Non-Proprietary

DPC-NE-2005P

Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX D

Oconee Plant Specific Data

Mark-B11 Fuel

Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design

April 1997

This Appendix contains the plant specific data and limits for the Oconee Nuclear Station with Mark-B11 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 Oconee plant (two loop Babcock and Wilcox PWR's) as described in Reference D-1. The parameter uncertainties and statepoint ranges were selected to bound the unit and cycle specific values of the Oconee station. This analysis models the improved, small diameter, mixing vane grid, Mark-B fuel assembly denoted as the Mark-B11 design. Four lead test assemblies began operation in Oconee 2 in May of 1996. FCF is scheduled to issue a Mechanical Design Topical Report to the NRC in December 1997.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference D-3 and the Oconee eight and nine channel models approved in Reference D-1 are used in this analysis. Due to the fuel assembly design change, some specific data supplementary to Table 3-1 in Reference D-1 requires updating. This data is listed in Table D-1. Table D-1 includes fuel rod, control rod, and instrument guide tube diameters, the number of mixing and non-mixing vane grids, and the fuel rod length. The following section compares Mark-B design fuel assemblies with the Mk-B11 fuel assemblies.

Previous Mark-B design fuel assemblies consisted of 0.430 inch diameter fuel rods with 2 inconel and 6 intermediate non-mixing vane zircaloy grids. The Mark-B11 fuel assembly design is composed of fuel pins with a 0.416 inch outside diameter and two inconel grids and six intermediate zircaloy grids, one non-mixing grid and five mixing vane grids. The higher pressure drop and higher cladding surface heat flux of the Mark-B11 design is offset by the larger flow area and the presence of the mixing vane grids to result in improved assembly thermal performance.

The VIPRE-01 models approved in Reference D-1 are used to analyze the Mark-B11 fuel with the following exceptions:

1) The Mark-B11 fuel assembly geometry information is listed in Table D-1.

2) The turbulent mixing factor has been changed from 0.01 to 0.038 for the Mark-B11 fuel assembly design due to the presence of mixing vane grids. The numerical value was determined and provided by the

fuel supplier. This is consistent with FCF's 17x17 Mark-BW fuel assembly product and has been confirmed by Mark-B11 LDV test data.

3) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI. The Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 and is discontinuous at a quality equal to 1.0 (Reference D-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 D-3). This eliminates the discontinuity at a quality equal to 1.0. Therefore, the EPRI model provides a full range (i.e., void fraction range, 0 - 1.0) of applicability required for performing DNB calculations. Also, for overall model compatibility, the subcooled void model was changed from LEVY, as specified in Reference D-1, to the EPRI correlation for the Mark-B11 fuel.

To evaluate the impact of changing bulk void models on DNB prediction, forty-four Mark-B11 CHF test data points (Reference D-2) were compared using both the Levy/Zuber-Findlay and EPRI/EPRI subcooled void/bulk void combinations in VIPRE-01. These data points cover a pressure range of 1005 to 2425 psia and an inlet temperature range 361.3 to 604.3°F. The mass flux at the MDNBR location varied from 0.542 to 2.963 Mlbm/hr-ft^2. The void fraction at the MDNBR location varied from 0.106 to 0.711. The equilibrium quality at the MDNBR location varied from -0.104 to 0.198. The results of this comparison are as follows:

	Levy/Zuber-Findlay	EPRI/EPRI
Minimum DNBR (Avg)	0.991	0.996

The minimum DNBR results show a minimal difference of 0.54% (0.005 in DNB). Therefore, the EPRI bulk void model and EPRI subcooled void correlation will be used in Mark-B11 analysis.

Critical Heat Flux Correlation

The NRC approved BWU-Z form of the BWU critical heat flux correlation with the Mark B11V multiplier described in Reference D-2 is used for all Mark-B11 analyses. This correlation was developed by FCF for application to the Mark-B11 fuel design. The analysis in Reference D-2 was performed with the LYNXT thermal-hydraulic computer codes. This correlation was programmed into the VIPRE-01 thermal-hydraulic computer code by Duke Power Company and the Mark B11V data base analyzed in its entirety. The results of this analysis are shown in Table D-2. The resulting Average M/P value, data standard deviation, and CHF correlation limit are within 1% of the values reported in Reference D-2, page E-4 (also shown on Table D-2 under LYNXT column).

Figures D-1 through D-4 graphically show the results of this evaluation. Figure D-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the data base. Figures D-2 through D-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 D-2.

Based on the results shown in Table D-2 and Figures D-1 through D-4, the BWU-Z form of the BWU CHF application correlation with the Mark-B11V multiplier, licensed in Reference D-2, can be used in DNBR calculations with VIPRE-01 for Mark-B11 fuel.

Statistical Core Design Analysis

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table D-3. These statepoints represent the range of conditions to which the statistical DNB analyses limit will be applied. The range of key parameter values analyzed is listed on Table D-6.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table D-4. The uncertainties were selected to bound the values calculated for each parameter at Oconee. The uncertainties have not changed except for the rod power hot channel factor (Fq), core flow measurement, and DNBR correlation. The uncertainty for Fq has changed due to fuel design changes. The core flow measurement uncertainty was increased to ensure that it is bounding. This results in a more conservative SDL. The DNBR correlation uncertainty is the same as that stated in Reference D-5, page 4-3.

DNB Statistical Design Limits

The statistical DNBR limit for each statepoint evaluated is listed on Table D-5. Section 1 of Table D-5 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 this report (DPC-NE-2005) and Appendix C (Reference D-4). All of the DNBR distributions are normally distributed. The maximum statistical DNBR value in Table D-5 (full core of Mark-B11 fuel) for 5000 propagations is [ ]. Therefore, the statistical design limit, using the BWU-Z form of the BWU CHF correlation with the Mark-B11V multiplier for Mark-B11 fuel at Oconee, is [ ] for the range of parameters given in Table D-6.

Transition Cores

The transition core model determines the impact of the geometric and hydraulic differences between the resident Mark-B10 series fuel and the new Mark-B11 design. The 9 channel model described in Reference D-1 is used to evaluate the impact of transition cores containing Mark-B11 fuel. In Figure 4-5 in Reference D-1, Mark-B11 fuel is used instead of Mark-B6/7 and Mark-B10F/G fuel instead of Mark-B5. Therefore, channels 1 - 7 are modeled as Mark-B11 fuel, Channel 8 is modeled as Mark-B10F/G fuel, and Channel 9 is modeled as Mark-B11 fuel. The transition 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 out of the higher pressure drop mixing vane grid (Mark-B11) fuel is calculated.

A transition core penalty is evaluated by determining the DNBR impact on a Mark-B11 limiting assembly when analyzed with the 9 channel model. Once determined, several methods are available to conservatively compensate for the penalty. One method of compensating for the reduction in DNB performance due to the hydraulic effects of the conservatively modeled transition core is to explicitly apply a penalty to the Mark-B11 fuel generic peaking limits based on a full Mark-B11 core. Another option is to calculate maximum allowable peaking limits specifically modeling the transition core loading pattern in the detailed 64 channel model approved in Reference D-1. These methods will be used, as necessary, to determine the DNB effect of transition cores.

To evaluate the statistical DNB impact of the transition core, the most limiting statistical DNB statepoint (Statepoint 22 on Table D-5) was evaluated using the 9 channel model. This statepoint is designated TR22 in Table D-5. At 5000 cases, the statistical DNBR for statepoint TR22 is slightly greater than the limit for statepoint 22, but less than the statistical design limit, [ ]. Therefore, the statistical design limit, [ ], is bounding for Mark-B10/B11 transition cores; as well as, full Mark-B11 cores.

FIGURE D-1

Measured CHF Versus Predicted CHF

Mark-B11 Vane Data Base

FIGURE D-2

Measured to Predicted CHF Versus Mass Velocity

Mark-B11 Vane Data Base

FIGURE D-3

Measured to Predicted CHF Versus Pressure

Mark-B11 Vane Data Base

FIGURE D-4

Measured to Predicted CHF Versus Quality

Mark-B11 Vane Data Base

TABLE D-1 MARK-B11 FUEL ASSEMBLY DATA

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, in. (Nom.)	0.416
Thimble tube diameter, in. (Nom.)	0.530(1)/0.567(2)
Instrument guide tube diameter, in. (Nom.)	0.554
Fuel rod pitch, in (Nom.)	0.568
Fuel assembly pitch, in. (Nom.)	8.587
Fuel rod length, in. (Nom.)	154.16

(1) Above lowest mixing vane grid (MV) and between MV grids.

(2) Below the first mixing vane grid and above the top of the last mixing vane.

GENERAL FUEL CHARACTERISTICS

Grids:	Material	Quantity	Location	Type
	Inconel	2	Upper and Lower	Non-Mixing Vane
	Zircaloy	6	Intermediate	1 Non-Mixing Vane, 5 Mixing Vane

Fuel Rods:	Material	Quantity
	Zircaloy-4	208

Fuel Cycle Design Assembly Features

Fuel Assy.	Mark
Designation:	B11
Features:	Smaller clad outside diameter and mixing vane grids.

TABLE D-2 VIPRE-01 BWU-Z Correlation with Mark-B11V Multiplier Verification

CHF Test Database Analysis Results

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # Of data	216	216
N, degrees of freedom (n-1)	215	215
M/P, Average measured to predicted CHF	1.0084	1.0040
σ (M/P/N)	0.0859	0.0868
K(215,0.95,0.95),one sided tolerance factor Ref. D-2)	1.830	1.830
DNBR(L)= 1/(M/P - Kσ)= 1/[1.0040 - 1.830(0.0868)]	1.175	1.183

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-B11 15x15 Mixing Vane
Design Limit DNBR, VIPRE-01	1.19

TABLE D-3 Oconee SCD Statepoints

Statepoint Number	Power(1) (% RTP)	RCS Flow(2) % DF	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak $F\Delta H$
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						
26						
27						
TR22						

1) 100% RTP = 2568 Megawatts Thermal
2) 100% design flow is equal to 352,000gpm.

TABLE D-4 Oconee Statistically Treated Uncertainties

Parameter	Type	Type of Distribution	Uncertainty	Standard Deviation
Reactor System				
Core Power*	Measurement	Normal	+/-2.0%FP	+/-1.0%FP
Core Flow	Measurement	Normal	+/-4.2% design	+/-2.1% design
Pressure	Measurement	Normal	+/-30.0 psi	+/-15.0 psi
Temperature	Measurement	Normal	+/-2.0°F	+/-1.0°F
Nuclear				
FΔH	Calculation	Normal	---	+/-2.84%
Fz	Calculation	Normal	---	+/-2.91%
Z	Calculation	Uniform	+/-6 inches	---
Fq	Calculation	Normal	[	]
Hot Channel Flow Area	Measurement	Uniform	[ ]	---
DNBR	Correlation	Normal	---	9.268%
DNBR	Code	Normal	[	]

* Percentage of 100% RTP (69.75 MWth wherever applied).

TABLE D-4 Continued   Oconee Statistically Treated Uncertainties

Parameter	Justification
System Pressure	This uncertainty accounts for random uncertainties in various instrumentation components. Since the random uncertainties are normally distributed, the square root of the sum of the squares (SRSS) that results in the pressure uncertainty is also normally distributed.
Inlet Temperature	Same approach as Pressure uncertainty.
Core Power	The core power uncertainty was calculated by statistically combining various random uncertainties associated with the measurement of core power. Since the random uncertainties are normally distributed, the SRSS that results in the core power uncertainty is also normally distributed.
Core Flow	Same approach as Core Power uncertainty.
Radial Power, $F_{\Delta H}$	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly power and the measurement of the assembly power. This uncertainty distribution is normal.
Axial Peak Power, F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty is normally distributed.
Axial Peak Location, 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 D-4 Continued Oconee Statistically Treated Uncertainties

Parameter	Justification
Rod Power HCF, Fq	This uncertainty accounts for the increase in rod power due to manufacturing tolerances. The uncertainty in calculating the peak pin from assembly radial peak is also statistically combined with the manufacturing tolerance uncertainty to arrive at the correct value. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to assure the MDNBR channel location is consistent for all cases.
Hot Channel Flow Area	This uncertainty accounts for manufacturing variations in the instrument guide tube subchannel flow area. This uncertainty is uniformly distributed and is conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
DNBR - Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normally distributed.
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 normally distributed.

TABLE D-5 Oconee Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation With Performance Factor

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				
25				
26				
27				
TR22				

TABLE D-5 Continued   Oconee Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation With Performance Factor

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1-T				
3-T				
17-T				
21-T				
22-T				
24-T				
TR22-T				

TABLE D-6 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum
Core Power (%RTP)		
Pressure (psia)		
T inlet (deg F)		
RCS Flow (% Design)		
$F_{\Delta H}$, 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

D-1. DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, October 1989.

D-2. The BWU Critical Heat Flux Correlations Applications to the Mark-B11 and Mark-BW17 MSM Designs, Addendum 1 to BAW-10199P-A, Babcock and Wilcox, Lynchburg, Virginia, September 1996.

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

D-4 DPC-NE-2005P, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, Appendix C, November 1996.

D-5 The BWU Critical Heat Flux Correlations, Babcock and Wilcox, Lynchburg, Virginia BAW-10199P-A, December 1995.