Attachment 8

DPC-NE-2005 Appendix H Robinson Plant Specific Data (Redacted)

DPC-NE-2005

Duke Energy
Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX H

Robinson Plant Specific Data

Advanced W 15x15 HTP Fuel

Application of HTP CHF Correlation to the Advanced W 15x15 HTP Fuel Design

November 2014

Note: Bracketed text, tables, and figures are "D" (Duke) and/or "A" (AREVA NP) proprietary information.

This Appendix contains the plant specific data and limits for the Robinson Nuclear Plant with Advanced W 15x15 HTP fuel using the HTP critical heat flux correlation. The thermal hydraulic statistical core design analysis process was performed as described in the main body of this method report.

Plant Specific Data

This analysis is for the Robinson plant, a three loop Westinghouse PWR. This analysis models the 0.424 inch diameter Advanced W 15x15 HTP fuel assembly design. This assembly is a derivative of that described in the Reference H-2 AREVA Topical Report. The Advanced W HTP design incorporates the High Mechanical Performance or HMP bottom grid, M5® fuel rod cladding, and MONOBLOC™ guide tubes relative to the Reference H-2 design.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference H-3 is used in this analysis. A fourteen channel model was developed for the Advanced W 15x15 HTP fuel design comparable to the Oconee 15x15 Mark-B-HTP fuel model described in Reference H-1. Due to the fuel assembly design differences, some specific data supplementary to Reference H-1 required modification. The thermal-hydraulic characteristics of the Advanced W 15x15 HTP fuel are presented in Table H-1 and the model adjustments are

shown in Figure H-1. Table H-1 includes fuel rod, control rod, and instrument guide tube diameters, the number and design of the grids, and the fuel rod length.

The VIPRE-01 model approved in Reference H-1 is used to analyze the Advanced W 15x15 HTP fuel with the following modifications:

1) The Advanced W 15x15 HTP fuel assembly geometry information and model layout as described in Table H-1 and Figure H-1.

2) A radial power distribution based on the Robinson fuel geometry and current peaking limits as shown in Figure H-2.

3) The number of axial nodes in the model was increased due to the addition of Integral Flow Mixing (IFM) grids to the Advanced W 15x15 HTP fuel design at Robinson.

4) The EPRI bulk void fraction model is used instead of the Zuber-Findlay model. The Zuber-Findlay bulk void model is not applicable for void fractions above 85% (Reference H-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 H-3). This eliminates the discontinuity at higher void fractions. 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 EPRI subcooled void model is used. The EPRI void models are also used for the Mark-B11, Advanced Mark-BW, Mark-B-HTP, and RFA fuel analyses as approved in Appendix D, E, F, and G of DPC-NE-2005 respectively.

Critical Heat Flux Correlation

The NRC approved HTP critical heat flux correlation form in Reference H-4 is used for the Advanced W 15x15 HTP analyses in VIPRE-01. This correlation was developed by AREVA for application to the HTP fuel design. The original application was to the HTP fuel designs (Advanced W 15x15 HTP and Advanced W 17x17 HTP) with the XCOBRA-IIIC code, Reference H-4. The same database was subsequently analyzed in Reference H-5 with the LYNXT thermal-hydraulic computer code. [

] A

The HTP correlation form [ ] A, D was added to the VIPRE-01 thermal-hydraulic computer code by Duke Energy and the CHF test data base analyzed in its entirety. The results of this analysis are shown in Table H-2. The resulting VIPRE-01 average P/M is [ ], D the standard deviation [ ], D and the correlation DNBR limit is lower than the value of XCOBRA-IIIC (Reference H-4, shown on Table H-2 under XCOBRA column). Figures H-3 through H-6 graphically shows the results of this evaluation. Figure H-3 shows there is [ ]. D Figures H-4 through H-6 show there is [ ] D. These plots also include the additional [ ] A, D uncorrelated data used to expand the correlation range of applicability. Similar to the XCOBRA-IIIC (Reference H-4) and LYNXT (Reference H-5) results, the VIPRE-01 results for the [ ] A, D uncorrelated data show that the correlation conservatively predicts CHF for the extended range of applicability.

Based on the results shown in Table H-2 and Figures H-3 through H-6, the HTP CHF correlation can be used in DNBR calculations with VIPRE-01 for Advanced W 15x15 HTP fuel. Table H-3 shows the correlation allowable parameter range and correlation DNBR limit with VIPRE-01. Note that the higher correlation DNBR limit will be used for non-statistical analyses of the Advanced W 15x15 HTP fuel design with VIPRE-01.

Statistical Core Design Analysis

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table H-4. These statepoints represent the range of conditions to which the statistical DNB analyses limit will be applied. The resulting range of key parameter values analyzed is listed on Table H-7.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table H-5. The uncertainties were selected to bound the values calculated for each parameter at the Robinson plant with the Advanced W 15x15 HTP fuel design.

DNB Statistical Design Limits

The statistical DNBR limit for each statepoint evaluated is listed on Table H-6. Section 1 of Table H-6 contains the 500 case runs and Section 2 contains the 5,000 case runs. All of the DNBR distributions are judged to be normally distributed. The maximum statistical DNBR value in Table H-6 for 5,000 propagations is [ ] D. Therefore, the statistical design limit using the HTP CHF correlation in VIPRE-01 for Advanced W 15x15 HTP fuel at Robinson is conservatively selected to be 1.35.

REFERENCES

H-1. DPC-NE-3000-PA, Revision 5a, Thermal-Hydraulic Transient Analysis Methodology, October 2012

H-2. EMF-89-164(P)(A) Revision 0, Mechanical Licensing Report for H.B. Robinson High Thermal Performance Fuel Assemblies, Advanced Nuclear Fuels Corporation, November 1989

H-3. EPRI NP-2511-CCM-A, VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, Vol. 1-4, Revision 4.5, February 2014

H-4 EMF-92-153(P)(A), Revision 1, HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel, January 2005

H-5 BAW-10241(P)(A), Revision 1, BHTP DNB Correlation Applied with LYNXT, July 2005

FIGURE H-1

Advanced W 15x15 HTP VIPRE-01 14 Channel Model

D

x Fuel Rod

x Instrument Tube

y Flow Channel

y = Channel Index

x = Rod Index

FIGURE H-2

Advanced W 15x15 HTP VIPRE-01 14 Channel Model

Radial Power Distribution

D

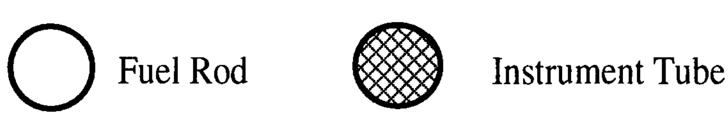

FIGURE H-3

VIPRE-01 Measured CHF versus Predicted CHF

HTP Database

D

FIGURE H-4

VIPRE-01 Measured to Predicted CHF versus Mass Flux

HTP Database

D

FIGURE H-5

VIPRE-01 Measured to Predicted CHF versus Pressure

HTP Database

D

FIGURE H-6

VIPRE-01 Measured to Predicted CHF versus Quality

HTP Database

D

TABLE H-1 Advanced W 15x15 HTP FUEL ASSEMBLY DATA

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, inches (Nom.)	0.424
Thimble tube diameter, inches (Nom.)	0.544
Instrument guide tube diameter, inches (Nom.)	0.544
Fuel rod pitch, inches (Nom.)	0.563
Fuel assembly pitch, inches (Nom.)	8.466
Active Fuel Length, inches (Nom)	144.0
Fuel rod length, inches (Nom.)	151.98

GENERAL FUEL CHARACTERISTICS

Component	Material	Quantity	Position	Type
Grid	Ni Alloy 718	1	Lower	HMP, Non-Mixing
	Zirc-4	6	Intermediate and Upper	HTP, Castellation
	Zirc-4	3	Intermediate	IFM, Castellation
Fuel Rod	M5®	204		
Guide Tube	Zirc-4	20		
Instrument Tube	Zirc-4	1		

HMP = High Mechanical Performance / Structural
HTP = High Thermal Performance / Structural
IFM = Intermediate Flow Mixer / Non-Structural

TABLE H-2

VIPRE-01 HTP Correlation Verification

VIPRE-01 / XCOBRA-IIIC Statistical Results

	VIPRE-01	XCOBRA
n, # of data	[ ]D	[ ]A
P/M, Average predicted to measured CHF*	[ ]D	[ ]A
σ (P/M)*	[ ]D	[ ]A
DNBR Correlation Limit	1.120	1.141

* - Statistics based on the correlated data

TABLE H-3

CHF Test Database Analysis Results

Parameter Ranges

Pressure, psia	1385 to 2425
Mass Velocity, M lbm/hr-ft^2	0.504 to 3.563
Inlet Enthalpy, Btu/lbm	383.6 to 646.1
Thermodynamic Quality at CHF	less than 0.514
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Advanced W 15x15 HTP
Correlation DNBR Limit	1.141

TABLE H-4

Robinson 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						
28						

D

1) 100% RTP = 2,339 Megawatts Thermal
2) 100% RCS Flow = 258,100 gpm

TABLE H-5 Robinson Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Distribution
Reactor Power*	+/- 0.3% / 0.18%	Normal
Coolant Flow		
Measurement	+/- 2.7% / 1.64%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 40 psia	Uniform
Temperature	+/- 4 degrees F	Uniform
Radial Peaking		
Measurement	+/- 4.0% / 2.43%	Normal
$F^{E}_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Axial Peaking		
F_Z	+/- 4.5% / 2.75%	Normal
Z	+/- 3 inches	Uniform
DNBR		
Correlation	[ ]$^{A, D}$	Normal
Code/Model	[ ]^{D}	Normal

* Percentage of 100% RTP (7.017 MWth) wherever applied

TABLE H-5 Robinson Statistically Treated Uncertainties Continued

Parameter	Justification
Reactor Power	The core power uncertainty was calculated by statistically combining various component uncertainties associated with the measurement of power such as sensor accuracy, drift, etc. Since the component uncertainties are random and are normally distributed, the combination of these uncertainties using the sum of the squares (SRSS) methodology results in a core power uncertainty that is also normally distributed.
Coolant Flow	
Measurement	Same approach as Core Power.
Bypass Flow	The core bypass flow is the parallel flow paths in the reactor vessel non-fuel regions and is calculated based on the driving pressure drop. The core bypass uncertainty is explicitly applied in the calculation of Core Inlet flow for each state point condition. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining various component uncertainties associated with the measurement of pressure such as sensor accuracy, drift, etc. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as Pressure.
Radial Peaking	
Measurement	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly and pin power as well as uncertainty from the measurement system components such as instrumentation drift or reproducibility, conversion of instrument readings to power, etc. This uncertainty is applied as a normal distribution.

TABLE H-5 Robinson Statistically Treated Uncertainties Continued

Parameter	**Justification**
$F^{E}_{\Delta H}$	This uncertainty accounts for the effect on peaking due to manufacturing variations in the variables affecting the heat generation rate along the flow channel and for the effect on peaking due to reduced hot channel flow area. The uncertainty is determined by statistically combining all the manufacturing tolerances. The uncertainty is normally distributed and is conservatively applied as one-sided to assure the MDNBR channel location is consistent for all cases.
Axial Peaking	
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty is normally distributed.
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.
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 various model sizes. The uncertainty distribution is normally distributed.

TABLE H-6 Robinson Statepoint Statistical Results

SECTION 1

Advanced W 15x15 HTP Fuel, HTP 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				
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21				
22				
23				
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25				
26				
27				
28				

D

TABLE H-6 Robinson Statepoint Statistical Results

Continued

SECTION 2

Advanced W 15x15 HTP Fuel, HTP Critical Heat Flux Correlation

5,000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
5				
7				
9				
21				
23				
27				

D

TABLE H-7 Robinson Key Parameter Ranges

Parameter	Maximum	Minimum
Core Power (%RTP)		
Coolant Flow (% Design)		
Core Exit Pressure (psia)		
Core Inlet Temperature (° F)		
FΔH, Fz, Z		

D

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