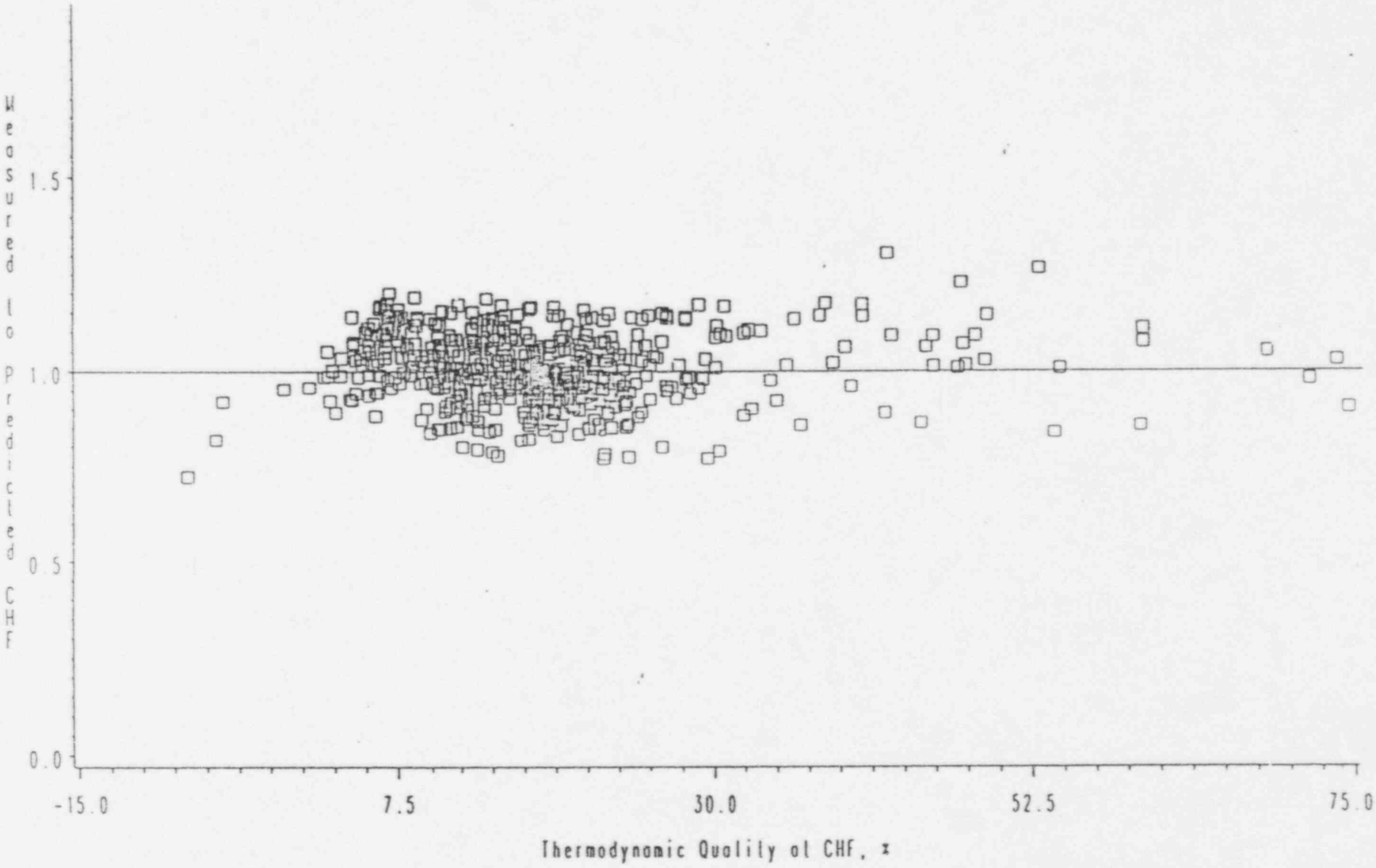

TABLE C-1 VIPRE-01 BWU-Z Correlation Verification

CHF Test Database Analysis Results

VIPRE-01 Statistical Results

Number Of Data Points	530
Average M/P	1.00850
Standard Deviation	0.09217
Upper D Prime	3469.0
Lower D Prime	3407.0
D Prime Value	3453.68

Accept Normality at 5% Level

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	Mark-BW 17x17
Design Limit DNBR, VIPRE-01	1.18

TABLE C-2 McGuire/Catawba SCD Statepoints

Stpt No.	Power* (% RTP)	RCS Flow (K gpm)	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F @ 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						

* 100% RTP = 3411 Megawatts Thermal

TABLE C-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	+/- 15.25% / 9.27%	Normal
Code/Model	[ ]	Normal

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

TABLE C-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.

TABLE C-3 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.
$F^E_{\Delta H}$	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 half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.

TABLE C-3 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
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 C-4 McGuire/Catawba Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation

500 Case Runs

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

BWU-Z Critical Heat Flux Correlation

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	DNBR
1				
7				
9				
12				

TABLE C-5 McGuire/Catawba Key Parameter Ranges

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

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

REFERENCES

C-1. DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, December 1991.

C-2. The BWU Critical Heat Flux Correlations, BAW-10199-P, Babcock and Wilcox, Lynchburg, Virginia, December 1994 (SER received April 5, 1996).

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