Attachment 9

DPC-NE-2005 Appendix I Harris Plant Specific Data (Redacted)

DPC-NE-2005

Duke Energy
Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX I

Harris Plant Specific Data

Advanced W 17x17 HTP Fuel

Application of HTP CHF Correlation to the Advanced W 17x17 HTP Fuel Design

November 2014

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

This Appendix contains plant specific data and limits for the Harris Nuclear Plant with Advanced W 17x17 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

The Harris Nuclear Plant is a three loop Westinghouse PWR. This analysis models the 0.376 inches fuel rod outer diameter Advanced W 17x17 HTP fuel assembly design. This assembly is a derivative of the fuel assembly described in AREVA Topical Report shown in Reference I-2. The Advanced W 17x17HTP design incorporates the High Mechanical Performance or HMP bottom grid, M5® fuel rod cladding, and MONOBLOC™ guide tubes relative to the Reference I-2 design.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference I-3 is used in this analysis. A fourteen channel model, based on the Oconee Nuclear Station 15x15 Mark-B-HTP fuel design in Reference I-1, was developed for the Harris Nuclear Plant Advanced W 17x17 HTP fuel design. Due to the fuel assembly design differences, some specific data supplementary to Reference I-1 are updated. This data is listed in Table I-1 and the model adjustments are shown in Figure I-1. Table I-1 includes fuel rod, control rod, and instrument guide tube outer diameters, the number and design of the grids, and the fuel rod length.

The Oconee Nuclear Station 15x15 Mark-B-HTP fuel design fourteen channel VIPRE-01 model approved in Reference I-1 is used to analyze the Harris Nuclear Plant Advanced W 17x17 HTP fuel with the following modifications:

1) The Advanced W 17x17 HTP fuel assembly geometry information and model layout as described in Table I-1 and Figure I-1.

2) A modified radial power distribution based on the Harris Nuclear Plant Advanced W 17x17 HTP fuel assembly geometry and current peaking limits as shown in Figure I-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 17x17 HTP fuel design at Harris Nuclear Plant.

4) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI model. The Zuber-Findlay bulk void model is not applicable for void fractions above 85% (Reference I-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 I-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 subcooled void model was changed from the LEVY model, as specified in Reference I-1, to the EPRI model. This same modeling change was implemented for the Mark-B11, Advanced Mark-BW, Mark-B-HTP, and RFA fuel products and 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 I-4 is used for the Advanced W 17x17 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 I-4. The same database was subsequently analyzed in Reference I-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 I-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 I-4, shown on Table I-2 under XCOBRA column). Figures I-3 through I-6 graphically shows the results of this evaluation. Figure I-3 shows there is [

][D] Figures I-4 through I-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 I-4) and LYNXT (Reference I-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 I-2 and Figures I-3 through I-6, the HTP CHF correlation can be used in DNBR calculations with VIPRE-01 for Advanced W 17x17 HTP fuel. Table I-3 shows the correlation allowable parameter range and design limit with VIPRE-01. Note that the higher correlation limit will be used for VIPRE-01 analyses of the Advanced W 17x17 HTP fuel design.

Statistical Core Design Analysis

Statepoints

The state point conditions evaluated in this analysis are listed in Table I-4. 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 I-7.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table I-5. The uncertainties were selected to bound the values calculated for each parameter at the Harris Nuclear Plant with the Advanced W 17x17 HTP fuel design.

DNB Statistical Design Limits

The statistical DNBR limit for each statepoint evaluated is listed on Table I-6. Section 1 of Table I-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 I-6 for 5,000 case runs is [ ] D. Therefore, the statistical design limit, using the HTP CHF correlation in VIPRE-01 for Advanced W 17x17 HTP fuel at Harris, is conservatively seleted to be 1.34.

References

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

I-2. EMF-93-074(P)(A) and Supplement 1, Generic Mechanical Licensing Report for Advanced 17x17 Fuel Design, Siemens Power Corporation, April 1994.

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

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

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

FIGURE I-1

ADVANCED W 17X17 HTP FUEL DESIGN

VIPRE-01 14-CHANNEL MODEL

[ ] D

xx Fuel Rod

xx Instrument Guide Tube

xx Control Rod Guide Tube

xx = Rod Index

yy Flow Channel

yy = Channel Index

FIGURE I-2

ADVANCED W 17X17 HTP FUEL DESIGN

VIPRE-01 14-CHANNEL MODEL RADIAL POWER DISTRIBUTION

[ ]D

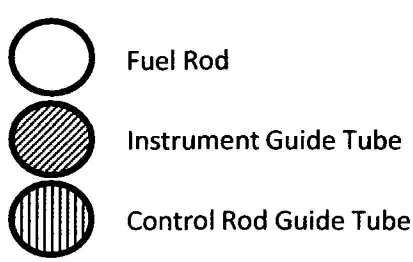

FIGURE I-3

VIPRE-01 MEASURED CHF VERSUS PREDICTED CHF

HTP DATABASE

D

FIGURE I-4

VIPRE-01 MEASURED TO PREDICTED CHF VERSUS MASS FLUX

HTP DATABASE

[ ] D

FIGURE I-5

VIPRE-01 MEASURED TO PREDICTED CHF versus PRESSURE

HTP DATABASE

[ ] D

FIGURE I-6

VIPRE-01 MEASURED TO PREDICTED CHF VERSUS QUALITY HTP DATABASE

D

TABLE I-1
ADVANCED W 17x17 HTP FUEL ASSEMBLY DATA
(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod outer diameter, inches (nominal):	0.376
Thimble tube diameter, inches (nominal):	0.482
Instrument guide tube diameter, inches (nominal):	0.482
Fuel rod pitch, in (nominal):	0.496
Fuel assembly pitch, inches (nominal):	8.466
Active Fuel Length, inches (nominal):	144.0
Fuel rod length, inches (nominal):	151.51

GENERAL FUEL CHARACTERISTICS

Component	Material	Quantity	Position	Type
Grid	Nickel Alloy-718	1	Lower	HMP, Non-Mixing
	Zirc-4	7	Intermediate/Upper	HTP, Castellation
	Zirc-4	3	Intermediate	IFM, Castellation
Fuel Rod	M5®	264		
CRGT	Zirc-4	24		
IGT	Zirc-4	1		

CRGT = *Control Rod Guide Tube*
IGT = *Instrument Guide Tube*
HMP = *High Mechanical Performance / Structural*
HTP = *High Thermal Performance / Structural*
IFM = *Intermediate Flow Mixer / Non Structural*

TABLE I-2

VIPRE-01 HTP CORRELATION VERIFICATION

VIPRE-01 / XCOBRA-IIIC STATISTICAL RESULTS

	VIPRE-01	XCOBRA-IIIC
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 I-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 17x17 HTP*
DNBR Correlation Limit	*1.141*

TABLE I-4

HARRIS NUCLEAR PLANT SCD STATEPOINTS

D

State Point	Core Power[1]	Coolant Flow[2]	Core Exit Pressure	Core Inlet Temperature	Axial Peak Magnitude	Axial Peak Location	Radial Peak
	% RTP	% DF	psia	°F	F_z	Z	F_{DH}
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							

1) 100% RTP = 2,948 MWt
2) 100% RCS Flow = 293,540 gpm

TABLE I-5

HARRIS STATISTICALLY TREATED UNCERTAINTIES

PARAMETER	UNCERTAINTY / STANDARD DEVIATION	DISTRIBUTION
Core Power:*	± 0.34% / 0.21%	Normal
Coolant Flow		
Measurement:	± 2.2% / 1.34%	Normal
Bypass Flow:	± 1.5%	Uniform
Core Exit Pressure:	± 35.0 psia	Uniform
Core Inlet Temperature:	± 3.5 degrees F	Uniform
Radial Power Distribution		
$F^N_{\Delta H}$ (measurement):	± 4.0% / 2.43%	Normal
$F^E_{\Delta H}$ (engineering):	± 3.0% / 1.82%	Normal
Axial Power Distribution		
F_Z:	± 4.5% / 2.75%	Normal
Z:	± 3 inches	Uniform
DNBR		
Correlation:	[ ] A,D	Normal
Code/Model:	[ ] D	Normal

* Percentage of 100% RTP (10.023 MWth) wherever applied.

TABLE I-5 (continued)

HARRIS STATISTICALLY TREATED UNCERTAINTIES

PARAMETER	JUSTIFICATION
Core Power	The core power uncertainty is calculated by combining various component uncertainties associated with the measurement of core power. 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 uncertainty.
Bypass Flow:	The core bypass flow is the parallel core flow paths in the reactor vessel non-fuel regions and is dependent on the driving pressure drop. The bypass flow uncertainty is explicitly applied in the calculation of core inlet flow rate for each state point condition. This uncertainty was conservatively applied with a uniform distribution
Core Pressure	The reactor coolant pressure uncertainty is calculated by statistically combining various component uncertainties associated with the measurement of pressure. This uncertainty is conservatively applied as a uniform distribution.
Core Temperature	Same approach as Pressure uncertainty.
Radial Power Distribution	
$F^N_{\Delta H}$ (measurement):	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly and pin power, and the measurement of the assembly power. This uncertainty is applied as a normal distribution.
$F^E_{\Delta H}$ (engineering):	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.

TABLE I-5 (continued)

HARRIS STATISTICALLY TREATED UNCERTAINTIES

PARAMETER	JUSTIFICATION
Axial Power Distribution	
F_Z:	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. This uncertainty is applied as a normal distribution.
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 length. The uncertainty distribution is conservatively applied as uniform.
DNBR	
Correlation:	This uncertainty accounts for the CHF correlation's ability to predict DNB. This uncertainty is applied as a normal distribution.
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. This uncertainty is applied as a normal distribution.

TABLE I-6

HARRIS STATEPOINT STATISTICAL RESULTS

SECTION 1

ADVANCED W 17X17 HTP FUEL
HTP CRITICAL HEAT FLUX CORRELATION
(500 CASE RUNS)

State Point	Mean	Standard Deviation	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				
28				

D

TABLE I-6 (continued)

HARRIS STATEPOINT STATISTICAL RESULTS

SECTION 2

ADVANCED W 17X17 HTP FUEL
HTP CRITICAL HEAT FLUX CORRELATION
(5000 CASE RUNS)

State Point	Mean	Standard Deviation	Coefficient of Variation	Statistical DNBR
5	[			] D
7				
9				
25				
27				

TABLE I-7

HARRIS KEY PARAMETER RANGES

Key Parameter	Maximum	Minimum
	[	] D
Core Power (% RTP)		
Coolant Flow (% Design)		
Core Exit Pressure (psia)		
Core Inlet Temperature (°F)		
$F_{\Delta H}$, F_z, Z		

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