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September 21, 1998

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001
Attention: Document Control Desk

Subject: Duke Energy Corporation

Oconee Nuclear Station, Units 1, 2 and 3
Docket Numbers 50-269, 50-270, and 50-287

Response to NRC Request for Additional Information on Appendix D to Topical Report DPC-NE-2005-P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology"

This submittal contains information that Duke Energy Corporation considers PROPRIETARY and is being made pursuant to 10CFR 2.790.

By letter dated July 1, 1998 the NRC requested additional information on Appendix D to Topical Report DPC-NE-2005P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology." This topical report had been previously submitted for NRC review by Duke letter dated April 22, 1997.

The questions contained in the July 1 NRC letter, and the corresponding Duke answers, are provided in the attachment to this letter. Additionally, Table D-1, which is also included in the attachment, has been revised to correct a typographical error.

Some of the information contained in the attachment is considered proprietary. In accordance with 10CFR 2.790, Duke Energy Corporation requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of the affected information is included with this letter. A non-proprietary version of the affected material is also included.

A001

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc:

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AFFIDAVIT OF M. S. TUCKMAN

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

M. S. Tuckman

M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in the attachment to Duke Energy Corporation letter dated September 21, 1998; SUBJECT: Response to NRC Request for Additional Information on Topical Report DPC-NE-2005P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methology." This information enables Duke to:

(a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

(b) Support Facility Operating Licenses/Technical Specifications amendment requests for Babcock & Wilcox PWRs.

(c) Perform safety evaluations per 10CFR50.59.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

M. S. Tuckman

M. S. Tuckman

(Continued)

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 22nd day of

September, 1998

Linda Case Smith

~~Mary P. Nelms,~~ Notary Public
Linda Case Smith

My Commission Expires: ~~January 22, 2001~~ May 6, 2000

SEAL

NRC Questions On Mark-B11 SCD Submittal

Questions shown in italics, answers immediately follow.

1. *The safety evaluation report (SER) for DPC-NE-2005P-A requires that in all applications of the statistical core design methodology, the uncertainties and distributions used in the analysis will be justified on a plant-specific basis. This has not been done in Appendix D, which presents plant-specific data for Oconee with Mark-B11 fuel. Comparing Table D-4 of Appendix D with Table A-2 of Appendix A (which contains the approved uncertainties and distributions for the Oconee units with Mark-B10 fuel), there are four major changes, none of which are explained adequately. Specifically:*

   a) *The core flow uncertainty has been increased from +/- 2.0 percent design (with standard deviation of +/-1.0 percent design) to +/-4.2 percent design (with standard deviation +/-2.1 percent design). The report says simply that the value was increased "to ensure that it is bounding." Bounding in what way? How was this determined? It appears to be an arbitrary adjustment to what should be a real indicator of the uncertainty in the measured flow. What is the justification for this change?*

The current Chapter 15 analyses for Oconee were performed by FCF. Duke Power has recently reanalyzed the Chapter 15 transients and submitted a topical report that is currrently being reviewed (DPC-NE-3005). As part of this effort, Duke recalculated the flow uncertainties for combinations of 4, 3, and 2 operating reactor coolant pumps. Table D-4 has been revised to show the flow uncertainties used in the BWU-Z SCD analyses (the original table only listed the maximum flow uncertainty for 2 pump operation). The statepoints listed in Table D-3 were propagated using the appropriate flow uncertainty. For example, statepoint 22 is the limiting statepoint for the 2 pump coastdown transient. Thus, this statepoint was propagated using a flow uncertainty of 4.2 % (std. deviation of 2.1 %).

Statepoints using the flow uncertainties for 3 and 2 operating reactor coolant pumps have also been propagated using the BWC correlation. The statistical DNB limit for all cases using the higher flow uncertainties was less than the SCD limit given in Appendix A. Thus, no NRC submittal was required based on the criteria given in Table 7 of DPC-NE-2005.

b) Table A-2 includes the parameter Fq" (local heat flux hot channel flow (HCF)), which is an uncertainty to account for the decrease in departure from nuclear boiling ratio (DNBR) at the point of minimum DNBR due to engineering tolerances. It also accounts for flux depression at a spacer grid, and has a value of [ ] (with standard deviation [ ]). This parameter has been omitted in Table D-4. What is the justification for this change?

The local heat flux hot channel factor accounts for the effects on DNBR of local variations in pellet enrichment and weight on local (hot spot) power, and flux depressions at spacer grid. Small local heat flux spikes have been shown to have no effect on the critical heat flux (CHF) per the Oconee 1 Cycle 14 Reload Report, DPC-RD-2018.

DPC-RD-2018 was submitted as supplementary information in support of Technical Specification changes for Oconee Unit 1 (Amendment 191, TAC 80378), Unit 2 (Amendment 191, TAC 80379), and Unit 3 (Amendment 188, TAC 80380). NRC approved the Technical Specification change. Duke considered the NRC's implementation of the recommended Technical Specification changes to be implicit approval of DPC-RD-2018.

Removal of Fq" from DNBR analyses was justified in DPC-RD-2018 based on WCAP-8202 and CENPD-207. The WCAP-8202 evaluation concluded that the data and analysis clearly indicate no effect on the minimum DNBR due to large local heat flux spikes. The spikes tested were in the region of MDNBR and were 20% greater than the heat flux in the immediate vicinity. The conclusion of these reports is that the local heat flux spikes associated with fuel densification have no affect on DNBR. This effect is generic to PWR fuel types and was confirmed to be applicable by the fuel vendor. Additionally, the magnitude of Fq" calculated by the vendor for Mark-B11 fuel is much smaller, [ ]. Based on this information and the approved reload report submittal, the Fq" factor was omitted from the Mark-B11 analyses and, therefore, Table D-4.

c) Table A-2 includes the parameter F_q (rod power HCF), which is an uncertainty to account for rod power increases due to manufacturing tolerances. This parameter also includes the uncertainty in calculating the pin peak from the assembly radial peak, and has a value of [ ] (with standard deviation [ ]) for Mark-B10 fuel. In Table D-4, this parameter has the value of [ ] (with standard deviation of [ ]) for Mark-B11 fuel. How was this uncertainty determined, and why is it larger for Mark-B11 fuel than for Mark-B10 fuel?

A rod power hot channel factor of [ ] was specified in the original issue of DPC-NE-2005 for Mark-B10 fuel. This value has increased to [ ] for the Mark-B10 fuel (beginning with Oconee 1, 2, and 3 Batch 17) and [ ] for the Mark-B11 fuel to account for dry blending of UO_2 powder to achieve the desired enrichment. The Statistical Design Limit (SDL) given in DPC-NE-2005 was shown to still be valid for Mark-B10 fuel using the increased rod power hot channel factor evaluated as per the process described in Table 7 of DPC-NE-2005P-A.

The value for F_q used in the SCD analysis is calculated as follows based on the rod power hot channel factor of [ ] for Mark-B11 provided by the fuel manufacturer and the radial peak uncertainty of [ ] per DPC-NE-1004P-A.

$$F_q\% = [\quad]$$

The standard deviation is calculated as follows:

$$\sigma(F_q)\% = \frac{[\quad]}{1.645}$$

$$= [\quad]$$

The 1.645 is the one-sided 95/95 statistical K factor for an infinite number of points.

The rod power hot channel factor, Fq, is provided as part of the fuel fabrication process and is formally transmitted in batch specific design and fabrication data supplied by the fuel manufacturer for each reload batch. If the rod power hot channel factor were greater than [ ], then the fuel manufacturer would notify Duke Power and the impact on the SDL will be evaluated.

d) The HCF area uncertainty is reported as [ ] in Table D-4, unchanged from the value in Table A-2, for Mark-B10 fuel, even though there are significant differences in the assembly geometry of Mark-B11 fuel. In addition, the value reported for the parameter Fq indicates that there are significant differences in the manufacturing tolerances for the fuel rods in Mark-B11 fuel, which would seem to imply that there should also be significant differences in the flow channel geometry variations. What is the justification for using the value of [ ] for this parameter?

The value listed in Table D-4 was provided by the fuel manufacturer. The fuel fabricator will verify through inspection of the final fuel assemblies and components that the uncertainty on flow area assumed in the analysis is valid. The same inspection techniques employed in earlier designs will be used for the Mark-B11 fuel. Comparison to acceptance criteria for the Mark-B11 fuel will ensure compliance with the [ ] flow area uncertainty. Water channel data taken on the Mark-B11 lead test assemblies were evaluated and found to be acceptable. If the flow area uncertainty is greater than [ ], then the fuel manufacturer will notify Duke Power and the impact on the SDL will be evaluated as per the process described in Table 7 of DPC-NE-2005P-A.

2. *The description of how transition cores will be treated is unclear. Please provide additional information, addressing the following points:*

   a) What is a "transition core penalty," and how is it determined?

A generic transition core penalty is determined by comparing the DNBR results from a full core of Mark-B11 fuel with the DNBR results from a conservative Mark-B11/Mark-B10 transition core. The 9 channel transition core model licensed in DPC-NE-2003 and described in Appendix D of DPC-NE-2005 is used in this analysis. The MDNBR (or allowable radial peaking) is calculated with both models for a range of fluid conditions and axial peaking combinations. The largest penalty calculated from this matrix of conditions is used as the transition core penalty.

The process for determining a generic transition core penalty is as follows:

1. Develop an 8 channel Mark-B11 full core model and a 9 channel Mark-B11/Mark-B10 transition core model (per DPC-NE-2003P-A and described in Appendix D of DPC-NE-2005).

2. Evaluate each model for a range of fluid conditions (shown below) that is representative of fluid conditions for which the Maximum Allowable Peaking limits are developed using VIPRE-01. These fluid conditions are evaluated at the axial peaking conditions shown below.

Parameter	**Maximum**	**Minimum**
Core Power (% RTP)	110	80
RCS Flow (% Design Flow)	107.5	80.3
T inlet (deg F)	572.8	529.2
Pressure (psia)	2242	1830
Fz (normalized axial peak)	1.1, 1.4, 1.7, 2.1	
z (location of axial peak)	0.2, 0.4, 0.6, 0.8	

   The radial peaking results from VIPRE-01, for both the Mark-B11 full and transition core models at each fluid condition, were compared to determine the limiting fluid condition. Then a complete set of MAP curves were developed for both the full core Mark-B11 and the transition core models. These peaking results were compared, and a maximum transition core peaking penalty was determined. In addition, the axial dependency of the transition core penalty was determined. The response to question 2c) specifies the options for the application of the transition core penalty.

b) *The local pressure drop differences between the Mark-B10 and Mark-B11 fuel assemblies mean that the local assembly flow distributions may be very different in a mixed core, due to differences in inter-assembly crossflow patterns. The departure from nucleate boiling (DNB) behavior of a mixed core may, therefore, be significantly different from that of a full core of Mark-B11 fuel only. Justify the assumption that the BWU-Z CHF correlation can be applied to Mark-B11 fuel in a core containing both Mark-B10 and Mark-B11 fuel*

In mixed cores, the possibility of large axial velocity upsets at or around dissimilar grids exists. These upsets imply different local thermal-hydraulic conditions in surrounding subchannels. It has been questioned as to whether traditional steady state CHF correlations are applicable in this instance.

The FCF CHF correlation form (BWU) is composed of three parts: a uniform part dependent solely on the local thermal-hydraulic conditions of pressure, mass velocity and thermodynamic quality at the axial location of CHF, a non-uniform F factor modification dependent on the shape of the axial heat flux input, and a multiplicative geometric factor dependent on the overall fuel assembly grid spacing and heated length. It is with the uniform, local conditions part that the mixed core conditions question surfaces.

CHF correlations are developed from data from full length electrically heated bundles in 5-by-5 rod arrays. For each data point, the inlet conditions of coolant mass velocity, pressure and temperature are known, as is the power (heat flux) required to produce a DNB event. The local thermal-hydraulic conditions at the axial location of CHF must then be calculated with a computer code.

The proof of applicability of a CHF correlation, then, is how well it can predict the critical heat flux that was measured in the DNB event using the calculated local conditions. Thus, the applicability of a CHF correlation is dependent not only on its form and data base, but on the accuracy with which the local conditions can be calculated in any given situation. Because of the size of the test section (a 5-by-5 rod array) and the use of a series of single spacer grids (axially), normal CHF tests do not exhibit large hydraulic axial differences. FCF, however, has performed one test with widely varying subchannel axial resistances producing the large velocity upsets representative of mixed core conditions. This test was a 5-by-5 test of the Mark B zircaloy grid modeled as the corner intersection of four fuel assemblies. LDV testing of the intersection grid showed velocity depressions as large as 50% between the intersection subchannel and the surrounding unit cell subchannels. This test was conducted at the Babcock & Wilcox Alliance Research Center and is documented in BAW-10143P-A (BWC correlation of Critical Heat Flux, April, 1985). In the topical, the measured to predicted (M/P) CHF results were compared for two traditional test bundles and the intersection bundle. The guide tube bundle (B15) had an average M/P of 0.971, the unit cell bundle (B16) 0.985 and the intersection bundle (B17) 0.976. The difference in M/P results is statistically

insignificant. This qualified the BWC correlation for use with the Mark B fuel assembly design.

The local conditions necessary for the BWC correlation were calculated with a thermal-hydraulic computer code. The local conditions for the normal unit and guide tube bundles had very little axial upset, while the intersection bundle (which produces conditions representative of a mixed core) had severe upsets resulting from the two to one velocity upsets. The fact that the BWC correlation performed consistently on conditions representative of both homogeneous and mixed cores confirms that the FCF local conditions CHF correlations are valid for both homogeneous and mixed core applications as long as the local conditions can be accurately predicted by the subchannel thermal-hydraulic computer code.

In this particular application, the velocity upsets calculated in the Mark-B11 transition core analysis are on the order of 10%. These calculations assume the limiting geometry (a single Mark-B11 assembly surrounded by Mark-B10 fuel). Since the test data included local depressions as large as 50%, the FCF test results bound by a significant amount the transition core configuration.

c) Two options are described (see p. D-7) that will be used to "conservatively compensate" for the transition core penalty. The report states that they will be applied "as necessary" to determine the DNB effect of a transition core. What are the criteria for selecting one or the other of the two options? How will it be determined that the selected option is "conservative" for a given transition core?

The three methods for penalizing a transition core are to

1) Penalize the DNBR limit used in the analyses directly or
2) Penalize the Maximum Allowable Peaking Total (MATP) limits determined for a transition core
3) Use a combination of the two above.

The penalty applied using either method 1 or 2 is based on the most limiting transition core statepoint determined as described in the response to Question 2(a) above. This ensures either option is conservative for the transition core.

Option 3) listed above is the current one selected to provide a bounding, conservative transition core penalty while maximizing core design flexibility. As described in Question 2(a), the transition core penalty was evaluated with a subset of axial peak locations (F_Z, Z) over a wide range of fluid conditions. Then, the fluid condition with the largest penalty was evaluated with a complete set of axial peaks (F_Z from 1.1 to 2.1, Z from 0.01 to 1.0). This is the same set of axial peak locations used to generate the MATP limits and resulting curves described in DPC-NE-2003.

In this analysis, the transition core penalty shows axial shape dependence. Due to this relationship, it is reasonable to include part of the penalty directly in the applicable MATP limits. Based on the analysis described above, the transition core penalty is applied as follows:

1. A 0.5% peaking penalty (1.5% DNB penalty) is applied to the retained DNB margin available between the SDL and the DDL for Mark-B11 transition cores. This directly applies the penalty to all DNB calculations.

2. A 1% radial peaking penalty is applied to selected axial peak locations. These are generally the large axial peaks (F_Z of ≥ 1.4) in the top half of the core (at $Z \geq 0.6$). Again, these axial peak locations were determined by comparison of a complete set of MATP curves. This penalty will be applied to all Mark-B11 MATP's limits at the axial peaking locations necessary.

Duke will retain the option of applying any of the three methods described as a conservative transition core penalty such that cycle design impact is minimized.

d) One option of the two described on p. D-7 is to explicitly apply a penalty to the Mark-B11 fuel generic peaking limit based on a full Mark-B11 core. What is the penalty? How is it determined? How will it be determined that the penalty adequately accounts for the effects of a mixed core on DNB behavior?

The transition core penalty is determined as described in the response to Question 2(a) above. This adequately accounts for the mixed core effect as explained in the response to Question 2(b). As stated in the response to Question 2(c), the transition core penalty can be applied to the maximum allowable peaking (MAP) limits calculated for a full Mark-B11 core. This reduces the allowable peaking in the transition core to account for the hydraulic and geometry effects. This also ensures that the MDNBR in all transition core analyses is greater than the licensed SDL.

As with previously licensed transition core methods, the transition core geometry for a reload cycle can be specifically modeled using the 64 channel model described in DPC-NE-2003. This larger model allows analyses of the actual cycle loading pattern to determine the impact of a mixed core on the maximum allowable peaking limits for the transition cycle.

3. *Table D-2 (p. D-14) claims a pressure range of 400 to 2465 psia for the BWU-Z correlation with the Mark-B11V multiplier. The database supporting this form of the correlation includes tests only over the pressure range 695 to 2425 psia. In addition, there is a distinct nonconservative bias evident in the correlation's predictions with decreasing pressure (See Figure D-3, p. D-11). The BWU-Z correlation for Mark-BW17 fuel (as documented in BAW-10199-A) has demonstrated bias with decreasing pressure, and the SER for this correlation specifies a separate design limit DNBR of 1.59 for pressures below 700 psia. If the BWU-Z correlation with the Mark-B11V multiplier is to be applied to conditions where the pressure is below 700 psia, what value will be used for the design limit DNBR and how will it be determined?*

See Question 5 for response.

4. *The SER for DPC-NE-2005P-A requires that the selected state points for an application of the SCD methodology shall be justified to be appropriate, on a plant-specific basis. Documentation of this justification in Appendix D consists only of the statement on p. D-1 "...state point ranges were selected to bound the unit and cycle-specific values of the Oconee Station." However, the document also notes that the values of key parameter ranges used to define the state points (Table D-6, p. D-21) are "based on the currently analyzed state points," and further notes that "ranges are subject to change based on future state point conditions." The procedure and justification for selecting state points is unclear, and additional information is needed. Specifically, please provide a more detailed description of how the state points are selected for the Oconee plant-specific data, with particular attention to how bounding values are to be determined for B11 and mixed B10/B11 cores.*

The power/flow/pressure/temperature ranges for the SCD analyses are determined by the steady state and transient analyses for which DNBR is calculated. The Safety Analysis group provides the statepoint conditions to be evaluated in the SCD analysis. These statepoints represent expected ranges of operation in Chapter 15 transients. The statepoints shown in Table D-6 currently bound the range of conditions for Oconee where the SCD methodology is used to calculate DNBR. As necessary, additional statepoints from Safety Analysis are evaluated using the approved methodology in DPC-NE-2005 to verify that the Statistical DNB Limit determined is still bounding for the new set of conditions.

5. *The calculation with the VIPRE-01 code using BWU-Z correlation form for B11 fuel show essentially the same results as those obtained with LYNX over the correlation's database (as documented in Addendum 1 of BAW-10199). However, the BWU-Z correlation as modified for analysis of B11 fuel has not yet been approved by the staff, and the topical report describing this correlation, Addendum 1 of BAW-10199, is still being reviewed. This means that the design limit DNBR for the parameter ranges stated in Table D-2 may not be the final approved value or range of applicability. Specifically, the database for the form of the correlation spans a pressure range of 700 to 2400 psia, not 400 to 2465 psia range stated in Table D-2. Also, the plot in Figure D-3 (see p. D-11) shows a distinct nonconservative bias with decreasing pressure (which is identical to the trend shown for the correlation in the Addendum 1 submittal). There is also a nonconservative bias with the increasing power, clearly shown by plot of measured versus predicted Critical Heat Flux (CHF) in Figure D-1. What would be the effect on the thermal-hydraulic statistical core design analysis for Oconee if the DNBR design limit of the CHF correlation for B11 fuel were to be increased, or if the range of applicability of the correlation were to be limited to pressures of 700 to 2400 psia?*

The pressure range reported in Table D-2 is consistent with the conclusion made in Addendum 1 of BAW-10199. The Addendum 1 conclusion states that the correlation parameter range for the BWU-Z correlation with the Mark-B11V multiplier is the same as the BWU-Z correlation. The data base for the Mark-B11 fuel included a pressure range of 595 psia to 2425 psia as stated in Table E-7 of Addendum 1 to BAW-10199. Also, Figure D-3 shows a slight conservatism with decreasing pressure. Likewise, Figure D-1 shows a slight conservatism with increasing power.

The pressure range for the statepoints evaluated in Appendix D is 1600 psia to 2242 psia. The pressure/temperature conditions for these statepoints were selected to bound the range of fluid conditions at Oconee which will use the statistical DNBR methodology. Other DNB calculations are performed via the non-statistical DNB method. Non-statistical DNB calculations will use the applicable design limit DNBR (from the approved BWU-Z correlation, see table below). The correlation design limit DNBR (1.193) applies only at or above a nominal pressure of 1000 psia (Reference D-5 of Appendix D). In the lower pressure region (below a nominal pressure of 1000psia) the design limit DNBR in the following table will be used (Reference D-5):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.199

Attached is Table D-2 which has been updated to clarify the pressure dependency of the design limit DNBR. Also, references D-2 and D-5 have been updated to reflect the current revision of the approved topicals.

If a statepoint with pressure less than 1600 psia were identified, it would be propagated using the applicable CHF correlation standard deviation. A statepoint with pressure less than 1000 psia is not expected for Oconee SCD analyses. If a statepoint with a pressure less than 1000 psia were analyzed, the applicable design limit DNBR will be used and the impact of the higher correlation standard deviation on the statistical design limit would be directly calculated. This verifies the statistical design limit for the statepoint is bounded. If the SDL for the new statepoint is greater than the licensing limit, the higher SDL will be used when analyzing the lower pressure conditions. This is in accordance with the methodology as described in Table 7 of DPC-NE-2005.

Any changes to the CHF correlation or restrictions in its application resulting from the NRC review process will be communicated to Duke Power by the fuel vendor. If the Mark-B11 CHF correlation range of applicability is changed, the SCD analysis would be revised as needed to reflect the modification. The correlation will not be used for DNB calculations outside the parameter range stated in the approved correlation topical. If the correlation standard deviation increases above the value used in the analyses, the limiting statepoint will be re-propagated to verify the SDL given in Appendix D.

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
Instrument guide tube diameter, in. (Nom.)	0.554(1)/0.567(2)
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	1.193*

* The correlation design limit DNBR (1.193) applies only at or above a nominal pressure of 1000 psia (Reference D-5). In the low pressure region (below a nominal pressure of 1000 psia) the design limit DNBR in the following table will be used (Reference D-5):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.199

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 Pump: +/-2.0%	+/-1.0%
			3 Pump: +/-3.2%	+/-1.6%
			2 Pump: +/-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				
$F\Delta H$	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).

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, 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, Rev. 1, 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, April 1996.

PROPRIETARY INFORMATION

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