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DPC-NE-2005A, Rev. 1

DUKE POWER COMPANY
THERMAL-HYDRAULIC
STATISTICAL CORE
DESIGN METHODOLOGY

K.R. Epperson
J.L. Abbott

Submitted: September 1992

Approved: February 1995

Appendix C Approved: November 1996 | 1

Duke Power Company
Nuclear Generation Department
Charlotte, North Carolina

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 24, 1995

Mr. H. B. Tucker
Senior Vice President
Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

Dear Mr. Tucker:

SUBJECT: ACCEPTANCE FOR REFERENCING OF THE MODIFIED LICENSING TOPICAL REPORT, DPC-NE-2005P, "THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY" (TAC NO. M85181)

The staff has completed its review of the subject topical report submitted by the Duke Power Company (DPC) by letter dated September 28, 1992. The report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

The staff does not intend to repeat its review of the matters described in the report and found acceptable when the report is referenced in a license application, except to ensure that the material presented is applicable to the specific plant involved. Staff acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, DPC must publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should NRC criteria or regulations change so that staff conclusions regarding the acceptability of the report are invalidated, DPC will be expected to revise and resubmit their documentation, or to submit justification for continued effective applicability of the topical report without revision of their documentation.

Sincerely,

Robert Jones for

Gary M. Holahan, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure: NRC Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY

TOPICAL REPORT DPC-NE-2005P

FOR

THE DUKE POWER COMPANY

1.0 INTRODUCTION

By letter dated September 28, 1992 (Ref. 1), the Duke Power Company (DPC) submitted for staff review and approval a report for use in core thermal-hydraulic analysis. DPC submitted additional information on September 29, 1993 (Ref. 2) and again on February 15, 1994 (Ref. 3). This topical report and the supplemental submittals document the development of core thermal-hydraulic analysis based upon the statistical core design (SCD) methodology using the VIPRE-01 computer code (Ref.4) for the DPC plants: McGuire, Catawba, and Oconee nuclear stations.

The SCD method is a thermal-hydraulic analysis technique which computes departure from nucleate boiling (DNB) margin by statistically combining core and fuel bundle uncertainties. The submittal provides a description and justification for applying uncertainties to the DPC DNB ratio (DNBR) limit calculations using a statistical rather than a deterministic (traditional) method. In 1991, NRC, as part of a reload review, approved limited application of the SCD methodology described in References 5 and 6 for McGuire and Catawba applications. By submitting this topical report, DPC proposes to extend the use of this methodology to all DPC plants for the DNB analysis.

The objective of the subject topical report, therefore, is twofold: (1) to formally present a description of the DPC SCD methodology and (2) to justify its use for all DPC plants. In addition, DPC presented its rationale for setting two separate statistical design limits. The underlying core thermal-

hydraulic methodology based upon the use of VIPRE-01 was approved for all DPC plants (Refs. 5 and 7).

The SCD methodology presented in the topical report is discussed in generic terms in this review. Where applicable, plant-specific features are discussed separately.

2.0 STAFF EVALUATION

The review of "Thermal/Hydraulic Statistical Core Design Methodology," report DPC-NE-2005P, was performed with technical assistance from International Technical Services (ITS). The ITS review findings are contained in the Technical Evaluation Report (TER) which is attached to this safety evaluation report. The staff has reviewed the TER and has concurred with all its findings.

The traditional method for accounting for design and modeling uncertainties that enter into the determination of a DNBR assumes that key input parameters to the core thermal-hydraulic code are simultaneously at their worst level of uncertainty. The currently licensed SCD methodology for McGuire and Catawba assumes that, while the input parameters are occasionally at their worst case values, the input uncertainties are independent and it is highly unlikely that all the input parameters will take on their worst-case values simultaneously. Therefore, the application of the SCD method differs from the deterministic techniques in that the DNBR limit is obtained from statistical analysis of a series of computations as a result of propagation of uncertainties about a statepoint and associated distribution of the DNBR values. DPC has applied the SCD method to simulate the direct computation of DNBR with VIPRE-01.

The TER discusses the DPC VIPRE methodology, the current and revised SDC methodology, the selection of key parameters and uncertainties, propagation of uncertainties, calculation of the statistical design limit (SDL) for DNBR, flexibility of the methodology, and the statistical DNB behavior and use of two DNBR limits.

During the review of DPC-NE-2005P, questions were raised on the use of two DNBR limits. However, the responses were not detailed enough to supply the information needed for resolving the possible use of two DNBR limits. Therefore, the use of the SCD methodology is approved now for only the single, most-conservative DNBR limit.

3.0 CONCLUSION

The staff has reviewed the subject topical report together with the DPC responses and has found them to be acceptable with respect to documentation of the statistical core design methodology using the VIPRE computer code subject to the following restrictions:

1. The statistical core design (SCD) methodology developed by DPC, as described in the submittal, is direct and general enough to be widely applicable to any pressurized-water reactor (PWR) fuel or reactor, provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the critical heat flux (CHF) correlation subject to the conditions in the VIPRE safety evaluation report (SER). DPC committed in their topical report that its use of specific uncertainties and distributions will be justified on a plant specific basis, and also that its selection of statepoints used for generating the statistical design limit will be justified to be appropriate. This methodology is approved only for use in DPC plants.

2. Of the two DNBR limits, only the use of the single, most-conservative DNBR limit is approved.

4.0 REFERENCES

1. Letter from H. B. Tucker (DPC) to USNRC, submitting "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005P, September 28, 1992.

2. Letter from H. B. Tucker (DPC) to USNRC, submitting "Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," September 29, 1993.

3. Letter from H. B. Tucker (DPC) to USNRC, submitting "Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," February 19, 1994.

4. Electric Power Research Institute, "Acceptance for Referencing of Licensing Topical Report VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM Vols. 1-4," May 1, 1986.

5. Duke Power Company, "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004, December 1988.

6. Letter from H. B. Tucker (DPC) to USNRC, submitting "Handouts Presented in the October 7 & 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.

7. "Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2003P-A, August 1988.

ITS/NRC/94-3

TECHNICAL EVALUATION:
THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY
DPC-NE-2005P
FOR
DUKE POWER COMPANY

P.B. Abramson
H. Komoriya

Prepared for
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under NRC Contract No. NRC-03-90-027
FIN No. L1318

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TECHNICAL EVALUATION:
THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY
TOPICAL REPORT DPC-NE-2005P
FOR
THE DUKE POWER COMPANY

1.0 INTRODUCTION

DPC-NE-2005P, dated September 1992 (Ref. 1) was submitted by Duke Power Company (DPC) for NRC review and approval. Additional information was submitted on September 29, 1993 (Ref. 2) and on February 19, 1994 (Ref. 3). This topical report and the supplemental submittals document the development of core thermal-hydraulic analysis based upon statistical core design (SCD) methodology using the VIPRE-01 computer code (Ref. 4) for the DPC plants; McGuire, Catawba (M/C) and Oconee Nuclear Stations.

The SCD method is a thermal-hydraulic analysis technique which computes DNB margin by statistically combining core and fuel bundle uncertainties. The submittal provides a description and justification for applying uncertainties to the DPC DNBR limit calculations using a statistical rather than a deterministic (traditional) method. In 1991 NRC, as part of a reload review, approved limited application of the SCD methodology described in References 5 and 6 for M/C applications. By submitting this topical report DPC proposes to extend the use of this methodology for the DNB analysis of all DPC plants.

The objective of the subject topical report, therefore, is twofold: (i) to formally present a description of the DPC SCD methodology; and (ii) to justify its use for all DPC plants. In addition, DPC presented their rationale for setting two separate statistical design limits. The underlying core thermal-hydraulic methodology based upon the use of VIPRE-01 was approved for all DPC plants (Refs. 5 and 7).

The SCD methodology presented in the topical report is discussed in generic terms in this review. Where applicable, plant specific features are discussed separately.

2.0 SUMMARY OF TOPICAL REPORT and SUPPLEMENTS

The topical report DPC-NE-2005 and its associated submittal document descriptions of DPC's VIPRE-01 based statistical core design (SCD) methodology for all of DPC's nuclear stations. The SCD methodology described in the topical has been approved as part of another review in a limited scope. The submittal formalizes the documentation of methodology description and justification for applying uncertainties to the DPC DNBR limit calculations using a statistical rather than a deterministic (traditional) method, since DPC proposes to extend the application of this methodology to

DNB analysis of all DPC plants.

In addition, DPC presented its rationale, and limited justification, for setting two separate statistical design limits due to sensitivity of DNB to the axial power distributions.

3.0 DPC Statistical Core Design Methodology

The traditional method for accounting for design and modeling uncertainties that enter into the determination of a DNBR assumes that key input parameters to the core thermal-hydraulic code are simultaneously at their worst level of uncertainty. The currently licensed SCD methodology for McGuire and Catawba assumes that, while the input parameters are occasionally at their worst case values, the input uncertainties are independent and it is highly unlikely that all the input parameters will take on their worst case values simultaneously. Therefore, the application of the SCD method differs from the deterministic techniques in that the thermal-hydraulic limit analyses are performed by statistical analysis of a series of computations as a result of propagation of uncertainties about a statepoint and associated distribution of the DNBR values. DPC has applied the SCD method to simulate the direct computation of DNBR with VIPRE-01.

3.1 DPC's VIPRE Methodology

DPC has, in place, NRC approved DNB methodology using the VIPRE-01 computer code for all DPC plants. Both the current and revised SCD methodologies are based on use of such NRC approved VIPRE methodology.

3.2 Current Methodology

The current DPC SCD methodology, based upon the B&W SCD method, relies upon the use of the response surface model (RSM) to evaluate the impact of uncertainties associated with each of the key parameters upon the DNB behavior. Therefore, the range of applicability of the SCD method (therefore the RSM) is limited by the range of values from which the composite design points, used to determine the RSM equation, are selected.

In order to overcome the main limitation of the current SCD methodology with respect to the statepoints which fall outside of the SCD range but which must nevertheless be analyzed for certain transients, DPC developed a simplified method which used VIPRE-01 directly and avoided use of the RSM.

The simplified method bypasses the RSM by directly computing DNBR with the VIPRE-01 code based on the values for the key variables generated by the propagation of uncertainties through the use of the Monte Carlo method. An SCD limit is determined for each case as before and compared against the SDL.

3.3 Revised Methodology

The revised methodology, an extension of the simplified method, is similar to other SCD methodologies in that (1) key parameters are selected, (2) their associated uncertainties are propagated about a statepoint and (3) a large

number of DNBR's are calculated. However, with this methodology the intermediate step of developing the RSM is eliminated. Instead, statistical behavior at a statepoint is evaluated by observing the distribution of the DNBR values and the mean and standard deviation of DNB for the given conditions computed by use of a Monte Carlo method for selection of values of the independent variable. All DNBR calculations are performed directly by the use of the thermal-hydraulic code VIPRE.

This is an advantage since the applicability issue with the previous method is eliminated. Further, if an assumed uncertainty should become non-bounding, the limiting statepoint can be re-evaluated to determine the impact of the changed parameter on the SDL.

3.3.1 Selection of Key Parameters

The key parameters (including reactor power, core flowrate, core exit pressure, core inlet temperature, radial power distribution and axial peak magnitude and location) which significantly impact the calculation of DNBR used in the revised methodology are the same as those used in the previous SCD methodology.

As DPC stated, these key parameters associated with DNBR are generic to US PWRs and are independent of reactor design. Plant specific information determines the uncertainties associated with each parameter.

3.3.2 Selection of Uncertainties

As in the previous SCD formulation, in order to statistically combine the effect of the uncertainties of the parameters, DPC determined the uncertainties, uncertainty distributions and the uncertainty standard deviations. An uncertainty distribution is established for each of the seven variables with the nominal state conditions as the center. DPC's rational for assignment of uncertainty distribution was that a normal distribution was assumed when the uncertainty was due either to measurement uncertainty or a known statistical uncertainty distribution. Whenever such assumption could not be reasonably made, DPC chose the conservative approach of assuming a uniform distribution with estimated reasonable upper and lower bounds.

In addition to the seven variables related to the core and fuel conditions, two other variables related to the analysis method are assumed to impact computation of DNBR; code/model uncertainty and CHF correlation uncertainty. The uncertainties associated with the code/model allows for uncertainties due to the thermal hydraulic code and VIPRE core models.

The licensee stated in the topical report that the uncertainties and distributions will be justified on a plant-specific basis in the reload report for the first application of this methodology.

3.3.3 Propagation of Uncertainties

In order to combine the uncertainties to compute an overall DNBR uncertainty, a Monte Carlo method analysis is performed using the distribution of

uncertainties defined with each variables. A Monte-Carlo computation is used to select sets of values at random (weighted by the distribution functions) about a statepoint of interest selected from a list of statepoints which form the basis for the statistical design limit.

3.3.4 Calculation of Statistical Design Limit (SDL) for DNBR

Using the Monte-Carlo generated input for the DNB computation, VIPRE-01 is run to calculate the DNBR for each case in a statepoint. The statistical DNB evaluations are performed at two levels. The first level of evaluation taking 500 propagated cases per statepoint is used to determine the DNB behavior over the entire analysis space. The second group of statepoints have 3000 cases each and contains a selected subset of the first group used to evaluate the statistical DNBR values and to improve the associated variance. Statistical analysis is then performed on the set of MDNBRs so generated to determine the statistical design limit (SDL) to replace the traditional DNB limit.

The statistical design limit is determined from the largest coefficient of variation based on the DNBRs computed by the Monte Carlo computations referred to above which avoid DNB at a 95% probability/95% confidence level.

3.3.5 Flexibility of the Methodology

DPC selected a few cases to demonstrate the flexibility of the methodology to changes in any of the key parameter uncertainty distribution, fuel designs or statepoint conditions.

The methodology is direct and general enough to be widely applicable to any fuel or reactor provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the CHF correlation and the uncertainties and associated distributions are reasonable.

3.3.6 Statistical DNB Behavior and Use of Two DNBR Limits

From the cases run with the above described method, DPC observed a dependence of statistically determined DNBRs on the axial location of the power peak, the magnitude of that peak and DNB location. When examining a series of calculations with the peak located in the lower 2/3 of the core and what DPC characterized as "flatter" power profiles, DPC observed a non-linear relationship between the DNBR responses and the axial power peak location and magnitude of the power peak. The study also indicated, for those cases, that the predicted limiting SDLs involved the DNBR occurring at the end of channel. DPC concluded that higher sensitivity of DNBR to certain key parameters was accompanied by higher SDL for the statepoints selected.

Observing the DNBR sensitivity and coupling that to the fact that Chapter-15 type analyses are all performed with power profiles which do not yield DNBRs near the limit, DPC proposed to divide the continuous DNBR space into two regions by the degree of predicted DNBR sensitivity to the axial power distribution: (1) One region contains the DNBRs predicted using the end-of-channel MDNBR limited axial power distributions (flatter and bottom-peaked)

and (2) the other region contains all others. Correspondingly, DPC calculated separate statistical design limits for these regions. In the region associated with the flat and bottom-peaked power distributions, the predicted SDL was higher and a lower SDL value was computed in the other region. The higher limit is the one which would be used if the traditional one-limit methodology is to be approved.

The net result of DPC's proposed double limit would be that the lower limit would be used in all Chapter-15 type analyses. The method for determining the line separating the two areas has a fundamental impact since use of the lower limit results in a large gain in the margin.

However, the space to be divided is a continuous space, and there is a gradual transition from the region of higher DNBR limit to the region of lower DNB limits. DPC presented no definitive analytical method for dividing this space. Furthermore, any subdivision could result in reduction of the SDL from the current bounding DNBR limit and the result would be non-conservative.

4.0 CONCLUSION

The subject topical report together with DPC responses were reviewed and found to be acceptable with respect to documentation of the statistical core design methodology using the VIPRE computer code subject to the following limitations and restrictions:

1. The DPC developed statistical core design (SCD) methodology, as described in the submittal, is direct and general enough to be widely applicable to any PWR core, provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the CHF correlation subject to the VIPRE SER conditions. Furthermore, DPC must demonstrate that DPC's use of specific uncertainties and distributions based upon plant data and its selection of statepoints used for generating the statistical design limit are appropriate.

2. This methodology is approved only for use in DPC plants.

3. For the reasons set forth in Section 3.3.6, it is recommended that (i) of the two DNBR limits, only the use of the single most conservative DNBR limit be approved and (ii) the use of two SDLs not be approved at this time and that it be handled as a separate issue to be resolved in the future.

5.0 REFERENCES

1. "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005P, September 1992.

2. Letter from H.B. Tucker (DPC) to USNRC, Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," September 29, 1993.

3. Letter from H.B. Tucker (DPC) to USNRC, "Thermal/Hydraulic Statistical

Core Design Methodology, DPC-NE-2005," February 19, 1994

4. "Acceptance for Referencing of Licensing Topical Report VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM Vols. 1-4," May 1, 1986.

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

6. Letter from H.B. Tucker (DPC) to USNRC, submitting "Handouts Presented in the October 7 & 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.

7. "Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2003P-A, August 1988.

ABSTRACT

This report presents Duke Power Company's methodology for performing statistical core thermal-hydraulic analyses. This method uses the models and thermal-hydraulic code currently approved for the Oconee and the McGuire/Catawba Nuclear Stations. The analyses method is based on DNBR limits that statistically account for the effects on DNB of key parameters such as reactor power, temperature, flow, and core power distribution. This report details the methodology development, the application to Duke plants, and the process for future technical enhancements and application to non-Duke reactors.

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<u>Definitions</u>

Case - A unique set of conditions analyzed by the thermal-hydraulic computer code. These conditions are based on a statepoint and include individual statistical variations of each key parameter.

Design DNBR Limit (DDL) - A numerical DNBR value that includes margin above the statistical design limit and is used for DNBR analyses. The DDL is calculated by multiplying the SDL by a fixed factor such as 1.10.

Key Parameter - A physical parameter that is important to the calculation of DNBR.

Statepoint - A unique set of fluid and reactor conditions evaluated for DNBR performance. These conditions include reactor power, pressure, temperature, coolant flow rate, and a three dimensional nuclear power distribution.

Statistical Core Design (SCD) - An analysis method that statistically combines the effects of all key parameter uncertainties associated with DNB predictions.

Statistical Design Limit (SDL) - A numerical DNBR value resulting from a SCD analysis that ensures, with a 95% probability at a 95% confidence level, DNB will not occur.

Statistical DNBR - The numerical value calculated by the SDL equation for a specific statepoint.

1.0 INTRODUCTION

The thermal-hydraulic design methodology accounts for the effects on DNB of the uncertainties of key parameters such as power, pressure, temperature and flow. Statistically combining these effects yields a better quantification of the DNB margin which, in turn, enhances core reload design flexibility. This report details the thermal-hydraulic statistical core design methodology developed by Duke Power Company for application to pressurized water reactors.

Several different statistical DNB analysis methods have been approved and are currently in use by various vendors and utilities. All the methods have slight differences but the major similarity is the basic concept that statistical behavior is defined by the sensitivity of DNB to key parameters and their associated uncertainties. When this relationship is well defined, a high degree of confidence in the applicability of the statistical DNB limit is assured.

1.1 CURRENT METHODOLOGY

The Thermal-Hydraulic Statistical Core Design (SCD) analysis method currently licensed for use by Duke Power Company is based on a Response Surface Model (RSM) prediction of DNBR behavior over a range of key parameters (Reference 3). The RSM is used to evaluate the impact of uncertainties on each parameter about a statepoint for a large number of cases. Figure 1 shows an overall process flowchart for the RSM

based SCD analysis. This method has been approved by the NRC for use on the McGuire and Catawba Nuclear Stations.

1.2 REVISED METHODOLOGY

Duke Power Company has developed an alternative method to evaluate the statistical behavior of DNBR that both simplifies and enhances the accuracy of the original process. The simplified method uses the VIPRE-01 thermal-hydraulic computer code (Reference 1) to calculate the DNBR values for each set of reactor conditions. With this method, the intermediate step of developing and analyzing DNB response with the RSM is eliminated. Besides this enhancement, the overall process is identical to the currently approved methodology. Figure 2 shows the flowchart for the revised approach. Note that the major difference is the elimination of the first three steps shown in Figure 1. The revised methodology was used to determine the statistical design limit for three transient statepoints in Reference 3. Limited application of this methodology was reviewed and approved by the NRC for McGuire/Catawba thermal-hydraulic analyses as part of the review of Reference 3.

The revised SCD methodology is identical in most respects to other statistical thermal-hydraulic analysis methodologies. Key DNBR parameters are selected, their associated uncertainties are propagated about a statepoint, and a large number of DNBR's are calculated . The statistical behavior at that statepoint is evaluated by observing the distribution of the DNBR values and the mean and standard deviation of DNB for the given conditions. This same approach is repeated over a

range of statepoints. The Statistical Design Limit (SDL) is based on the largest coefficient of variation and therefore the largest statistical DNBR value for the statepoints considered.

The statistical analysis method described in this report is applied to both the Oconee (Babcock and Wilcox) and McGuire/Catawba (Westinghouse) plant designs. The main body of this report details the specifics of the method and gives typical results. Two Appendices are included that contain plant specific information and results. This is necessary due to the differences in CHF correlations, fuel design, and specific uncertainties for each plant design. Appendix A contains the specific information for Oconee and Appendix B contains the same information for McGuire/Catawba. The plant specific thermal-hydraulic models and computer code configurations described in Reference 2 (DPC-NE-2003P-A) and Reference 3 (DPC-NE-2004P-A) are used in this analysis without modification.

This method of developing an SCD limit provides a more accurate representation of statistical DNB behavior because the thermal-hydraulic code is used directly to perform all DNBR calculations. Rather than relying on an algorithm such as the RSM, this methodology consists of over 151,000 individual VIPRE-01 cases at various statepoints. Because of the mechanistic approach used by this analysis, [

]

1.3 FUTURE USES

One benefit of the revised thermal-hydraulic analysis method is the ability to analyze factors outside of the original scope of analysis for a particular plant. This is due to the fact that the thermal-hydraulic code is used directly to determine statistical behavior. For example, if an assumed uncertainty should become non-bounding, the limiting statepoint can be re-evaluated to determine the impact of the changed parameter on the SDL. This method can also be used to evaluate a statepoint outside the range of the original key parameters assumed. If the statepoint statistical DNBR does not exceed the SDL, the statepoint can apply the licensed limit.

If the statepoint statistical DNBR does exceed the limit, appropriate measures, such as increasing the design DNBR limit (DDL) for that statepoint's analyses, can be used to ensure conservative DNBR limits are used. (The design DNBR limit approach is discussed in Section 2.5 of this report and Section 6.5 of Reference 3). This higher design limit will mean lower allowable radial power distributions for the affected statepoint. The higher limit would apply to all the subsequent analyses performed on that set of conditions. Another alternative to increasing the design DNBR limit is to use the available margin between the existing SDL and design DNBR limits to account for the change.

Secondly, this statistical analysis method shows generic DNB behavior that extends across fuel designs and plant types. The limiting SDL value is primarily affected by the particular Critical Heat Flux (CHF) Correlation used, the fuel assembly design, and the key parameter uncertainties. This allows the methodology to be applied to new or revised CHF correlations, new fuel assembly designs, or non-Duke plants, requiring only the submittal of an additional Appendix that provides the same information as included in the two attached.

2.0 STATISTICAL CORE DESIGN METHODOLOGY

The procedure for determining the statistical DNBR limit (SDL) contains four steps:

1. Selection of key parameters
2. Selection of uncertainties
3. Propagation of uncertainties
4. Calculation of the statistical DNBR limit (SDL).

The key parameters associated with DNBR are generic to pressurized water reactors and are independent of reactor design. The important plant specific information is the uncertainties associated with each parameter.

2.1 SELECTION OF KEY PARAMETERS

The key parameters used in this analysis are the same as those used in Reference 3 for SCD calculations. These are the parameters which significantly impact the calculation of DNBR and include:

- Reactor Power
- Core Flow Rate (including effects of core bypass flow)
- Core Exit Pressure
- Core Inlet Temperature
- Radial Power Distribution (including Hot Channel Factors)
- Axial Peak Magnitude
- Axial Peak Location

These seven parameters are used to set limits when performing reload thermal-hydraulic analyses. A statepoint in this analysis is a defined by a combination of all seven of these parameters.

The range of individual key parameter values in this analysis are based on statepoints that are using or will use the SCD DNB methodology. A majority of the statepoints analyzed have mean Minimum DNBR (MDNBR) values close to the statistical design limit itself. Table 1 shows typical statepoints that form the basis for the statistical design limit {Table 1 in the Appendices shows the statepoints analyzed for each plant}. Table 4 in the Appendices contains the range of values for each key parameter represented by the analyzed statepoints.

Since this method mechanistically evaluates each statepoint, new or revised statepoints can be easily evaluated in the same manner. If, for example, the plant is uprated to a higher licensed power level or the pressure/temperature points change or a new transient statepoint is calculated, a propagation of the revised conditions about the limiting point would be performed. If the licensed SDL is conservative, no further action would be required. If the statistical DNBR value is higher, appropriate compensatory measures will be applied to ensure the allowable DNB behavior for the statepoint is conservatively bounded.

Duke Power's reload methodology, described in References 4 and 5, gives special attention to the axial power distribution (axial peak location and magnitude) in determining acceptable DNB performance. The axial peak location and magnitudes evaluated in this analysis are concentrated about a selected region. The axial power distribution area of interest is based on the peak magnitudes and locations that are typically predicted during the standard cycle design process. Figures 3A and 3B show a graphic representation of typical axial peak values (Fz) and locations (Z) calculated by the physics codes. Figure 3A is for Oconee and Figure 3B shows the same data for McGuire and Catawba.

2.2 SELECTION OF UNCERTAINTIES

A statistical core design analysis combines the effects of individual key parameter uncertainties that significantly affect DNB. Typical uncertainties for a reactor design are shown in Table 2 {Table 2 in each of the Appendices shows the plant specific values}.

Distributions for the uncertainties are assumed to be either normal or uniform. The basis for the type of distribution assumed for each key parameter is included in the Appendices. Two additional uncertainties are included, one for the CHF correlation and one for code/model conservatism. The CHF correlation uncertainty is based on the standard deviation of the correlation data base and accounts for the correlation's uncertainty in DNB predictions. The code/model uncertainty allows for thermal-hydraulic code uncertainties and simplified versus detailed core model differences.

2.3 PROPAGATION OF UNCERTAINTIES

Multiple random cases are generated for each statepoint by independently varying all key parameters according to their associated uncertainty value and distribution. The SAS (Reference 6) statistical computer package random number function generators are used to create the necessary distributions. The key parameter distributions are calculated individually based on the type of uncertainty distribution and uncertainty magnitude.

There are two different types of uncertainties analyzed. The first type, denoted additive, is an uncertainty that has a fixed value. An example of this is the RCS temperature uncertainty of +/- 4 degrees F (see Table 2). The value is the same number of degrees F everywhere it is applied. The second type of uncertainty is called multiplicative and is based on a percentage of the parameter. An example of this is the radial power distribution uncertainty (3.25% in Table 2). Here,

the radial peak used in each statepoint has an impact on the magnitude of the uncertainty. This statistical method of application accounts for both the uncertainty magnitude and distribution type (normal or uniform).

A total of either 500 or 3000 propagated cases (one case being a set of the seven key parameters) are generated for each statepoint. The different propagation sizes are compared to verify that the statistical behavior is consistent between the two levels of analysis and to be confident that the most limiting SDL is determined. Table 3 contains an example of key parameter propagations that together make up ten DNB cases for a given statepoint. The values were extracted from a typical 500 case propagation.

As stated previously, this analysis method allows for direct evaluation of the impact of increased uncertainties. If an uncertainty value assumed in the original analysis is exceeded in the future, the limiting statepoint can be re-analyzed with the changed value. If the statepoint statistical DNBR does not increase above the licensed limit, no further action is required. If it does, proper compensatory measures can be applied.

2.4 CALCULATION OF THE STATISTICAL DNBR LIMIT

After the VIPRE-01 code is used to calculate the MDNBR's for each case in a statepoint, the code/model and CHF correlation uncertainties are applied and the coefficient of variation (CV) is calculated as

described in Reference 3. Cases that yield either a MDNBR value of less than 1.0 or that exceed the quality limit of the CHF correlation used are excluded from the data base prior to calculating the coefficient of variation. The distribution of MDNBR's is checked for normality by performing the D'Agostino (or D Prime) test on the final set of MDNBR values for each statepoint.

The appropriate Chi Square (Chi^2) and K factor (K) multipliers are determined based on the final number of MDNBR's for each statepoint. The statistical DNBR value for each statepoint is then calculated by the same equation as used in Reference 3,

$$SDL = 1.0 / \{1.0 - (K * Chi^2 * CV)\} .$$

Table 4 contains example results of the mean, standard deviation, coefficient of variation, and the statistical DNBR values calculated for the Table 1 statepoints. {Table 3 in the Appendices contains the plant specific data.}

Table 4 contains two groups of statepoints in separate sections. This is because the statistical DNB evaluations in this analysis were completed at two levels. The first level of evaluation (500 propagated cases/statepoint) is used to determine the DNB behavior over the entire analysis space. The intent of the 500 case runs is to determine DNB behavior with respect to axial and radial peaking conditions, core power level, and changes in fluid conditions.

The second group of statepoints have 3000 cases each and are a selected subset of the first group (denoted by -T after the statepoint number). This group is used to determine the SDL of DNB analyses for each reactor type. Figures 4A (500 cases) and 5A (3000 cases) graphically show the results for Oconee at a selected set of fluid conditions. Figure 6A shows the comparisons of the same axial peak locations and magnitudes for different fluid conditions. Figures 4B, 5B, and 6B show the corresponding graphs for the McGuire/Catawba statepoints.

2.4.1 VARIANCE OF STATISTICAL DNB BEHAVIOR

Comparing all these Figures showing the statistical DNBR for [ ] across a range of fluid conditions and for different fuel/reactor types, a significant dependency [ ] is observed. [ ] show a more limiting statistical DNBR behavior than the remaining points. To evaluate this, the sensitivity of DNBR [ ] was evaluated in two manners.

First, the sensitivity of DNB [ ] was determined. This was done by [ ] constant and analyzing [ ] Figure 7A shows the sensitivity of DNB [ ] for the BWC correlation (Oconee). Figure 7B shows the sensitivity for the BWCMV correlation (McGuire/ Catawba).

Two items of interest are displayed in this representation. The first fact is that the slope [ ] Secondly, the slope in this area [ ] on the remainder of the graph. The absolute value of the slope is the important factor in determining the statistical response of a key parameter (slope is the sensitivity of DNBR [ ] This indicates that [ ] will have a different statistical behavior than the area where the slope is less steep. Note the agreement between Figures 7A and 7B (different fuel assembly designs and CHF correlations). This consistency continues to affirm that this observation is a mechanistic DNB behavior.

The second sensitivity evaluation varied all key DNB parameters of a statepoint by their uncertainty magnitude and calculated the slope for each (Δ DNBR / Δ parameter). These results are shown in Table 5. This type of analysis shows [ ]

Additionally, there is another phenomenon that is also present with [

]

This more limiting statistical behavior has been evaluated for generic applicability and was found to occur for each reactor type and CHF correlation as shown by Figures 4A, 4B, and 4C. Figure 4C is the same core geometry and statepoints as 4B but with the DCHF-1 CHF correlation (Reference 7). The statistical behavior [

] All these factors point to the conclusion that this more limiting statistical variance [

] is a generic, mechanistic DNB behavior and as such is applicable to any CHF correlation and core model (Oconee, McGuire, Catawba, etc).

2.4.2 FLEXIBILITY OF THE ANALYSIS METHOD FOR MODIFIED PARAMETER EVALUATIONS

Several different comparisons are included to demonstrate the ability of this method to address changes in core models or uncertainty distributions. Table 6 shows the results of three different

evaluations. The first section includes two points that show the results of changing a single key parameter's uncertainty distribution from normal to uniform. Statepoints 33 and 34 from the McGuire/Catawba evaluation were identical in all respects except for the RCS flow distribution. In Statepoint 33, the distribution was normal (same for all other statepoints) and in Statepoint 34 the distribution was changed to uniform. The affects of this single parameter distribution change is readily calculated and shown to be negliable.

The section has two points that show the impact of a VIPRE-01 model change. Statepoints 37 and 38 both have identical conditions and uncertainties. Statepoint 37 used the eight channel McGuire/Catawba model from Reference 3 while Statepoint 38 used the fourteen channel model from Reference 8. Again, the comparison is easily accomplished and Table 6 shows the difference in the statistical DNBR values.

The third section contains a group of points that shows the comparison between Westinghouse OFA and Babcock Wilcox Mark-BW 17x17 mixing vane fuel. Four statepoints were run with both fuel types at the same fluid and power distribution conditions. The difference between the models is the changed subchannel flow areas, wetted and heated perimeters, gap connections, and grid form loss coefficients to correctly reflect each fuel type. The comparison shows that the OFA fuel model's behavior is the same as the Mark-BW model and the Mark-BW SDL conservatively bounds OFA fuel for McGuire and Catawba analyses.

2.4.3 FUTURE APPLICATIONS OF SCD METHODOLOGY

The fact that this analysis method is direct allows this statistical approach to be applied to any fuel type or reactor using an NRC approved thermal-hydraulic model and CHF correlation. Even if DNB behavior showed a stronger or weaker functionality for a different core design or CHF correlation, this method would correctly reflect this behavior in the statistical design limit or limits determined. If a new CHF correlation is used by Duke or if a different plant is analyzed, an additional Appendix will be submitted to the NRC detailing the model, CHF correlation, uncertainties, and statepoints used to determine the SDL for the plant specified.

2.5 APPLICATION OF THE SCD LIMIT

Since the statistical DNBR behavior demonstrated in this analysis shows [

]

The method for applying [

] Additionally, DNB analyses may be performed using a design DNBR limit (DDL) which includes margin above the statistical design limit [ ]

Should an analysis be performed that uses a new CHF correlation, for a non-Duke reactor, or for a new fuel design, statepoints [ ] will be analyzed to confirm the generic DNB behavior assumption and to determine the SDL [ ] This information will be reported to the NRC by submitting a new Appendix similar to Appendix A and B.

3.0 CONCLUSIONS

The methodology described in this report shows the major factors affecting statistical DNB behavior are [ ]

Since the statistical DNB behavior is controlled by these global parameters, [

]

This analysis method can be used to evaluate new fluid statepoints or revised uncertainties directly to determine the statistical limit. As long as the SDL is not exceeded, the established limits can be applied unmodified. If the statistical DNBR value for the new conditions is higher than the current limit, appropriate compensation measures such as increasing the design DNBR limit for the statepoint or using available margin between the design and statistical limits can be used. These actions penalize the statepoint by reducing the allowable radial peaking to ensure acceptable DNB behavior.

Since Duke's statistical thermal-hydraulic design methodology relies solely on DNB behavior, any PWR facility can be analyzed using this approach with an appropriate core model and bounding uncertainties. Also, new fuel designs or critical heat flux correlations can be evaluated to determine the appropriate SDL. The results of such an analysis would be submitted to the NRC for approval in the form of an additional Appendix that would contain the following:

1) Identification of the plant, fuel type, and CHF correlation with appropriate references to the approved fuel design and CHF correlation topicals.

2) Statement of the thermal-hydraulic code and model used with appropriate references to the approved code topical report.

3) A list of the key parameters, their uncertainty values, and distributions.

4) A list of the statepoints analyzed.

5) The Statistical Design Limits and how they are applied.

Table 7 contains a listing of some anticipated conditions and the corresponding actions.

4.0 SUMMARY

This report describes the analysis method used to determine the statistical core design DNB limit for reactor core thermal-hydraulic analyses. This methodology is used to account for the impact on DNB of the uncertainties of key parameters such as power, pressure, temperature, and core peaking. The methodology determines the statistical behavior of DNBR with respect to all these key parameters for many different statepoints and provides a method of applying the SCD DNB limits derived.

Duke has observed a significant statistical DNB behavior dependency
[

] The

specific SCD DNB limits for the Oconee and McGuire/Catawba units are stated in the Conclusions section of the attached Appendices.

5.0 <u>REFERENCES</u>

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

2. Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003P-A, Duke Power Company, Charlotte, North Carolina, October 1989.

3. McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2004P-A, Duke Power Company, Charlotte, North Carolina, December 1991.

4. Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors, DPC-NE-2011P-A, Duke Power Company, Charlotte, North Carolina, March 1990.

5. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002A, Duke Power Company, Charlotte, North Carolina, October 1985.

6. SAS Language Reference, Version 6, First Edition, SAS Institute Incorporated, Cary, North Carolina, 1990.

7. DCHF-1 Correlation For Predicting Critical Heat Flux In Mixing Vane Grid Fuel Assemblies, DPC-NE-2000A-P, Duke Power Company, Charlotte, North Carolina, September 1987.

8. Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000A-P, Revision 1, Duke Power Company, Charlotte North Carolina, December 1991.

9. BWC Correlation Of Critical Heat Flux, BAW-10143P-A, Babcock and Wilcox, Lynchburg, Virginia, April 1985.

10. BWCMV Correlation Of Critical Heat Flux In Mixing Vane Grid Fuel Assemblies, BAW-10159P-A, Babcock and Wilcox, Lynchburg, Virginia, May 1986.

TABLE 1. Typical Reactor SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	$F\Delta h$
DNB Limit Line Statepoints						
1						
3						
4						
12						
14						
17						
26						
Loss Of RCS Flow Transient Statepoints						
21						
24						
29						
Uncontrolled Bank Withdrawal Transient Statepoint						
33						
Nominal Operating Statepoints						
16						
27						

TABLE 2. Typical Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Reactor 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	+/- 3.25% / 1.98%	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	+/- 16.78% / 10.2%	Normal
Code/Model	[ ]	Normal

TABLE 3. Typical Monte Carlo Propagation Statepoint Values

(Values After Uncertainty Propagation of Stpt. # 1 from TABLE 1)

Base Statepoint

Case#	Power	Press	Temp	Flow	Fz	Z	FΔh
0							

Propagation

Case#	Power	Press	Temp	Flow	Fz	Z	FΔh
1							
50							
100							
150							
200							
250							
300							
350							
400							
450							
500							

TABLE 4. Example of Typical Statepoint Statistical Results

Section 1 - 500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
DNB Limit Line Statepoints				
1				
3				
4				
12				
14				
17				
26				
Loss Of RCS Flow Transient Statepoints				
21				
24				
29				
Uncontrolled Bank Withdrawal Transient Statepoint				
33				
Nominal Operating Statepoints				
16				
27				

TABLE 4 - continued   Example of Typical Statepoint Statistical Results

Section 2 - 3000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
DNB Limit Line Statepoints				
3-T				
4-T				
12-T				
14-T				
Nominal Operating Statepoint				
16-T				

TABLE 5. Individual Key Parameter Slopes At Statepoint Conditions

Key Parameter*	Stpt 6	Stpt 25	Stpt 9	Stpt 21

[ ]

The statepoints listed above are from the McGuire/Catawba 500 case runs. [ ] Statepoints 6 and 25 [ ] Statepoints 9 and 21 [ ]

* All values shown are in %DNBR per unit of parameter (Δ DNBR / Δ parameter). For example, the first entry in the table of [ ] means a [ ] DNBR change for every 1% power change.)

Table 6. Uncertainty and Model Changes - Impact On Statistical DNBR Behavior

Uncertainty Distribution Change

The following two statepoints show the change in the statistical behavior for a fixed set of conditions if the RCS flow uncertainty distribution is changed from normal to uniform.

Statepoint #	RCS Flow Uncertainty Dist.	Coefficient Of Variation	Statistical DNBR
33	Normal		
34	Uniform		

Thermal-Hydraulic Model Detail Change

The following two statepoints show the change in the statistical behavior for a fixed set of conditions using two different VIPRE-01 models.

Statepoint #	McGuire/Catawba VIPRE-01 Model	Coefficient Of Variation	Statistical DNBR
37	8 Channel		
38	14 Channel		

Minor Fuel Geometry and Design Changes

The following eight statepoints show the change in the statistical behavior for the geometry and form loss coefficient changes between Mark-BW and OFA fuel assemblies for the same fluid and peaking conditions.

MARK-BW			**OFA**		
Statepoint #	Coefficient Of Variation	Stat. DNBR	Statepoint #	Coefficient Of Variation	Stat. DNBR
6			40		
12			41		
14			42		
16			43		

TABLE 7. SDL Evaluation And Re-Submittal Criteria

The following table lists different events or conditions that would require an evaluation of the applicability of an approved SDL and the subsequent actions based on the results of the analysis.

CONDITION	ACTION
Revised uncertainty larger than the limiting value used in the original analysis.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Revised uncertainty distribution.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
New statepoint.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Minor modifications to the current fuel design.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
A modified CHF correlation.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Change to a new fuel design/fuel type.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
A new CHF correlation.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
Duke analysis of a non-Duke reactor.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
New Thermal-Hydraulic Code.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.

FIGURE 1

RSM BASED SCD FLOWCHART

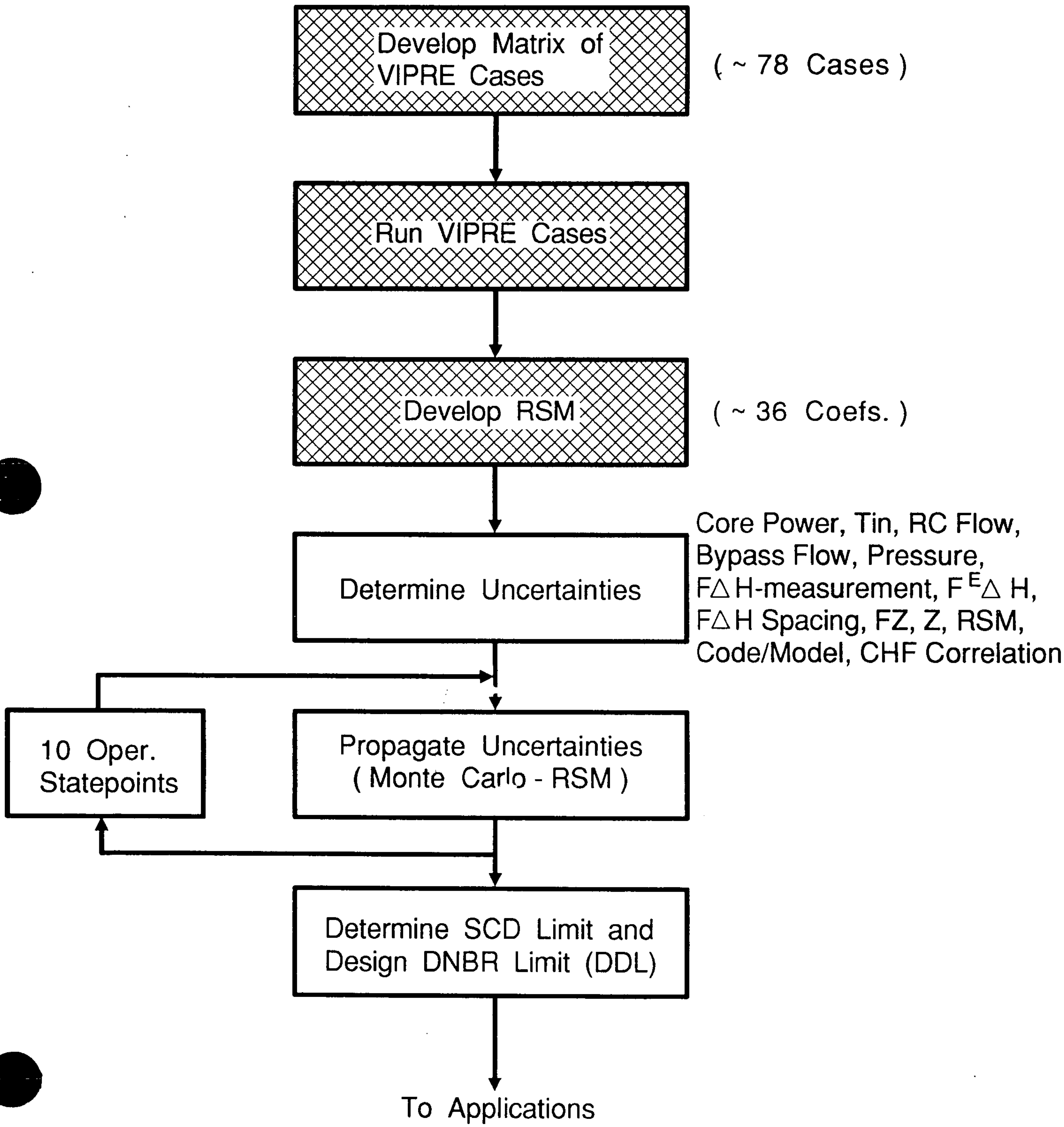

FIGURE 2

REVISED SCD FLOWCHART

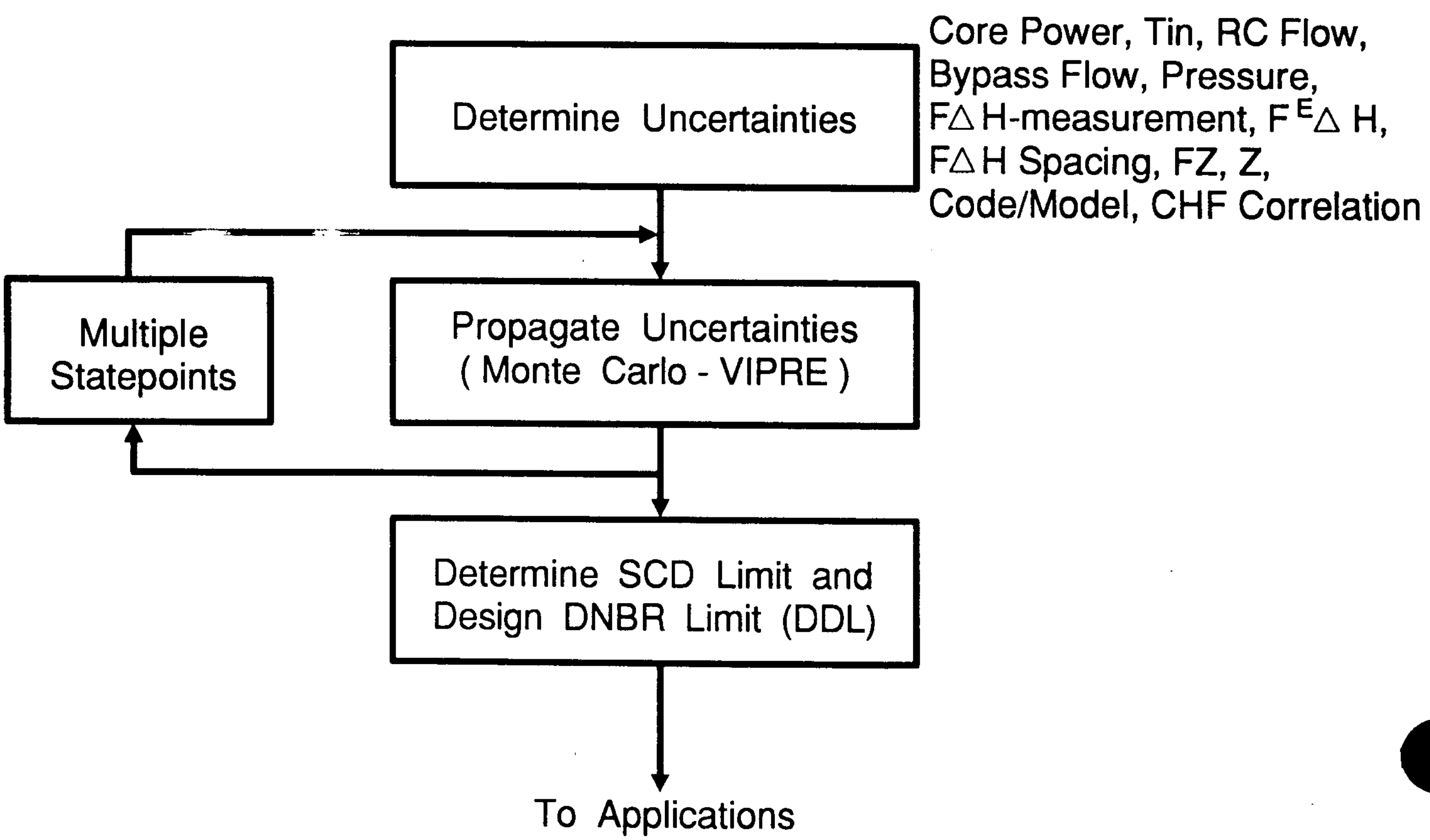

FIGURE 3A Oconee Physics Code Axial Power Distributions (Peak Magnitude and Locations)

31 Axial Peak

Z

FIGURE 3B M/C Physics Code Axial Power Distributions (Peak Magnitude and Location)

Axial Peak

Z

FIGURE 4A

Oconee SDL Distribution At Constant Conditions, BWC

500 Case Propagations

FIGURE 4B

M/C SDL Distribution At Constant Conditions, BWCMV

500 Case Propagations

FIGURE 4C

M/C SDL Distribution At Constant Conditions, DCHF-1

500 Case Propagations

FIGURE 5A

Oconee SDL's For 3000 Case Statepoints, BWC

FIGURE 5B

M/C SDL's For 3000 Case Statepoints, BWCMV

FIGURE 6A

Oconee SDL's For Various Conditions

FIGURE 6B
M/C SDL's For Various Conditions

FIGURE 7A Sensitivity of DNBR [ ] BWC

PROPRIETARY

FIGURE 7B Sensitivity of DNBR [ ] BWCMV

[ ]

FIGURE 8A

BWC

FIGURE 8B

BWCMV

FIGURE 9

Example [ ] Application

ATTACHMENT 1

Response to Request for Additional Information

September 29, 1993

Duke Power Company
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DUKE POWER

September 29, 1993

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
McGuire Nuclear Station
Docket Nos. 50-369, -370
Catawba Nuclear Station
Docket Nos. 50-413, -414
Thermal/Hydraulic Statistical Core Design Methodology,
DPC-NE-2005

By letter dated September 28, 1992, Duke Power Company submitted Topical Report DPC-NE-2005, "Thermal/Hydraulic Statistical Core Design Methodology." The NRC staff issued a request for additional information (RAI) dated July 27, 1993. Attached are the responses to the questions contained in the RAI.

In accordance with 10 CFR 2.790, Duke Power Company requests that the attached information relating to DPC-NE-2005 be considered proprietary. Information supporting this request is included in the affidavit which appears as Attachment I.

If we can be of assistance in your review please call Scott Gewehr at (704) 382-7581.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

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File: GS-801.01

ATTACHMENT I
AFFIDAVIT OF M. S. TUCKMAN

1. I am Senior Vice President, Nuclear Generation Department, Duke Power Company ("Duke"), and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing, and am authorized to apply for its withholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 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 10 CFR 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.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 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 proprietary version of the report DPC-NE-2005, "Thermal/ Hydraulic Statistical Core Design Methodology" and supporting documentation, and omitted from the non-proprietary versions.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN

This information enables Duke to:

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

(b) Respond to NRC requests for information regarding the transient response of Babcock & Wilcox and Westinghouse reactors.

(c) Support license amendment and Technical Specification revision request for Babcock & Wilcox and Westinghouse reactors.

(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.

(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 a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN (Page 3)

M. S. Tuckman, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

M. S. Tuckman

M. S. Tuckman

Sworn to and subscribed before me this 29th day of September, 1993. Witness my hand and official seal.

Linda L. Kessler

Notary Public

My commission expires May 7, 1994.

LINDA L. KESSLER NOTARY PUBLIC MECKLENBURG COUNTY, N.C.

Request for Additional Information and Responses To Topical Report DPC-NE-2005P

The questions are shown in italics and the responses immediately follow.

1. *Explain DPC's intent for this topical report. Does DPC seek its review with respect to its plants or generic PWR application? How does DPC plan to deal with the restrictions and requirements imposed by the VIPRE-01 code SER?*

The intent of this submittal is to outline a statistical Departure from Nucleate Boiling methodology. In DPC-NE-2005, DPC has outlined a statistical analysis method that is based on inherent behavior of the DNBR phenomena in pressurized water reactors. The numerical value of the Statistical Design Limit (SDL) will vary, depending on the CHF correlation used and parameter uncertainties assumed. However, direct use of the VIPRE-01 thermal hydraulic code (rather than the RSM) to calculate the phenomenological statistical variance of DNBR insures the direct applicability of this method to many varying fuel designs and parameter conditions.

DPC seeks the following approval from the NRC regarding this report:

1) Review and approval of the methodology <u>and</u> the stated statistical DNB limits for use at Oconee, McGuire, and Catawba based on the information in the body of the report and the site specific information in the Appendices.
2) Review and approval of the use of the methodology for future analyses of non-DPC reactors consistent with the commitments made in Section 1.3 and 2.5 of the report. This involves development or justification of the models and uncertainties used for any other site. If DPC were to extend this method to another PWR facility, a separate submittal will be made detailing the intent and justification for specific modeling assumptions, choice of flow models and correlations, and plant specific input data, as well as the resulting statistical DNB limits. The form of this submittal would be an additional Appendix to this report. This meets item (3) of Section 3 of the VIPRE-01 SER. The SDL would be calculated using the methodology outlined in the body of the report.

2. *DPC previously submitted two sets of DNB models for each type of plant. One was approved for use in steady-state type calculations and the other for use in transient type calculations. Since there are differences between these models on the basic level of model/input selection, discuss the impact of these differences on SDL determined. The SCD is developed based upon a series of steady state calculations. Explain how the SDL is used for transient analysis.*

Both models used by DPC were included in the statistical propagations detailed in the report. This is explained on page 14 of the report. Statepoints 37 and 38 in Appendix B are identical in fluid and peaking conditions. Statepoint 37 was propagated with the eight channel M/C model from Reference 5 and Statepoint 38 used the fourteen channel

model from Reference 6. Table 6 of the report (page 27) shows the results of this comparison. The difference in Statistical DNBR's is negligible.

The determination of whether the SCD limit can be used for a transient is based on the fluid conditions at the point of minimum DNBR (MDNBR) during the transient. If the power, pressure, temperature, and flow rate of this statepoint fall within the parameter range listed in Table 4 of the appropriate Appendix, the SDL can be used. All the statepoint statistical propagations are made from a single set of fluid and peaking conditions.

3. *Discuss how "appropriate compensatory measures" will be applied to ensure the allowable DNB behavior for the statepoint is conservatively bounded.*

Please refer to the Definitions page (Page V) of the report for the following definitions:

Design DNBR Limit, DDL
Statistical Design Limit, SDL
Statistical DNBR

The term statistical DNBR applies to a specific statepoint, the SDL is the licensed limit, and the DDL is the MDNBR value used in steady-state and transient DNB analyses.

As stated in the report, new statepoints or revised uncertainties can be evaluated directly with this method. As long as the statistical DNBR value is less than the SDL, the statepoint is conservatively bounded by the SDL. If, however, the statistical DNBR value is larger than the SDL, actions can be taken to ensure that DNB predictions for the condition meet the required 95/95 acceptance criteria. These compensatory measures include either

1) Increasing the design DNBR limit for the statepoint.
2) Using available margin present between the statistical and design DNB limits (between the SDL and the DDL).

Increasing the design DNBR limit (DDL) will increase the minimum DNBR that is allowed in the analysis of the statepoint This requires another key transient analysis input (such as maximum allowable peaking) to be reduced. Penalizing a statepoint in this manner will ensure the required DNBR protection is maintained.

Another equally valid method is to apply any unused margin already available between the statistical (SDL) and design DNBR limit (DDL). This margin is inherently retained in all analyses by using the DDL in design calculations which includes margin above the SDL. A portion of this margin is currently used to account for such things as reactor vessel flow anomalies, instrumentation biases that cannot be statistically compensated for, and physical changes to the fuel assembly not accounted for in standard models (such as rod bow). The margin remaining after all of the DNBR penalties are accounted for can be used to compensate for the increase in SDL required for a particular statepoint. Either of these methods will conservatively adjust the MDNBR limit that must be met in the analysis to ensure adequate protection is maintained.

4. *Explain why RCS flow is varied only between 100 and 106.5% and not below 100% (see Table A-1 on p. A-3) even for low flow cases.*

The percent flow listed for the low flow cases in Table A-2 is in error. The flow rate used for all the statepoints identified as Low Flow in the Comments column was [ ] Additionally, the Minimum flow value listed on Table A-4 for percent design RCS flow should be [ ] The corrected pages are included with this response.

Additionally, the flow chart in Figure 2 also contains a typo. The Propagate Uncertainties box should have the words (Monte Carlo - VIPRE) underneath. The RSM is not used at all in the revised method described by the report. A corrected page 30 is also included.

5. *Explain thoroughly how ranges of uncertainties and their associated standard deviations are determined.*

The numerical range of each uncertainty is selected to bound the value calculated for the parameter. This ensures that conservative statistical behavior is calculated and allows for changes in the uncertainty value without requiring re-analysis of the SDL.

(a) *Explain how uncertainties in instrumentation are accounted for. What is meant by the term "random uncertainty" (see Table A-2)? Explain how it is related to instrument error uncertainty.*

The term "random uncertainty" used in Table A-2 of the report means the instrument uncertainties such as sensor calibration accuracy, rack drift, sensor drift, etc., that are combined by the SRSS method. The term was used because the biases which are constant in sign (either positive or negative) are not included in the propagation of an uncertainty and must be accounted for by another means, such as a DNB penalty.

(b) *Identify the sources of the quantitative ranges of uncertainties and their associated standard deviations (for both types of plants).*

The source of the quantitative ranges and the standard deviations are provided on Table 1 of this response for each plant. The statistical propagations for each normally distributed parameter are based on the standard deviation numerical values. Uniform uncertainty propagations are based on the uncertainty numerical magnitude.

6. *Explain thoroughly the mechanistic DNB behavior observed in Figures 7A and B.*

Figures 7A and 7B in the report show the sensitivity of DNBR to axial peak location and magnitude. This sensitivity was calculated by holding all other parameters (power, pressure, temperature, flow, and radial peaking) constant. Both the BWC (7A) and BWCMV (7B) CHF correlation results are shown. These graphs show that the response of DNBR varies with axial peak conditions.

(a) *Discuss why the sensitivity to the axial peaks and locations is significantly stronger for Oconee than it is for M/C.*

The evaluations contained in the report indicate that the numerical value of the SDL is dependent on the CHF correlation used in the analysis. Table 5 in the report contained individual parameter sensitivities to DNB for the BWCMV CHF correlation in both axial peak areas defined. Table 2 in this response contains an identical sensitivity evaluation for the BWC and DCHF-1 CHF correlations in both axial peak areas.

For the region of higher statistical behavior, comparison of the BWC and BWCMV sensitivities shows the sensitivity calculated for each key parameter with the BWC correlation has slightly higher sensitivity to DNBR. This results in a higher final calculated SDL. The sensitivity values are more consistent when the same evaluation is made in the lower SDL area and the corresponding statistical DNBR's for the two correlations are almost identical. Correspondingly, the DCHF-1 correlation has lower sensitivities in both areas and has the lowest statistical DNBR in both cases.

Again, Table 2 in this response as well as Table 5 in the report (page 26) shows that the behavior is remarkably consistent between Oconee and McGuire/Catawba and is linked to axial power distribution. There is a difference in the numerical value of the statistical DNBR, and the key to this is the CHF correlation being used. DPC's conclusion is that the general behavior is mechanistic and this is proven by the consistent behavior when the sensitivity is calculated for different fuel types (15x15 non-mixing vane and 17x17 mixing vane), different fuel vendors (Westinghouse and Babcock & Wilcox), and even different CHF correlations (BWC, BWCMV, and DCHF-1).

(b) *DPC's conclusion based upon Figure 6A and B on p. 13 is not clear. Explain further.*

The discussion on page 13 and Figures 4A, 4B, 4C, 5A, 5B, 5C, 6A and 6B of the report show how the statistical DNB behavior is much more dependent on axial peak location than on the fluid parameter values for a particular statepoint, the fuel type, or the CHF correlation. The Figure 4 and 5 series show how the statistical DNB behavior changes with shifts in the axial power distribution. The axial peak location has a large impact on the statistical DNB value. By contrast, Figures 6A and 6B show how little the statistical DNB behavior changes with

large changes in the statepoint pressure, temperature, flow rate, and core power variables. This means that if the SDL is determined in either of the axial power distribution areas for one set of fluid conditions, this SDL value would be consistent even if the fluid conditions changed dramatically.

7. *Provide a table which identifies which DNB methodology is used for each transient and explain each such selection.*

The McGuire/Catawba DNB transients currently analyzed using the SCD methodology are listed in Table 3 of this response. No transients are currently analyzed for Oconee with the SCD methodology. All of the transients analyzed with the SCD methodology were selected based on the values of the individual parameters at the point of MDNBR during the transient as explained in the response to Question 2. If these values are within the range for each parameter defined on Table 4 of the appropriate Appendix, the SCD limit can be applied to the transient.

As discussed by the note below Table 4-A and 4-B, this parameter list is subject to change. One of the advantages of the explicit evaluation method describe in the report is the ability to specifically evaluate new conditions for SCD limit applicability. If a new statepoint has a parameter(s) outside the given range, it would be analyzed and if the current SCD limit is conservative, the table would be updated to show the expanded range. The transient that generated the statepoint would then be included on the internal DPC list (Table 3 of this response). This increased parameter range would not be reported directly to NRC.

8. *Explain the last two paragraphs of Section 2.4. Discuss the need to perform statistical DNB analysis in two levels and with two different sample sizes.*

The two different sample sizes were used to minimize the total number of cases propagated for each set of fluid conditions analyzed. The first level of 500 cases per statepoint is used to quickly evaluate the behavior of a statepoint with respect to the two axial peak areas. This shows the statistical DNB behavior and approximate numerical SDL value for the fluid conditions being evaluated.

The second group of 3000 case statepoints are selected to calculate the limiting SDL value for the reactor type being analyzed. The increase in number of cases to 3000 provides a more thorough evaluation of the statistical DNB response and improves statistically the Chi Square and K factor multipliers used to conservatively increase the coefficient of variation in the final SDL calculation. The licensed statistical design limit is greater than the largest value calculated in all the 3000 case propagations for each axial peak area.

DPC may increase the number of cases at a particular statepoint for future evaluations to take advantage of the improved effect on the statistical multipliers. This increase in the number of cases is consistent with the methodology as presented and does not in any way reduce the conservatism of the SDL limit calculated. Increasing the number of cases

simply reduces the statistical uncertainty associated with calculation of the coefficient of variation.

9. Explain the rational for and appropriateness of selection of certain sets of statepoints to determine the impact of changes on statistical DNBR behavior (see Table 6).

The evaluations in Table 6 of the report show how little the statistical DNB behavior is affected by small modifications in the analysis. The first section shows the change for identical conditions and models with a change in one parameter uncertainty distribution (normal versus uniform). Section 2 shows the change if a different VIPRE-01 model is used with the same fluid conditions, peaking conditions, and uncertainty distributions. The last section shows the change with the same VIPRE-01 model, fluid conditions, and uncertainties but with a different fuel design. As discussed in the response to question 6b, Figures 6A and 6B demonstrate the there is very little change in statistical DNB behavior for large changes in the statepoint pressure, temperature, flow rate, or core power variables. Thus, the sensitivity of the SDL to other changes can be evaluated using a single statepoint.

All of these evaluations were included to further demonstrate that the statistical DNB behavior and SDL are more closely related to the CHF correlation and axial power distribution than to small perturbations in individual uncertainties, VIPRE-01 models, or fuel type. This evaluations also provide the basis for the criteria for re-submittal or in-house evaluation detailed on Table 7 (as explained in the response to Question 10).

10. Explain Table 7.

Table 7 in the report is intended as a guide for use by DPC in evaluating what action must be taken for anticipated changes (a revised uncertainty, new fuel type, etc.). In all cases, the evaluations will use the methodology detailed in the report. Basically, changes that are anticipated to have a negligible or very small impact on the SDL will require internal DPC evaluation. Only changes that have a significant impact on the calculated SDL number will be submitted to the NRC for approval.

An example of the kind of anticipated events is a change in an uncertainty magnitude. For this instance, limiting SCD statepoints in each axial power distribution area will be evaluated to determine the impact on the SCD limit. If the statistical DNBR value is the same or smaller than the SDL, no additional work is required. If the value is larger, appropriate compensation measures will be used to conservatively compensate for the change (as described in the answer to Question 3). This same approach will be used for different uncertainty distributions, new fluid or peaking condition statepoints, or minor modifications to the fuel assembly design.

For changes that will have a much bigger impact on the statistical DNB behavior, the impact of the change will be evaluated and a new Appendix to this report submitted for NRC approval. This additional Appendix will have the same format and content as the

two already included in the report. Examples of when this approach would be used are a completely new fuel assembly design, a new thermal hydraulic code, a new CHF correlation, or DPC analysis of a third party's reactor.

A slight change to Table 7 is also included in the response to this question. The original table required that a modified CHF correlation would require submittal of a new Appendix. This has been changed to require an evaluation only. The term modified means the form of the CHF correlation is the same, just a single factor or multiplier has been changed or added. This change is because a modified correlation will not impact the statistical DNB behavior and will not significantly change the SDL compared to the original correlation. A modified correlation will still require a separate CHF correlation topical submittal to the NRC. Any other changes that affect the correlation form will be considered a new CHF correlation.

11. Provide the SDL if no distinctions are made of axial power distributions.

The results of the entire analysis completed in the report show how mechanistic the statistical DNB response is to axial power distribution. This mechanistic behavior was determined by direct use of the thermal hydraulic codes, models, and correlations used in DNB predictions. This behavior is consistent with different fluid conditions, fuel geometries, and CHF correlations. The one consistent fact is the larger statistical variation for a specific set of axial peaks. The use of two statistical DNB limits to address this behavior is a straight forward application. Use of a single limit would be unnecessarily conservative. However, if the appropriate distinctions are not made for the generic DNB behavior with axial power distributions, the SDL for all cases will be the largest value calculated for all the conditions evaluated. If this restriction were imposed, the SDL would be 1.43 for Oconee and 1.40 for McGuire/Catawba.

TABLE 1

Uncertainty Ranges And Standard Deviations

The following table shows the source of the quantitative range of each uncertainty and its associated standard deviation. Section 1 of the table contains the Oconee information and Section 2 the sources for the McGuire/Catawba values.

SECTION 1 - Oconee

Parameter	Source
Power	Standard deviation of 1.0% based on DPC calculations. Uncertainty value is a 2σ value (2%).
Pressure	Standard deviation of 15 psi based on DPC calculations. Uncertainty value is a 2σ value (30 psi).
Temperature	Standard deviation of 1.0 degrees Fahrenheit based on DPC calculations. Uncertainty value is a 2σ value (2 deg F).
Flow	Standard deviation of 1.0% design flow based on DPC calculations. Uncertainty value is listed as a 2σ value (2%).
$F\Delta H$	Standard deviation of 2.84% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for Oconee (Reference 1).
F_Z	Standard deviation of 2.91% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for Oconee (Reference 1).
Z	Uncertainty range of +/- 6 inches. Selected based on the noding size of nuclear codes. No standard deviation (uniform uncertainty).
Local Heat Flux HCF	Uncertainty range of [ ] Based on calculated values from the nuclear fuel vendor. Standard deviation is calculated from [ ] uncertainty value [ ]

Rod Power HCF	Uncertainty range of [ ] Based on calculated values from the nuclear fuel vendor. Standard deviation is calculated from [ ] uncertainty value [ ]
Hot Channel Flow Area	Uncertainty range of [ ]. Based on calculated values from the nuclear fuel vendor. No standard deviation (uniform uncertainty).
CHF Correlation	Standard deviation of 8.88% calculated from the BWC CHF test data base (Reference 2).
Thermal Hydraulic Code / Model	Uncertainty range of [ ] Value used in Reference 3. Standard deviation is calculated from the [ ] uncertainty value [ ].

SECTION 2 - McGuire/Catawba

Parameter	Source
Power	Uncertainty Range of 2%. Selected from Reference 5. Kept at 2% to bound specific uncertainties calculated for M/C. Standard deviation is calculated from 2% uncertainty value (2/1.64 = 1.22%).
Pressure	Uncertainty Range of 30 psi. Selected from Reference 5. Kept at 30 psi to bound specific uncertainties calculated for M/C. No standard deviation (uniform uncertainty).
Temperature	Uncertainty Range of 4 degrees Fahrenheit. Selected from Reference 5. Kept at 4 degrees to bound specific uncertainties calculated for M/C. No standard deviation (uniform uncertainty).
Flow	Uncertainty Range of 2.2%. Selected from Reference 5. Kept at 2.2% to bound specific uncertainties calculated for M/C. Standard deviation is calculated from 2.2% uncertainty value (2.2/1.64 = 1.34%).
$F\Delta H$ Measurement	Standard deviation of 1.98% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for M/C (Reference 1).

Engineering HCF	Uncertainty range of 3.0%. Selected based on the value in Technical Specifications. Standard deviation is calculated from the 3% uncertainty value (3/1.64 = 1.82%).
Spacing	Uncertainty range of 2.0%. Selected from Reference 5. Standard deviation is calculated from the 2% uncertainty value (2/1.64 = 1.22%).
F_Z	Standard deviation of 2.68% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for M/C (Reference 1).
Z	Uncertainty range of +/- 6 inches. Selected based on the noding size of nuclear codes. No standard deviation (uniform uncertainty).
CHF Correlation	Standard deviation of 10.2% calculated from the BWCMV CHF test data base (Reference 4).
Thermal Hydraulic Code / Model	Uncertainty range of [ ] Value used in Reference 5. Standard deviation is calculated from the [ ] uncertainty value [ ].

TABLE 2

Comparison of the DNB Parameter Sensitivity of Different CHF Correlations With Consistent Axial Power Distributions

The following table shows the DNB sensitivity of each key parameter for the BWC CHF correlation (Oconee), the BWCMV CHF correlation (McGuire/Catawba), and the DCHF-1 CHF correlation (McGuire/Catawba). The first comparison is of a statepoint in the higher SDL area and the second is in the lower SDL area. The fluid and radial peaking conditions for each statepoint are given in the Appendices.

CHF Correlation	1.3 Peak @ 0.2 Z	1.3 Peak @ 0.8 Z
BWC	Statepoint 63	Statepoint 75
BWCMV	Statepoint 6	Statepoint 9
DCHF-1	Statepoint 6	Statepoint 9

1.3 Axial Peak, 0.2 Z

Parameter	BWC	BWCMV	DCHF-1
Power (%)			
Pressure (psi)			
Temperature (Deg F)			
Flow (%)			
FΔH (%)			
F_Z (%)			
Z (per 6 inches)			
SDL			

1.3 Axial Peak, 0.8 Z

Parameter	BWC	BWCMV	DCHF-1
Power (% RTP)			
Pressure (psi)			
Temperature (Deg F)			
Flow (%)			
FΔH (%)			
F_Z (%)			
Z (per 6 inches)			
SDL			

All values shown are in terms of % DNB per unit of parameter.

TABLE 3
SCD Transient Limiting Statepoints

The following table shows all the M/C transients currently evaluated with the SCD methodology. The determination of whether the transient uses the SCD approach is the value of all the key parameters (power, pressure, temperature, flow, peaking) at the point of MDNBR during the transient. All values listed are from the MDNBR point of the transient.

Transient	Core Power	Core Inlet Flow (Kgpm)	Core Inlet Temperature	Pressure	$F_{\Delta H}$	F_Z	Z
Feed Line Break	[						
Partial Loss of RCS flow							
Total Loss of RCS Flow							
Uncontrolled RCCA Withdrawal / Subcritical							
*Uncontrolled RCCA Withdrawal / 100%							
*Uncontrolled RCCA Withdrawal / 100%							
Uncontrolled RCCA Withdrawal / 50%							
*Uncontrolled RCCA Withdrawal / 10%							
*Uncontrolled RCCA Withdrawal / 10%							
Single RCCA Withdrawal							
Statically Misaligned RCCA							
Dropped RCCA							]

* This accident was analyzed with two different reactivity insertion rates.

\# This accident was analyzed with a $F_{\Delta H}$ range of [

].

REFERENCES

1) Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Company, Charlotte, North Carolina, November 1992.

2) BWC Correlation for Critical Heat Flux, BAW-10143P-A, Babcock And Wilcox , Lynchburg, Virginia, April 1985.

3) Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003P-A, Duke Power Company, Charlotte, North Carolina, August 1988.

4) BWCMV Correlation Of Critical Heat Flux In Mixing Vane Grid Fuel Assemblies, BAW-10159P-A, Babcock And Wilcox, Lynchburg, Virginia, February, 1989.

5) McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2004P-A, Duke Power Company, Charlotte, North Carolina, December 1991.

6) Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000P-A, Revision 1, Duke Power Company, Charlotte, North Carolina, December 1991.

Revised Pages For Topical Report DPC-NE-2005P

The bar in the right hand margin notes revised lines.

TABLE 7. SDL Evaluation And Re-Submittal Criteria

The following table lists different events or conditions that would require an evaluation of the applicability of an approved SDL and the subsequent actions based on the results of the analysis.

CONDITION	ACTION
Revised uncertainty larger than the limiting value used in the original analysis.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Revised uncertainty distribution.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
New statepoint.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Minor modifications to the current fuel design.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
A modified CHF correlation.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Change to a new fuel design/fuel type.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
A new CHF correlation.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
Duke analysis of a non-Duke reactor.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
New Thermal-Hydraulic Code.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.

FIGURE 2

REVISED SCD FLOWCHART

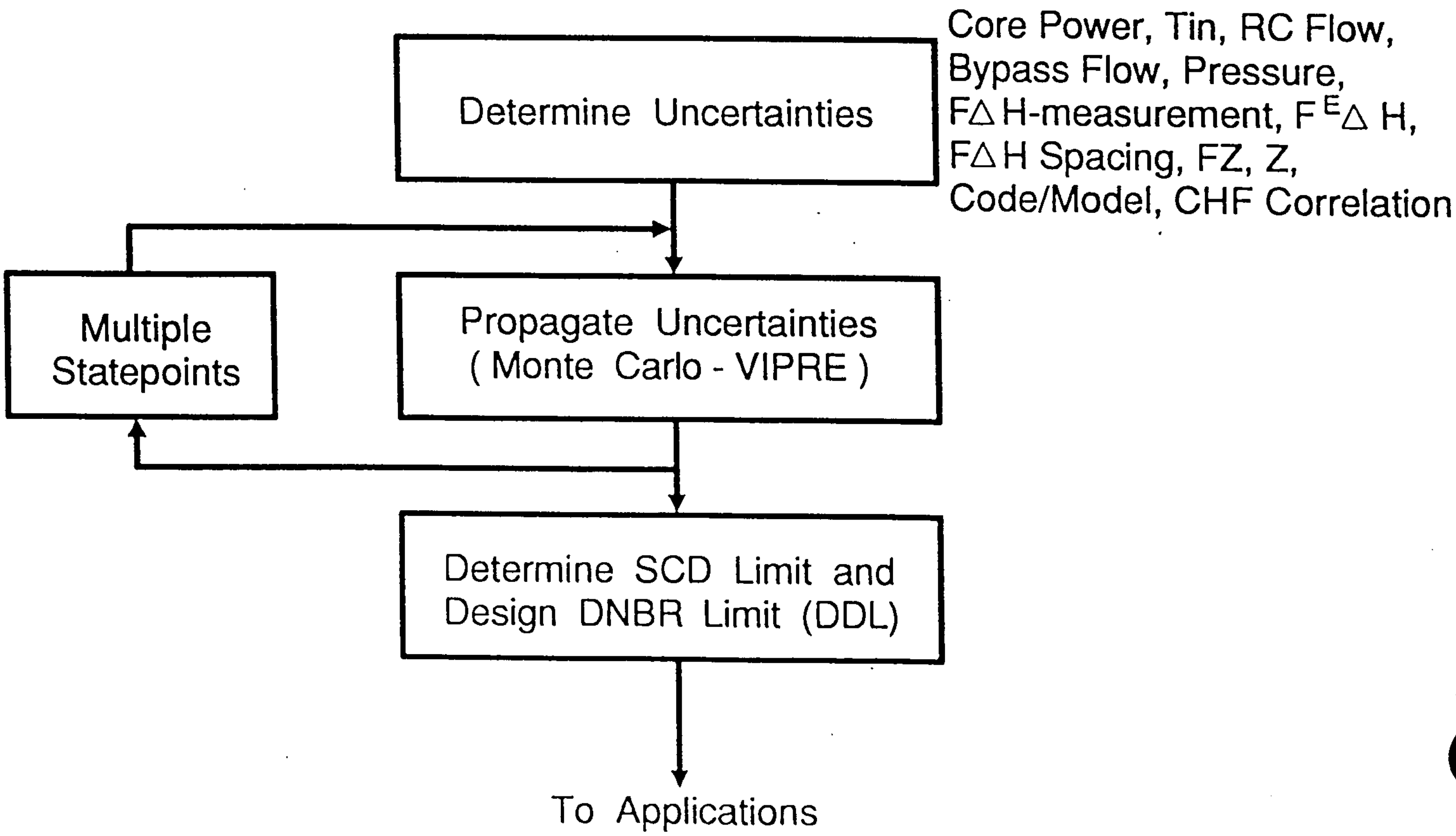

TABLE A-1. Oconee SCD Statepoints

TABLE A-4 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum

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

APPENDIX A

Oconee Plant Specific Data

This Appendix contains the plant specific data and limits for the Oconee Nuclear Station. The thermal hydraulic statistical core design was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the Oconee plant (two loop B&W PWR) with Mark-B fuel assemblies detailed in Reference 2. The BWC critical heat flux correlation described in Reference 9 is used.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal hydraulic computer code (Reference 1) and the Oconee eight channel model approved in Reference 2 are used in this analysis.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table A-1.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table A-2. The range of key parameter values is listed on Table A-4.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table A-3. Section 1 of Table A-3 contains the 500 case runs and Section 2 contains the 3000 case runs. All statepoint SDL values reported in this analysis are normally distributed. The statistical design limit using the BWC CHF correlation for Oconee was determined to be [ ]

Figure A-1 graphically depicts the application [ ]

TABLE A-1. Oconee SCD Statepoints

TABLE A-1 Continued Oconee SCD Statepoints

NOTES:

- 100% design flow is equal to four times 88,000 gpm/pump or 352,000 gpm total system flow.
- 100% Full Power (FP) is equal to 2568 MWth.

(1) Outlet temperature equals 581.0 °F.

(2)

TABLE A-2. Oconee Statistically Treated Uncertainties

Parameter	Type of Uncertainty	Type of Distribution	Uncertainty	Standard Deviation
Reactor System				
Power	Measurement	Normal	± 2.0 %FP	± 1.0 %FP
Temperature	Measurement	Normal	± 2.0 °F	± 1.0 °F
Pressure	Measurement	Normal	± 30.0 psi	± 15.0 psi
Core Flow	Measurement	Normal	± 2.0 %design	± 1.0 %design
Nuclear				
$F\Delta h$	Calculation	Normal	-----------	± 2.84 %
F_z	Calculation	Normal	-----------	± 2.91 %
z	Calculation	Uniform	± 6.0 in.	----------
Fq"	Calculation	Normal	[	
Fq	Calculation	Normal		]
Hot Channel Flow Area	Measurement	Uniform	[ ]	----------
DNBR	Correlation	Normal	-----------	± 8.88 %
DNBR	Code	Normal	[	]

TABLE A-2 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 System Pressure uncertainty.
Core Power	The core power uncertainty was calculated by statistically combining the various random uncertainties associated with the measurement of core power. Since the random uncertainties are normally distributed, the srss of them 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, Fz	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 A-2 Continued Oconee Statistically Treated Uncertainties

Parameter	Justification
Local Heat Flux HCF, Fq"	This uncertainty accounts for the decrease in DNBR at the point of MDNBR due to engineering tolerances. This uncertainty is also increased to account for flux depression at the spacer grids. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
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 ensure the MDNBR channel location is consistent for all cases.
Hot Channel Flow Area	This uncertainty accounts for manufacturing variations in the instrument guide tube sub-channel 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 is normally distributed.
DNBR - Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatism's. This uncertainty also accounts for the small DNB prediction differences between various model sizes. This uncertainty is normally distributed.

TABLE A-3. Oconee Statepoint Statistical Results

500 Case Runs

Statepoint #
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
29
30
31
32
33
34
35
36
37
38
39
40
41

TABLE A-3 Continued Oconee Statepoint Statistical Results

500 Case Runs

Statepoint #
42
43
44
45
46
47
48
49
50
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82

TABLE A-3 continued Oconee Statepoint Statistical Results

3000 Case Runs

Statepoint #

2-T
3-T
6-T
20-T
24-T
26-T
29-T
34-T
39-T
41-T
44-T
53-T
54-T
59-T
62-T
63-T
68-T
72-T
78-T

TABLE A-4 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum

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

FIGURE A-1

APPENDIX B

McGuire/Catawba Plant Specific Data

This Appendix contains the plant specific data and limits for the McGuire and Catawba Nuclear Stations. The thermal hydraulic statistical core design 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 either Mark-BW or Optimized Fuel Assemblies as described in Reference 3. The BWCMV critical heat flux correlation described in Reference 9 is used for analyzing both fuel types.

Thermal Hydraulic Code and Model

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

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table B-1.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table B-2. The range of key parameter values is listed on Table B-4.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table B-3. Section 1 of Table B-3 contains the 500 case runs and Section 2 contains the 3000 case runs. All statepoint SDL values listed in this analysis are normally distributed. The statistical design limit using the BWCMV CHF correlation for McGuire/Catawba was determined to be [ ] Figure B-1 graphically depicts the application [ ]

TABLE B-1. McGuire/Catawba SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	Radial Peak
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						

TABLE B-1 - Continued McGuire/Catawba SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	Radial Peak
24						
25						
26						
27						
28						
29						
30						
31						
32						
33						
34						
35						
36						
37						
38						
39						
40						
41						
42						
43						

TABLE B-2. 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	+/- 3.25% / 1.98%	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	+/- 16.78% / 10.2%	Normal
Code/Model	[ ]	Normal

TABLE B-2 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 B-2 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.
DNBR Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normal.

TABLE B-2 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
DNBR	
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 B-3. McGuire/Catawba Statepoint Statistical Results

500 Case Runs

Statepoint #
1
2
3
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8
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42
43

TABLE B-3 Continued McGuire/Catawba Statepoint Statistical Results

3000 Case Runs

Statepoint #
2-T
3-T
4-T
12-T
13-T
14-T
16-T
20-T
37-T
38-T
39-T

TABLE B-4 McGuire/Catawba Key Parameter Ranges

Parameter	Maximum	Minimum

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

FIGURE B-1

DPC-NE-2005-A

Duke Power Company Thermal-Hydraulic
Statistical Core Design Methodology

APPENDIX C

(Added as Revision 1)

McGuire/Catawba Plant Specific Data

Mark-BW Fuel

BWU-Z CHF Correlation

Submitted April 1996

Approved November 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 7, 1996

RECEIVED
NOV 14 1996
REGULATORY COMPLIANCE

Mr. M. S. Tuckman
Senior Vice President
Nuclear Generation
Duke Power Company
P. O. Box 1006
Charlotte, NC 28201

SUBJECT: SAFETY EVALUATION ON THE USE OF THE BWU-Z CRITICAL HEAT FLUX CORRELATION FOR MCGUIRE NUCLEAR STATION, UNITS 1 AND 2; AND CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M95267, M95268 AND M95333, M95334)

Dear Mr. Tuckman:

By letters dated October 13 and December 4, 1995, as supplemented by letters dated April 26 and September 5, 1996, Duke Power Company requested approval for applying the BWU-Z critical heat flux (CHF) correlation for analyses of the McGuire and Catawba reactor cores with Mark-BW 17x17 type fuel. The BWU-Z CHF correlation for the Mark-BW 17x17 type fuel is one of the three applications stated in Babcock and Wilcox Fuel Company's (BWFC's) (now Framatome Cogema Fuels) Topical Report BAW-10199P, "The BWU CHF Correlations." This topical report was reviewed and approved by the NRC by letter dated April 5, 1996.

Based on its review, the staff finds the proposed application of the BWU-Z CHF correlation for the McGuire and Catawba Mark-BW 17x17 type fuel acceptable. Our safety evaluation, which provides the results of the review, is enclosed.

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370
50-413, and 50-414

Enclosure: Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letters dated October 13, 1995 (Reference 1) and December 4, 1995 (Reference 2), as supplemented by letters dated April 26, 1996 (Reference 3) and September 5, 1996 (Reference 4), Duke Power Company (DPC or the licensee) requested the use of the BWU-Z critical heat flux (CHF) correlation for analyses of the McGuire and Catawba reactor cores, which consist of a full core of Mark-BW 17x17 type fuel assemblies.

2.0 DISCUSSION/EVALUATION

The licensee submitted Appendix C to DPC-NE-2005P-A to support plant-specific applications to the reload analyses for the McGuire and Catawba plants. Specifically, Appendix C contains the plant-specific data and limits with Mark-BW 17x17 type fuel using the BWU-Z form of the BWU CHF correlation, the VIPRE-01 thermal-hydraulic computer code (Reference 6), and Duke Power Company thermal-hydraulic (T-H) statistical core design (SCD) methodology (Reference 7). The licensee stated that the BWU-Z form of the BWU correlation used in the analyses for the McGuire and Catawba units is exactly the same as the correlation used in BAW-10199P (Reference 5).

In addition, the licensee used the approved method as described in Reference 7 regarding the statepoint propagation. In its calculation of the statistical limit, the licensee increased the number of cases from 3,000 to 5,000 per statepoint. The licensee stated that increasing the number of cases provided higher confidence of defining the bounding behavior and reducing the multipliers. The 5,000-case number was selected due to a balance between computer resources required for the calculation and the reduction in statistical uncertainty to determine a conservative Statistical Design Limit (SDL).

The maximum statepoint statistical value for departure from nucleate boiling ratio (DNBR) for the 5,000-case propagation is given in Table C-4 of Reference 3. This table also contains the values where case propagation is

less than the 5,000-case propagation. The 5,000-case value will be used in analyses with the BWU-Z form of the BWU CHF correlation for Mark-BW 17x17 type fuel at McGuire and Catawba.

The statistical design limit given in Table C-4 is applicable to this analysis only when all statepoint parameters fall within the McGuire/Catawba key parameter ranges given in Table C-5 of Reference 3.

DPC has also used the VIPRE-01 thermal-hydraulic computer code (Reference 6) to calculate the measured-to-predicted (M/P) CHF ratios with respect to mass velocity, pressure, or thermodynamic quality. The results show that the average M/P value and the data standard deviation are within 1% of the values reported in BWU CHF correlation (Reference 5).

A comparison between the BWU-Z ranges of applicability for Mark-BW 17x17 type fuel database given in Table 4-1 of Reference 5 and the parameter ranges provided in Table C-1 of Reference 3 shows a 0.01 difference in design limit DNBR using the LYNX and the VIPRE-01 code (1.19 design limit DNBR resulted from the LYNX code versus 1.18 design limit DNBR resulted from VIPRE-01 code). However, DPC will use the larger of the two non-statistical correlation limits.

The staff reviewed the submitals provided by DPC (Reference 1 through Reference 4), and found that the proposed use of BWU-Z CHF correlation is acceptable for use at the McGuire and Catawba plants. This conclusion is based on core analyses that (1) both plants have a full homogeneous core of Mark-BW 17 x 17 type fuel assemblies for upcoming reloads, (2) NRC-approved methodologies (T-H SCD, VIPRE-01, and BWU-Z CHF) are used, (3) the larger of the two correlation limits (VIPRE-01 or LYNX) will be used for non-SCD analyses, and (4) the conservative result from the 5,000-case propagation will be used for SCD analyses.

3.0 CONCLUSION

Based on the above discussions, the staff concludes that the proposed use of BWU-Z critical heat flux correlation for McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2, acceptable.

4.0 REFERENCES

1. Letter from M. S. Tuckman to USNRC requesting review the use of the BWU-Z critical heat flux correlation, dated October 13, 1995.

2. Letter from M. S. Tuckman to USNRC discussing Duke Power Company intent to use of the BWU-Z critical heat flux correlation, dated December 4, 1995.

3. Letter from M. S. Tuckman to USNRC submitting the Appendix to DPC-NE-2005P-A, "McGuire/Catawba Plant Specific Data, Mark-BW Fuel BWU-Z Critical Heat Flux Correlation," dated April 26, 1996.

4. Letter from M. S. Tuckman to USNRC responding to the USNRC's Request for Additional Information regarding Appendix C to DPC-NE-2005P-A, dated September 5, 1996.

5. BAW-10199P, The BWU Critical Heat Flux Correlations, BWFC, November 1994 (Approved by letter from R. C. Jones to J. H. Taylor, dated April 5, 1996).

6. DPC-NE-2004P-A, Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, December 1991.

7. DPC-NE-2005P-A, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, February 1995.

Principal Contributor: T. Huang

Date: November 7, 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 [ ].

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

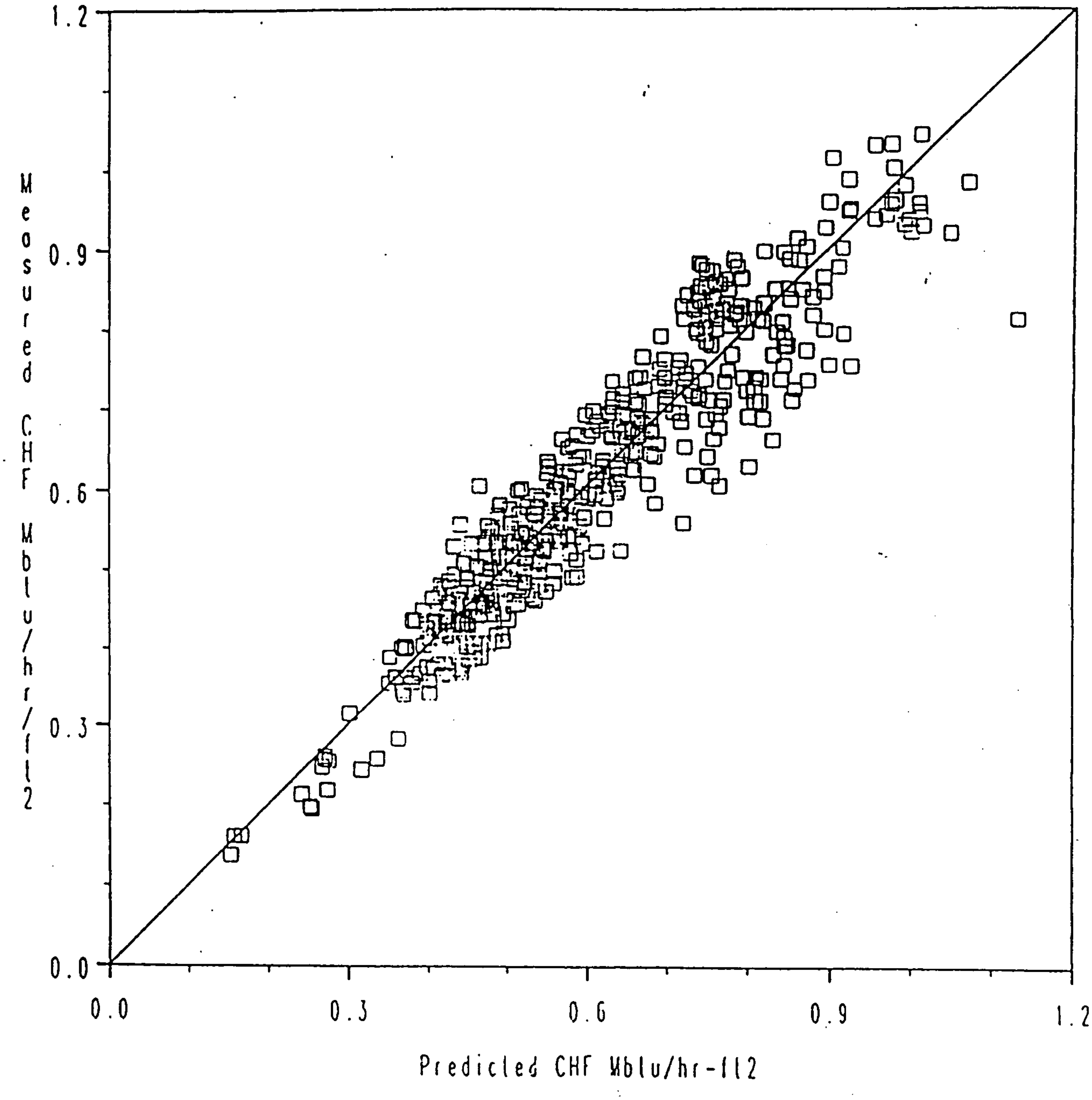

FIGURE C-2

Distribution of CHF Ratios

Mark-BW Data Base

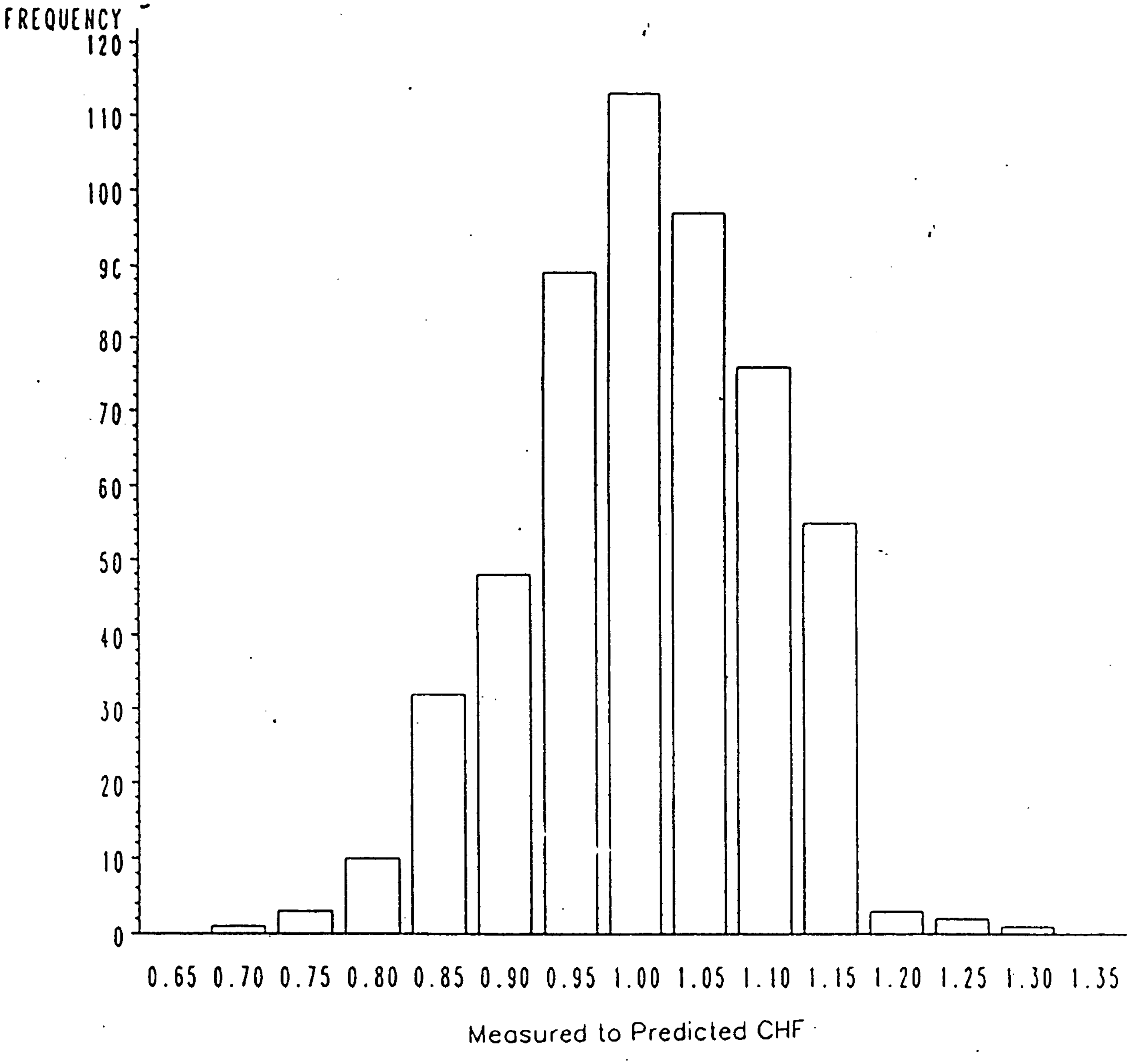

FIGURE C-3

Measured to Predicted CHF Versus Mass Velocity

Mark-BW Data Base

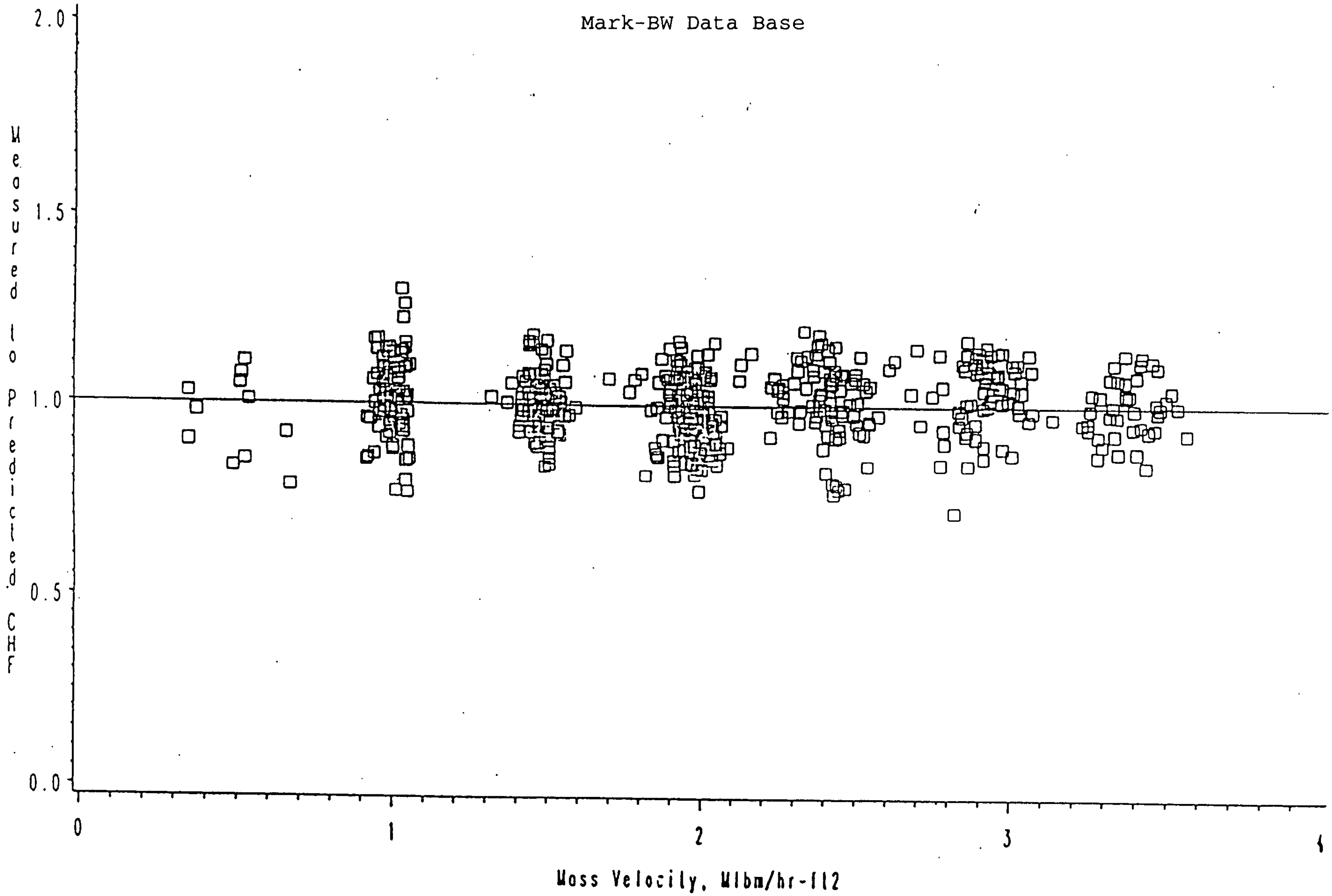

FIGURE C-4

Measured to Predicted CHF Versus Pressure

Mark-BW Data Base

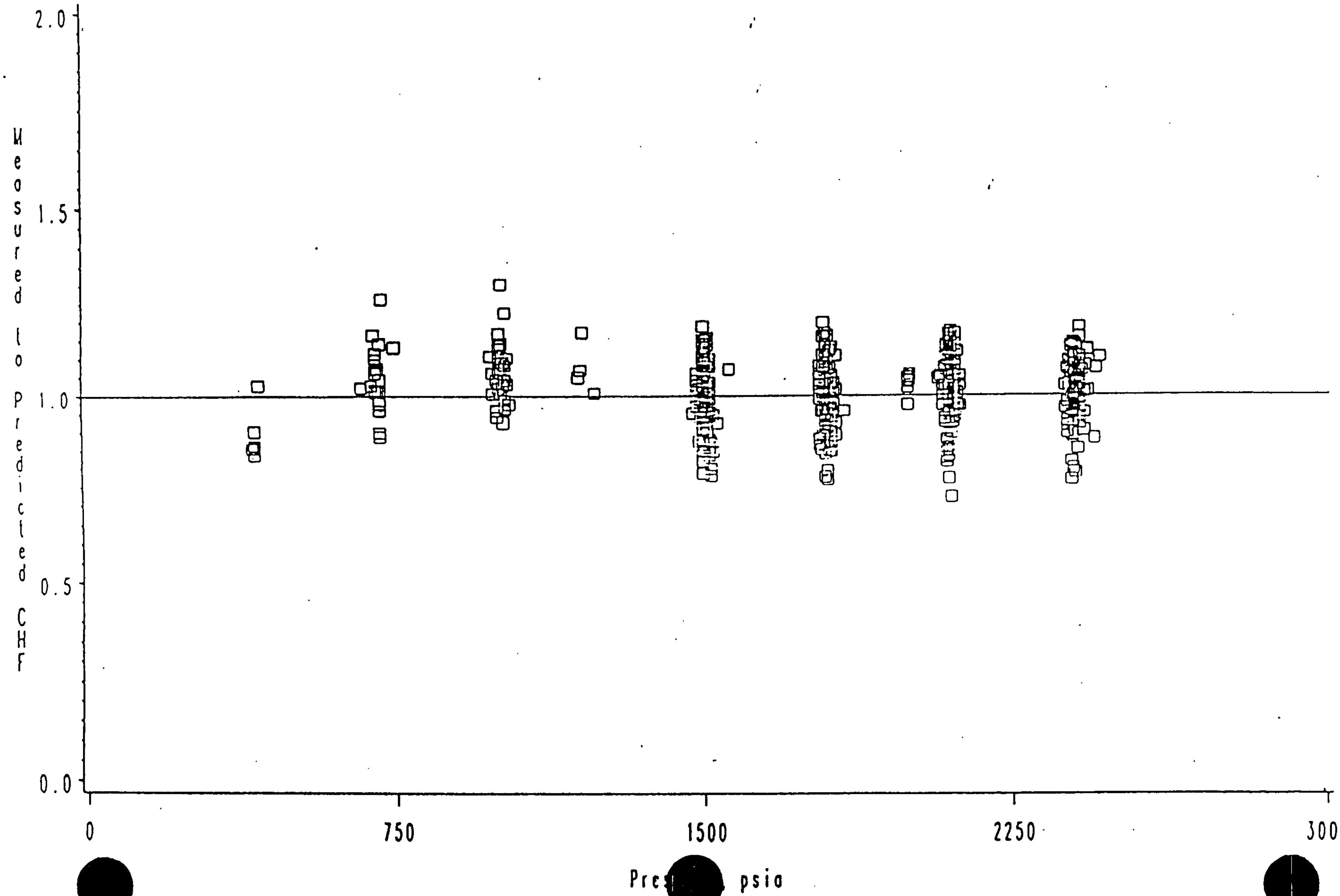

FIGURE C-5

Measured to Predicted CHF Versus Quality

Mark-BW Data Base

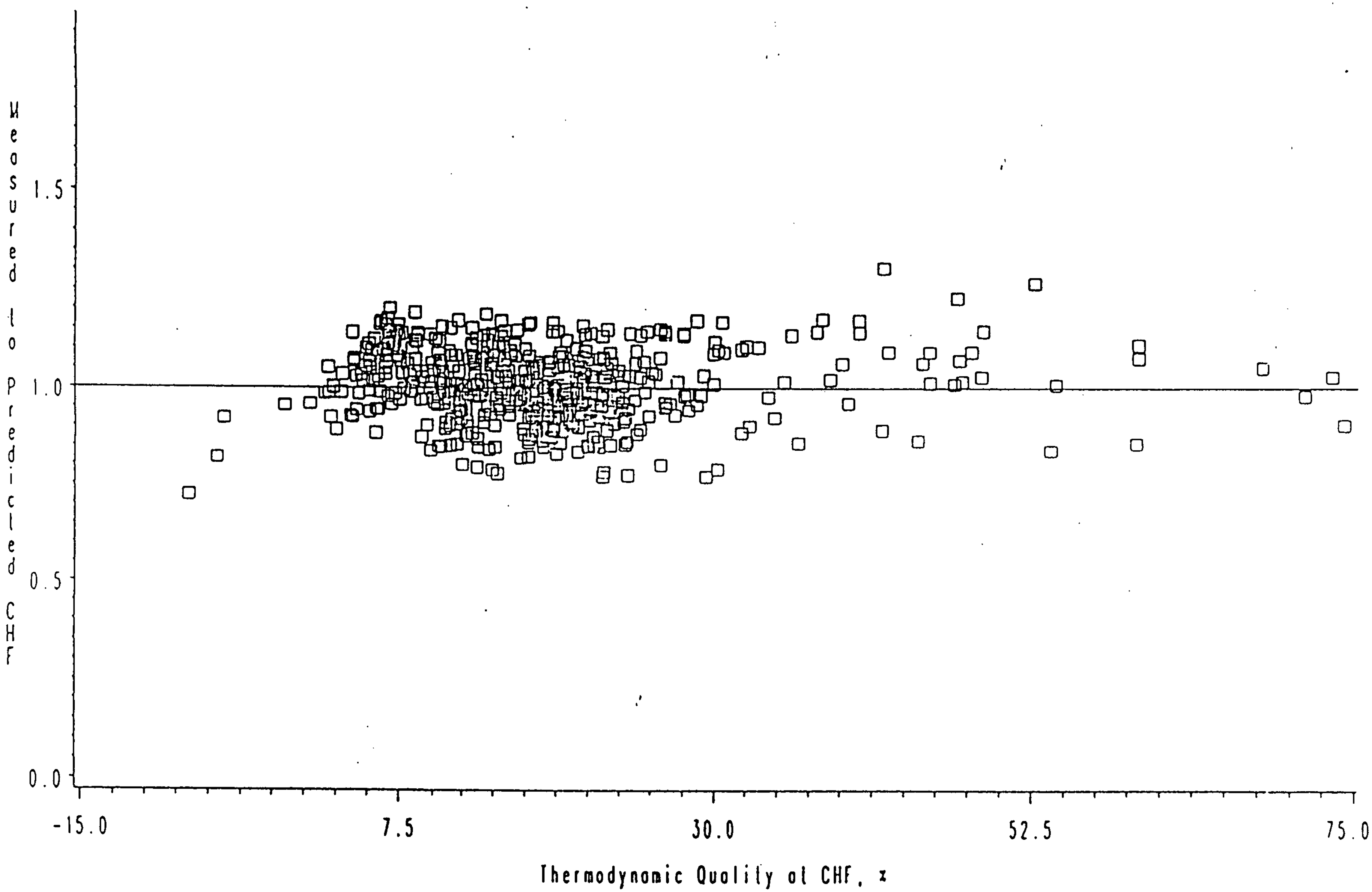

TABLE C-1 VIPRE-01 BWU-Z Correlation Verification

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 @ 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						

* 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				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				

TABLE C-4 Continued McGuire/Catawba Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical 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.

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DUKE POWER

September 5, 1996

U. S. Nuclear Regulatory Commission
Washington, D. D. 20555

Attention: Document Control Desk

Subject: McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nuclear Station
Docket Numbers 50-413 and -414
Use of BWU-Z Correlation by Duke Power;
Supplemental Information

By letter dated April 26, 1996, Duke Power requested NRC approval for use of the BWU-Z correlation at its McGuire and Catawba nuclear stations. A supplement was provided by letter dated December 4, 1995. The December 4, 1996 letter (paragraph 4) stated that the better thermal performance of the fuel can be used to reduce cycle fuel costs. This is due to fact the licensed BWU-Z correlation conservatively quantifies the inherent thermal margin of the Mark-BW17 fuel. This margin can be used in fuel cycle analyses to raise peaking, thereby saving fuel costs. Additionally, the December 4, 1996 letter contained a typographical error in the last sentence of Paragraph 5. The references identified should be 5 and 6, not 6 and 7 as the letter stated.

During telcons on August 21 and 27, 1996, between the NRC staff and Duke, additional information/clarification was requested by the Staff. Attached are the questions and associated responses.

Note that upon approval of the new Appendix C (to topical report DPC-NE-2005), which was transmitted by the April 26, 1996 letter and contains the technical basis for the use of BWU-Z, the topical report will be republished, including the new Appendix C, as DPC-NE-2005, Revision 1.

If there are any questions or additional information is required, please call Scott Gewehr at (704) 382-7581.

M. S. Tuckman

M. S. Tuckman

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S. A. Gewehr
ELL

Request for Additional Information To Topical Report DPC-NE-2005P, Appendix C

The questions are shown in italics and the responses immediately follow.

1) *What fuel type and core configuration are currently operating at McGuire and Catawba?*

McGuire and Catawba are both operating with a full (homogenous) core of Mark-BW17 fuel assemblies, also called Mark-BW 17x17. This will be the fuel type until a transition, beginning in the year 2000, to Westinghouse 17x17 mixing vane fuel. Transition to Westinghouse fuel will require licensing of a different critical heat flux correlation and corresponding statistical design limit applicable to that fuel type.

2) *Table C-1 lists the statistical results of the CHF test data base analysis with the VIPRE-01 thermal-hydraulic computer code. Explain the differences in values between this table and the table in BAW-10199P-A which documents the same data analysis with the LYNXT or LYNX 2 code.*

The information provided for the Mark-BW17 data base using the BWU-Z correlation on the top of page 4-3 in BAW-10199P-A shows the average M/P, Standard Deviation (corrected for N), and Design Limit DNBR (denoted DNBR(L)) for the test data when analyzed with LYNXT or LYNX 2. Table C-1 of the DPC-NE-2005 Appendix C is a direct comparison of the same analysis and the same test data with VIPRE-01 code. The VIPRE-01 code has a slightly higher average M/P and slightly lower standard deviation for the entire test data base when compared to LYNXT or LYNX 2. The combination of these two parameters gives the VIPRE-01 code a slightly lower Design Limit DNBR for the test data base.

The more conservative value for the parameter is selected by Duke Power Company (DPC) when performing an analyses. For example, the standard deviation listed in Table C-3 (DPC-NE-2005 Appendix C) for the correlation uncertainty is the higher of the LYNX and VIPRE-01 values rounded to two significant figures. The Design Limit DNBR calculated with VIPRE-01 is presented in Table C-1 for comparison only. The standard deviation is the only value that impacts the SCD calculation. If the BWU-Z form of the BWU correlation is used by DPC in non-SCD analyses, the larger of the two non-statistical correlation limits (the LYNX value listed on page 4-3 of BAW-10199P-A) will be used.

3) *Explain the method used to calculated the 500 and 5000 case statistical DNBR values for each statepoint and how the statistical limit is used.*

The method used to evaluated the BWU-Z form of the BWU correlation in Appendix C is identical to the procedure outlined in the main body of the DPC-NE-2005 report. This procedure is outlined starting in Section 2.0 on page 5. The key parameters listed in Section 2.1, page 6, are identical in Appendix C. The statepoints in Table C-2 of Appendix C were selected to bound the range of key parameters where the SCD analyses with BWU-Z will be applied.

The selection of uncertainties is discussed in Section 2.2, page 7, of DPC-NE-2005. Table C-3 lists the values used in the BWU-Z SCD analyses. These are identical to the values used in the BWCMV analysis (Appendix B of DPC-NE-2005) except for the correlation standard deviation (as explained in Question 2 above) and the FΔH measurement uncertainty which was increased slightly for the BWU-Z analysis.

The method for statepoint propagation is explained in Section 2.3, page 8, of DPC-NE-2005. The calculation of the statepoint statistical limit is explained in Section 2.4, page 9 through 11, of DPC-NE-2005. The equation for the SDL calculation is shown on page 10. Included in the equation are Chi Square and K factor multipliers to ensure a conservative limit based on the number of cases calculated. The mean and standard deviation values for a statepoint fluctuate slightly as the number of cases increase. Increasing the number of cases gives higher confidence that the data analyzed defines bounding behavior, therefore the multipliers are reduced. This ensures the SDL limit is equally conservative even though the final statistical DNBR value is smaller as the number of cases gets larger. An example of the way the values change with an increasing number of cases is shown on Table 1.

The main body of the report lists the number of cases as either 500 or 3000 per statepoint. The propagation method is identical regardless of the number of cases generated. In the response to Question 8 of Attachment II, Request For Additional Information, in DPC-NE-2005, DPC stated that the number of cases may be increased. This number was increased to 5000 for the BWU-Z analysis in Appendix C. As explained in the response to Question 8, this increase is consistent with the methodology and does not in any way reduce the conservatism of the SDL limit calculated.

The 5000 case number was selected as a balance between computer resources required for the calculations and the reduction in statistical uncertainty. For example, increasing the number of cases by two thirds from 3,000 to 5,000 reduces the K factor by 0.011 (from 1.692 at 3,000 to 1.681 at 5,000). Further increasing the number of cases to 10,000 would require another doubling of resources for the same K factor reduction (from 1.681 at 5000 to 1.670 at 10,000).

The 500 and 5000 case results for the BWU-Z analysis are listed in Table C-4 of Appendix C. As described in DPC-NE-2005, the 5000 case statepoints are selected based on the results of the 500 case statepoint propagations. The 5000 case runs are used to determine a conservative Statistical Design Limit (SDL) for the correlation SCD analyses. A value larger than the largest 5000 case statepoint statistical DNBR value is listed on page C-4. This is the statistical design limit that will be used in analyses with the BWU-Z form of the BWU correlation for Mark-BW fuel at McGuire and Catawba. The statistical design limit listed on page C-4 will be applicable to an analysis as long as all statepoint parameters fall between the Maximum and Minimum ranges listed on Table C-5.

TABLE 1

Statepoint 1 Values

Number Of Cases	Coefficient Of Variation*	Chi Square Multiplier	K Factor Multiplier	Statistical DNBR**
500	0.1514	1.05549	1.763	1.392
1000	0.1541	1.03848	1.727	1.382
1500	0.1528	1.03115	1.712	1.369
2000	0.1537	1.02684	1.703	1.368
2500	0.1534	1.02393	1.698	1.364
3000	0.1539	1.02179	1.692	1.363
3500	0.1538	1.02013	1.689	1.362
4000	0.1543	1.01880	1.686	1.361
4500	0.1540	1.01771	1.683	1.358
5000	0.1539	1.01681	1.681	1.357

* **Coefficient of Variation = Standard Deviation / Mean for the number of cases**

$$\textbf{**Statistical DNBR} = \frac{1}{[\,1 - \{(\text{Coeff. of Variation}) * (\text{Chi Square Mult.}) * (\text{K Factor Mult.})\}]}$$