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Westinghouse Thermal Design Procedure (WTDP)

Westinghouse

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Revision 0

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ACRONYMS

Acronym	**Description**
95/95	95% probability at the 95% confidence level
AO	Axial Offset
AOO	Anticipated Operational Occurrence
ASI	Axial Shape Index (-AO)
CE	Combustion Engineering
CE16NGF	Combustion Engineering 16x16 Next Generation Fuel
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
COLSS	Core Operating Limit System Setting
CPC	Core Protection Calculator
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
DNBR	DNB Ratio
$F^{E}_{\Delta H}$	Engineering Enthalpy-Rise Hot Channel Factor
$F^{N}_{\Delta H}$	Nuclear Enthalpy-Rise Hot Channel Factor
FSAR	Final Safety Analysis Report
GDC	General Design Criterion
IFM	Intermediate Flow Mixer
IOSGADV	Inadvertent Opening of Steam Generator Atmospheric Dump Valve
LCO	Limiting Condition for Operation
LOP	Loss of Offsite Power
LHR	Linear Heat Rate
LOCA	Loss of Coolant Accident
LSSS	Limiting Safety System Settings
M/P	Measured-to-Predicted Ratio
MSCU	Modified Statistical Combination of Uncertainties
NGF	Next Generation Fuel
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
P/M	Predicted-to-Measured Ratio
PDF	Probability Density Function
PVNGS	Palo Verde Nuclear Generation Station
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RG	Regulatory Guide
R_N	Random Number for Normal Distribution
R_U	Random Number for Uniform Distribution
RTDP	Revised Thermal Design Procedure
SAFDL	Specified Acceptable Fuel Design Limit
SAR	Safety Analysis Report
SCU	Statistical Combination of Uncertainties
SER	Safety Evaluation Report
SONGS	San Onofre Nuclear Generating Station

SRP	Standard Review Plan
STDP	Standard Thermal Design Procedure
UFSAR	Updated FSAR
UTL	Upper Tolerance Limit
V-5	VANTAGE 5 (fuel design)
VVER	Water-Water Energetic Reactor (Russian designed PWR)
WTDP	Westinghouse Thermal Design Procedure
μ	Mean Value
σ	Standard Deviation
[	]^{a,c}
[	]^{a,c}

1 INTRODUCTION AND REGULATORY REVIEWS

General Design Criterion (GDC) 10, "Reactor Design," in Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) Appendix A, "General Design Criteria for Nuclear Power Plants," (Reference 1), requires the reactor core to include appropriate margin to assure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during normal operation or Anticipated Operational Occurrences (AOOs), which are referred to as Condition I and II events. For a Pressurized Water Reactor (PWR), one of the SAFDLs is to prevent overheating of any fuel rod in the reactor core due to reaching Departure from Nucleate Boiling (DNB). Margin to DNB is quantified through the DNB ratio (DNBR), which is defined as a ratio of predicted heat flux from a DNB correlation to local heat flux on the fuel cladding surface. A DNB correlation is also referred to as a Critical Heat Flux (CHF) correlation. As specified in Section 4.4, "Thermal and Hydraulic Design," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (Reference 2), one of the acceptance criteria for the DNBR SAFDL is to ensure that there is a 95-percent probability at the 95-percent confidence level (95/95) that the hot fuel rod in the PWR core does not experience DNB during Condition I and II events. The DNB design criterion is also conservatively applied in some non-LOCA (non-Loss of Coolant Accident) Condition III and IV accident analyses, in order to estimate the number of failed fuel rods.

Over the years, Westinghouse has developed and applied several methods for statistical combination of the uncertainties to obtain 95/95 DNBR limits that met the acceptance criterion. The methods primarily used in current applications are the Revised Thermal Design Procedure (RTDP) (Reference 3) and the Statistical Combination of Uncertainties (SCU) (References 4 and 5). RTDP has been used for Westinghouse Nuclear Steam Supply System (Westinghouse-NSSS) plant applications, and SCU/Modified SCU (MSCU) (Reference 6) has been applied to the Combustion Engineering NSSS (CE-NSSS) plants with digital reactor protection systems.

The Westinghouse Thermal Design Procedure (WTDP) discussed in this report consolidates the existing methods of calculations of the statistical DNBR limit for Condition I and II events and statistical rods-in-DNB convolution for non-LOCA Condition III and IV events for PWR design applications. WTDP integrates the design process based on the existing SCU and RTDP methods. It is designed to implement a Monte Carlo approach that performs subchannel thermal hydraulic calculations to statistically combine uncertainties in a DNB correlation, computer codes, fuel and modeling parameters (also referred to as system parameters), and reactor parameters (also referred to as state parameters) to obtain the 95/95 DNBR limit. The number of fuel rods in DNB for the radiological dose evaluation of a Condition III or IV event is determined with additional inputs of the fuel census that relates fuel rod power versus number of fuel rods in the reactor core with fuel rod power versus DNBR and DNB probability distribution. A description of the DNBR limit calculation using the WTDP method is provided in Chapter 2. The method for calculating rods-in-DNB is described in Chapter 3. The WTDP intended applications are described in Chapter 4. Conditions for the WTDP applications are summarized in Chapter 5. Demonstrative calculations for different PWR designs are shown in the attachments to this report.

1.1 REVIEW OF REGULATORY REQUIREMENTS

Section 50.34 of Title 10 of the Code of Federal Regulations (10 CFR), "Contents of construction permit and operating license applications; technical information," contains general requirements for the safety assessment of structures, systems, and components important to safety. As part of the core reload design process, licensees are responsible for reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses remain bounding, licensees confirm those key inputs to the safety analyses (such as DNBR) are conservative with respect to the current design cycle. If key safety analysis parameters are not bounded, a re-analysis or a re-evaluation of the affected transients and/or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

Regulatory guidance for the review of thermal-hydraulic design methods with respect to the applicable General Design Criteria (GDC) is provided in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP), Section 4.4, “Thermal and Hydraulic Design.” SRP 4.4, Revision 2, Acceptance Criterion II.1 is based on meeting the relevant requirements of the following Commission regulation:

> GDC 10, as it relates to whether the design of the reactor core includes appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or anticipated operational occurrences (AOOs).

SRP 4.4 Acceptance Criterion II.2, which invokes GDC 12, as it relates to whether the design of the reactor core and associated coolant, control, and protection systems assures that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed, is not applicable to the content of WCAP-18240-P.

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs. WCAP-18240-P satisfies this requirement by specifying the departure from nucleate boiling (DNB) design basis which corresponds to a 95% probability at the 95% confidence level (the 95/95 DNB criterion) that DNB will not occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Fuel rod overheating due to DNB could result in cladding failure and an uncontrolled release of radioactive material to the reactor coolant system (RCS). Proper thermal-hydraulic design of the reactor core and associated systems is necessary to assure that sufficient margin exists with regard to maintaining adequate heat transfer from the fuel to the RCS. Compliance with GDC 10 within the tenets of WCAP-18240-P provides assurance that the integrity of the fuel and cladding will be maintained, thus preventing the potential for release of fission products during normal operation or AOOs.

1.2 REVIEW OF EXISTING APPROVALS

The methods used for the DNBR Limit Calculation and for the Rods-in-DNB Calculation, as discussed in WCAP-18240-P Sections 2.0 and 3.0, were previously approved in other topical reports or plant licensing submittals. Further background on those prior methods approvals is provided below:

WTDP Reference 3: “Revised Thermal Design Procedure,” WCAP-11397-P-A, Westinghouse Electric Corporation, February 1989.

This methodology was reviewed and approved by the NRC on statistical combination of uncertainties in a DNB correlation, fuel and reactor design parameters, and computer codes for a DNBR limit in compliance with the 95/95 acceptance criterion for Westinghouse-NSSS plant applications. Approved versions (“-A”) of the proprietary and non-proprietary reports were submitted to the NRC on April 5, 1989 (ADAMS Accession Number 8904250027)

WTDP Reference 4: “Statistical Combination of Uncertainties Part 1; Combination of System Parameter Uncertainties in Thermal Margin Analyses for San Onofre Nuclear Units 2 and 3,” CEN-283(S)-P, Revision 0, ABB Combustion Engineering, June 1984.

Part 1 of CEN-283(S)-P describes the statistical combination of system parameter uncertainties in thermal margin analyses for the San Onofre plant. A detailed description of the uncertainty probability distributions and response surface techniques for the 95/95 DNBR limit determination was provided in the report. Similar methods were previously reviewed and approved by the NRC for other CE-NSSS plants including:

Calvert Cliffs ("Statistical Combination of Uncertainties," CEN-124(B)-P, Part 1, December 1979, Part 2, January 1980, Part 3, March 1980);

St. Lucie-1 ("Statistical Combination of Uncertainties," CEN-123(F)-P, Part 1, December 1979, Part 2, January 1980, Part 3, February 1980);

ANO-2 ("Statistical Combination of Uncertainties," CEN-139(A)-P, November1980);

System 80 ("Statistical Combination of Uncertainties," Enclosure 1-P to LD-82-054), and

Fort Calhoun ("Statistical Combination of Uncertainties", CEN-257(0)-P, Part 2, November 1983).

The San Onofre Cycle 2 analysis covered by CEN-283(S)-P, Revision 0 (June 1984), included updates first imposed by the NRC at Arkansas Nuclear One, Unit 2, as documented in the “Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 24 to Facility Operating License No. NPF-6, Arkansas Power and Light Company, Arkansas Nuclear One, Unit No. 2,” Docket No. 50-368, dated June 19, 1981 (ADAMS Accession Number 8106260493).

The operation of San Onofre Units 2 and 3 during Cycle 2 was subsequently approved by the NRC, with updates, in “Issuance of Amendment No. 32 to Facility Operating License NPF-10 and Amendment No. 21 to Facility Operating License NPF-15, San Onofre Nuclear Generating Station, Units 2 and 3,” dated March 1, 1985 (ADAMS Accession Number ML022280336).

WTDP Reference 5: "Statistical Combination of Uncertainties Part 2; Uncertainty Analysis of Limiting Safety System Settings San Onofre Nuclear Generating Station Units 2 and 3," CEN-283(S)-P Revision 0, ABB Combustion Engineering, October 1984.

Part 2 of CEN-283(S)-P describes the methodology used for statistically combining uncertainties involved in the determination of the Linear Heat Rate (LHR) and DNBR Limiting Safety System Settings (LSSS) for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3, and for CE-NSSS System 80 plants. It describes statistical combination of state parameter and modeling uncertainties for the determination of the LSSS overall uncertainty factors related to the CETOP-D code applications.

WTDP Reference 6: “Modified Statistical Combination of Uncertainties,” CEN-356(V)-P-A, Rev.01-P-A, ABB Combustion Engineering, May 1988.

This methodology was review and approved by the NRC in “Issuance of Amendment No. 24 to Facility Operating License No. NPF-41 for the Palo Verde Nuclear Generating Station, Unit No. 1, TAC Nos. 65460, 65461, 65462 and 65691 through 65706,” dated October 21, 1987 (ADAMS Accession Number ML021690079). The report describes a methodology change to statistically combine uncertainty components from two groups of system parameters and state parameters to obtain overall uncertainty factors in determining the limiting safety system setting (LSSS) and limiting condition for operation (LCO) for the Palo Verde Nuclear Generation Station (PVNGS) COLSS and CPC system. The overall uncertainty factors could be calculated and applied as a function of burnup, axial shape index (ASI), and power in COLSS and CPC. This methodology has been referenced and used for existing CE-NSSS safety analyses and reload evaluations.

WTDP Reference 9: “Loss of Flow C-E Methods for Loss of Flow Analysis,” CENPD-183-A, ABB Combustion Engineering, June 1984.

This methodology was reviewed and approved by the NRC in “Acceptance for Referencing of Licensing Topical Report CENPD-183,” dated May 12, 1982, including “Topical Report Evaluation CENPD-183, Loss of Flow,” dated March 30, 1982 (ADAMS Accession Number ML16224A358). The report describes the statistical convolution technique for fuel rod failure calculations. The Staff concluded that the statistical convolution technique is acceptable for fuel rod failure calculations. Any application of a new fuel damage probability distribution using a different computer code or a DNB correlation is required for approval by the Staff.

WTDP Reference 10: Palo Verde Nuclear Generation Station Units 1, 2 and 3 Updated Final Safety Analysis Report (FSAR), Revision 19, June 2017.

The plant FSAR changes were reviewed and approved by the NRC. Section 15.4.8 of the UFSAR, Control Element Assembly Ejection, describes current application of the statistical rods-in-DNB evaluation method from CENPD-183-A to a Condition IV non-LOCA event.

WTDP Reference 14: M. A. Book and W. L. Greene, “Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF),” WCAP-16500-P-A Supplement 1 Revision 1, December 2010.

This methodology was reviewed and approved by the NRC in “Final Safety Evaluation for Westinghouse Electric Company Topical Report WCAP-16500-P, Supplement 1, Revision 1, ‘Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)’ (TAC No. ME0143),” dated December 28, 2009 (ADAMS Accession Number ML093280716) and “Final Safety Evaluation for Westinghouse Electric Company Addendum 1 to Topical Report WCAP-16500-P, Supplement 1, Revision 1, ‘Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)’ (TAC No. ME3583),” dated July 1, 2010 (ADAMS Accession Numbers ML101720183 and ML101720184). The report describes application of the CE-NSSS setpoint methodology including the MSCU process to the CE 16x16 Next Generation Fuel (NGF) reload evaluations.

1.3 REVIEW OF SRP ON DNBR LIMIT CALCULATION

The SRP 4.4 acceptance criteria meet the requirements of GDC 10 and are relevant to the evaluation of fuel design limits described in WCAP-18240-P. Assurance must be provided that there is at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience DNB during normal operation or AOOs. Previously approved thermal-hydraulic subchannel codes and DNB correlations will be used – WCAP-18240-P makes no changes in those areas.

Uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculational methods used in the WTDP assessments will be treated with at least a 95-percent probability at the 95-percent confidence level. The assessment of thermal margin also considers the uncertainties in instrumentation. The origin of each uncertainty parameter, such as fabrication uncertainty, computational uncertainty, or measurement uncertainty (e.g., reactor power, coolant temperature, flow), is identified or referenced for each application. Distribution of each parameter uncertainty has been previously justified for statistical combination, and the method used to combine uncertainties is described in WCAP-18240-P.

For the WTDP DNBR limit calculations, the NRC-approved Statistical Combination of Uncertainties (SCU) method for Combustion Engineering Nuclear Steam Supply System (CE-NSSS) plants is integrated with the Revised Thermal Design Procedure (RTDP) method for Westinghouse-NSSS plants. WTDP is designed to implement a Monte Carlo approach that statistically combines uncertainties in a DNB correlation, fuel and modeling parameters (also referred to as system parameters), and reactor parameters (also referred to as state parameters) to determine the DNBR limit. The major improvement in the WTDP calculation method over the SCU method [ ]^{a,c} for combining uncertainties in the system parameters with the uncertainty in the DNB correlation. As compared to RTDP, WTDP replaces [ ]^{a,c} for the limit calculation with the Monte Carlo approach. The 95/95 DNBR limit from WTDP is [ ]^{a,c} to that obtained from RTDP for a Westinghouse-NSSS plant as shown in Attachment A, "Sample calculation of 95/95 DNBR Limit for Westinghouse-NSSS plant design," Table A-4. The WTDP DNBR limit can be [ ]^{a,c} the limit from SCU for a CE-NSSS plant, as shown in Attachment B, "Sample Calculation of 95/95 DNBR Limit for CE-NSSS Design," Table B-2.

1.4 REVIEW OF SRP ON RODS-IN-DNB CALCULATION

For non-LOCA Condition III or IV radiological consequence analysis using WTDP, the amount of fuel failure is determined based on an NRC-approved statistical convolution method of calculating the number of fuel rods in DNB for CE-NSSS plants. Any fuel rod which experiences a calculated heat flux value reaching DNB during the event is conservatively assumed to fail for the radiological consequence evaluation. Since the DNBR limit as a SAFDL is defined on a 95/95 basis, there is only a 5% probability with the 95% confidence level that DNB would occur if a fuel rod DNBR is at the limit. The same method for CE-NSSS plants can be applied to the rods-in-DNB evaluations for Westinghouse-NSSS Non-LOCA Condition III or IV events, including locked rotor and control rod ejection accidents.

SRP 15.3.3 – 15.3.4 Revision 3 acceptance criteria for locked rotor accident analysis relevant to the scope of WCAP-18240-P are discussed in Subsection II.2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 limit for PWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below the limit, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. WCAP-18240-P requires that fuel rods experiencing DNB are assumed to fail for the purposes of radiological evaluations, but also considers probability of a fuel rod reaching DNB corresponding to its DNBR value.

SRP 15.4.8 Revision 3 acceptance criteria for rod ejection accident analysis relevant to the scope of WCAP-18240-P are discussed in Subsections II.2 and III.2.A. The number of fuel rods with clad failure must be determined from an acceptable procedure for calculating a DNB condition during the reactivity excursion. This determination may be done by reference to previous cases for the same nuclear steam supply system vendor. DNB must be calculated in accordance with the criteria reviewed and accepted under SRP Section 4.4. The rods-in-DNB calculation method described in WCAP-18240-P is based on a DNB correlation and its DNBR limit typically described in a plant Safety Analysis Report (SAR) Section 4.4.

NRC Regulatory Guide (RG) 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” dated May 2003, discusses acceptable assumptions related to radiological consequence evaluations. Section 3.6 of RG 1.195, “Fuel Damage in Non-LOCA DBAs,” states that the amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. The NRC staff has traditionally relied upon DNBR as a fuel damage criterion for estimating fuel damage for the purpose of establishing radioactivity releases. This criterion is also applied in WCAP-18240-P.

1.5 CONCLUSION OF REVIEWS

WCAP-18240-P consolidates existing NRC-approved methods to be applied to all PWRs with respect to analyses involving DNBR limits for Condition I and II events and statistical rods-in-DNB evaluations for non-LOCA Condition III and IV events. This single topical report consolidation facilitates analysis work and review activities, while providing an improved ability to accurately quantify analysis margins.

2 METHOD FOR DNBR LIMIT CALCULATION

The WTDP method for calculating the 95/95 DNBR limit is based on the existing SCU/MSCU method in References 4 through 6 and the existing input to RTDP (Reference 3) and SCU calculations. It combines uncertainties in reactor core and fuel parameters to obtain overall uncertainty factors for a DNBR design limit at a 95/95 basis. DNBR calculations are performed using a thermal-hydraulic subchannel code and a DNB correlation already approved for plant safety analysis and licensing applications.

2.1 INPUT TO CALCULATION

The WTDP DNBR limit calculation accepts input of uncertainties in system parameters and state parameters. The system parameters are related to a PWR fuel design. The state parameters are related to the reactor design. The uncertainty input is plant specific and typically consists of range and descriptive statistics such as mean, standard deviation, and distribution type (normal or uniform).

Uncertainties in the system (fuel-related) parameters include:

- Engineering enthalpy rise hot channel factor
- Engineering heat flux factor
- DNB correlation
- Subchannel computer code and modeling

[ ]a,c

- Systematic fuel rod pitch
- Systematic fuel rod clad outside diameter.

Uncertainties in the state (reactor-related) parameters for the DNBR limit calculation include:

- Reactor power
- Reactor power distribution and radial peaking factor
- Reactor coolant temperature
- Reactor coolant flow rate
- Reactor core bypass flow fraction
- Reactor pressure.

There is [ ]a,c when WTDP is applied as an alternative to either RTDP for Westinghouse-NSSS plant designs or SCU for CE-NSSS digital plant designs. As demonstrated in the attached sample calculations, the uncertainty input is justified on a plant specific basis. The statistical 95/95 DNBR limits for some plants may not include all the uncertainties in the system and state parameters listed above. If its uncertainty is not included in the DNBR limit, the parameter value is selected to be conservative for the DNBR calculation.

The uncertainty in the DNB correlation is input through the correlation statistics consisting of the mean and standard deviation of the measured-to-predicted (M/P) critical heat flux (CHF) ratio which is the reciprocal of DNBR. In order to preserve the approved correlation DNBR limit (Reference 7), the correlation input can be adjusted as follows:

[$]^{a,c}$

(2-1)

(2-2)

The approved correlation DNBR limit can be obtained from the Safety Evaluation Report (SER) on the correlation topical report.

2.2 CALCULATION PROCESS

The WTDP DNBR limit calculation consists of two sub-cases of sampling to obtain ΔDNBR. The first sub-case is based on input of reactor design conditions [$]^{a,c}$ to define the design space of the plant. DNBR calculations are performed at [$]^{a,c}$ design conditions from the design space [$]^{a,c}$ The second sub-case is DNBR calculations at the sampled condition of the first sub-case but perturbs the system and state parameters within their uncertainty ranges and distributions. In the second sub-case calculation, a parameter value is obtained from sampling [$]^{a,c}$ The parameter sampling is further described in Section 2.2.1.

The ΔDNBR from the two sub-cases is then combined with a sampled DNBR [$]^{a,c}$ It is sufficient to collect sampled conditions greater than [$]^{a,c}$ cases to generate a DNBR distribution. The sampled DNBR is described in Section 2.2.2.

A D-Prime normality (D') test (Reference 8) of the resultant DNBR distribution is performed. Based on the result, either normal or non-parametric statistics are used to derive a raw DNBR limit. If the distribution has been determined to be a normal distribution at a significance level of 5%, the raw DNBR limit value is further adjusted by using the [$]^{a,c}$ to account for finite sampling to obtain the 95/95 DNBR limit. Otherwise, the non-parametric 95/95 locator for the DNBR order statistics is applied to obtain the 95/95 DNBR limit. The DNBR limit for a plant design application can be increased to account for an additional penalty or margin requirement deterministically, such as to incorporate a rod bow DNBR penalty. The DNBR limit determination is described in Section 2.2.3.

2.2.1 Parameter Sampling

The method for parameter sampling based on a uniform distribution is shown in Equation 2-3. For example, uniform sampling is performed for the first sub-case in the design space defined [
]^{a,c}

[ ]^{a,c} (2-3)

The method for parameter sampling based on a normal distribution is shown in Equation 2-4. [
]^{a,c}

[ ]^{a,c}

2.2.2 Sampled DNBR

The ΔDNBR from the two sub-cases is combined with a sampled DNBR [
]^{a,c} to determine a DNBR containing the delta change for that particular sample "i". The ΔDNBR value for sample "i" is calculated as follows:

[ ]^{a,c}

[ ]a,c

The sampled DNBR calculation using Equations 2-5 through 2-8 is performed for at least [ ]a,c cases to obtain a DNBR distribution.

2.2.3 DNBR Limit Determination

The DNBR limit can be calculated in two ways as discussed below, depending on the normality test of its distribution.

2.2.3.1 95/95 Limit Based on Normal Distribution

When the DNBR distribution has been determined to be a normal distribution, the following two equivalent methods can be used to calculate the 95/95 DNBR limit. Both methods use the mean (μ) and standard deviation (σ) of the DNBR distribution and both must account for the finite number of samples in the DNBR distribution:

[ ]a,c

[ ]^{a,c} (2-9)

(2-10)

(2-11)

(2-12)

2. An appropriate Owen's factor, $k_{95/95}$, for the sample size is calculated and is used with the DNBR distribution mean and standard deviation values.

A 95/95 DNBR limit is calculated using Method 1:

$$DNBR_{95/95} = \mu_{95/95} + 1.645 * \sigma_{95/95} \tag{2-13}$$

Or by Method 2:

$$DNBR_{95/95} = \mu_{DNBR\ Distribution} + k_{95/95} * \sigma_{DNBR\ Distribution} \tag{2-14}$$

Both of the calculated 95/95 DNBR limit values can be further adjusted to incorporate additional DNBR margin. For example, additional margin can be incorporated into the DNBR limit to account for the fuel rod bow penalty as follows:

[ ]^{a,c} (2-15)

Any other deterministic adjustments can be made in a similar manner.

2.2.3.2 Distribution-Free 95/95 DNBR Limit

When the result of the D-prime test indicates that the DNBR distribution cannot be considered as a normal distribution at a 5% significance level, the non-parametric or distribution-free statistics are used to obtain the upper 95/95 tolerance limit. The non-parametric technique is based on order statistics and the binomial probability distribution (Reference 5). [

]^{a,c}

[

]^{a,c}

[ ]^{a,c} (2-16)

The 95/95 DNBR limit value can be further adjusted to incorporate additional margin. For example, additional margin can be incorporated into the DNBR limit to account for fuel rod bow as follows:

$$\Big[\qquad\qquad\qquad\qquad \Big]^{a,c} \qquad (2\text{-}17)$$

3 METHOD FOR RODS-IN-DNB CALCULATION

The amount of fuel failure can be determined based on an existing statistical convolution method of calculating the number of fuel rods in DNB for Non-LOCA Condition III or IV radiological consequence analyses (Reference 9). This method has been approved and applied to the CE-NSSS plant analyses, for example, a recent application in Reference 10. The statistical convolution method considers the probability of DNB on the calculated minimum DNBR value to determine the number of failed fuel rods in the reactor core. Any fuel rod heat flux reaching DNB during the transient is conservatively assumed to fail for the radiological consequence evaluations. However, there is only a 5% probability with the 95% confidence level that DNB would occur if a fuel rod DNBR is at the 95/95 DNBR SAFDL.

3.1 INPUT TO CALCULATION

The statistical rods-in-DNB calculation requires input of DNBR versus fuel rod power factor, DNB probability distribution, and the fuel census table which consists of the fuel rod power factor versus number of fuel rods in the reactor core.

3.1.1 DNBR versus Fuel Rod Power

The DNBR versus fuel rod power table is obtained from DNBR calculations using a subchannel code and an applicable DNB correlation for the Condition III or IV event. The method for the DNBR calculation using a subchannel code is described in code-related topical reports. For example, for a DNBR calculation using the VIPRE-W (Westinghouse version of VIPRE-01) code, the calculation method is described in Reference 11. [

]a,c

3.1.2 DNB Probability Distribution

The DNB probability distribution consists of the DNBR mean value (μ) and two standard deviations [ ]a,c The two separate standard deviations are used for conservative input of the probability distribution. [

]a,c

The DNB probability distribution is assumed to be normally distributed [ ]a,c Also, the probability of DNB is set to one

[

]a,c

3.1.3 Fuel Census Table

The fuel census table provides number of fuel rods in the core at any given fuel rod power for the particular event being examined. The definition of the fuel rod power for the census is consistent with that versus DNBR in Section 3.1.1. The fuel census table is obtained from neutronic calculations, and it can be plant or reload specific.

3.2 CALCULATION PROCESS

The number of fuel rods in DNB is calculated by the following procedure:

[ ] a,c

4 INTENDED APPLICATIONS

The intended applications of WTDP are for PWR 95/95 DNBR limit and rod-in-DNB calculations, similar to the existing methods applied to Westinghouse-NSSS and/or CE-NSSS plant designs. WTDP implementation will [ ]^{a,c} for the plant. The Westinghouse-NSSS DNBR limit, the CE-NSSS DNBR limit and rods-in-DNB applications are described further below.

4.1 WESTINGHOUSE-NSSS 95/95 DNBR LIMIT

The Westinghouse-NSSS plant designs include 2-loop, 3-loop and 4-loop Westinghouse-designed PWRs, the AP1000®[1] plant, some VVER-1000 plants, and any other PWR using RTDP (Reference 3) to calculate the 95/95 DNBR limit. In the RTDP application, uncertainties in fuel and reactor parameters and computer codes are convoluted with the uncertainty in a DNB correlation using [ ]^{a,c} to obtain the 95/95 DNBR limit.

In the WTDP application with the VIPRE-W code (Reference 11) and an applicable DNB correlation, there is no change [ ]^{a,c} Similar to RTDP, uncertainties in the following parameters are combined with the uncertainty in the DNB correlation:

- Reactor parameters (core power, coolant flow rate, coolant temperature, system pressure, and core bypass flow fraction)
- Radial power peaking factor ($F^N_{\Delta H}$)
- Engineering hot channel factor ($F^E_{\Delta H}$)
- Subchannel and transient codes.

For some plant accident analysis, uncertainties in the above parameters were treated deterministically in the DNBR calculation, which was often referred to as the Standard Thermal Design Procedure (STDP). The WTDP application does not affect the DNBR calculation using STDP, or any existing deterministic treatment of any plant parameter uncertainty.

The DNBR limit acceptance criterion remains the same as that in the Standard Review Plan (SRP): "There should be a 95-percent probability at the 95-percent confidence level that a hot fuel rod in the reactor core will not experience a DNB or a transition condition during normal operation or AOOs (Reference 2)." The approved DNB correlation limit from the Safety Evaluation Report (SER) is preserved [ ]^{a,c} The approved DNB correlation limit can be obtained from the correlation topical report. For example, the approved WRB-2M correlation limit is described in Topical Report WCAP-15025-P-A (Reference 12). The input of the uncertainty values is justified on a plant specific basis for each application.

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[1] AP1000 is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

The 95/95 DNBR limit calculation is based on the process described in Section 2.2. A sample calculation of the DNBR limit using the WTDP method and its comparison with the RTDP calculation are shown in Attachment A.

4.2 CE-NSSS 95/95 DNBR LIMIT

The CE-NSSS plant designs include PWRs designed by Combustion Engineering and any other PWR using SCU (References 4 and 5) to calculate the 95/95 DNBR limit. Uncertainties in the following parameters, referred to as system parameters, are incorporated into the DNBR limit calculation:

[ ]a,c

- The engineering enthalpy rise factor
- The systematic fuel rod pitch
- The systematic fuel rod clad outside diameter
- The engineering heat flux factor
- CHF correlation
- Subchannel code modeling.

In the WTDP application with the Westinghouse version of the VIPRE-01 code or VIPRE-W (Reference 11) and an applicable DNB correlation, there is no change [ ]a,c as described in the MSCU report (Reference 6). An applicable DNB correlation with VIPRE-W for CE-NSSS applications is described in WCAP-16523-P-A (Reference 13).

The major improvement in the WTDP calculation method, as compared to the SCU method (References 4 and 5), is to [ ]a,c Due to limitations on computing capabilities and costs at the time, the SCU method used a DNBR response surface process to calculate DNBR values based on a reduced number of subchannel code calculations. The response surface methodology followed the orthogonal center composite experiment design [ ]a,c

In the WTDP process, [ ]a,c and all DNBR values are calculated using the subchannel thermal hydraulic code approved for the plant application. The state parameter conditions are [ ]a,c The WTDP DNBR limit [ ]a,c is determined on a 95/95 basis using the process described in Section 2.2.

The input to the calculation is justified on a plant specific basis for each application. A sample calculation of the WTDP DNBR limit and a comparison with the SCU result using the response surface are shown in Attachment B.

WTDP supports implementation of a single subchannel code in DNBR uncertainty evaluations as part of CE-NSSS transient and setpoint analyses using the MSCU methodology (References 6 and 14). A simplified computer code, CETOP-D (Reference 15), was used for DNBR calculations in the transient and setpoint analyses in addition to a subchannel code, such as TORC (Reference 16), due to limitations of computing capabilities and costs at the time. The WTDP application with the Westinghouse version of the VIPRE-01 code, VIPRE-W, enables a simplification of the MSCU interface and process improvement by eliminating use of the CETOP-D code [ ]^{a,c} in the uncertainty evaluations and DNBR calculations. Such simplification does not change [ ]^{a,c} as described in References 6 and 14.

4.3 RODS-IN-DNB FOR CONDITION III & IV EVENTS

The statistical rods-in-DNB calculating method is similar to that in Reference 9 and has been applied to CE-NSSS PWR Non-LOCA Conditions III and IV DNB limiting events including those shown in Reference 10:

- Increased Heat Removal by the Secondary System
- Decrease in Reactor Coolant Flowrate
- Reactivity and Power Distribution Anomalies
- IOSGADV+LOP (Indavertent Opening of a Steam Generator Atmospheric Dump Valve plus the Loss of Offsite Power).

For the CE-NSSS events above, there is [ ]^{a,c} as described in Reference 9 and applied in Reference 10, such as the input to the rods-in-DNB calculation, the calculation procedure, the acceptance criterion, and the design interface.

The WTDP rods-in-DNB calculating method in Chapter 3 will be applied to the rods-in-DNB evaluations for Westinghouse-NSSS Non-LOCA Condition III or IV events including:

- Locked rotor
- Control Rod Ejection.

The input to the calculation as described in Section 3.1 is justified on a plant specific basis for each application. The fuel failure probability distribution is determined on a plant specific basis using the applicable DNB correlation and its DNBR limit. The input of the DNBR SAFDL can be the existing RTDP DNBR limit for the plant. A sample rods-in-DNB calculation for a Westinghouse-NSSS Condition IV event and a comparison with the result of the deterministic method are shown in Attachment C.

5 SUMMARY

The Westinghouse Thermal Design Procedure, WTDP, consolidates the existing methods and calculation procedures such as RTDP and SCU for statistical DNBR limit for Condition I or II events and statistical rods-in-DNB convolution for non-LOCA Condition III or IV events. WTDP is applicable to PWR plant designs, including the operating Westinghouse-NSSS and CE-NSSS plants in the U.S. The WTDP calculation method and process are described in Sections 2 and 3 for the 95/95 DNBR limit and rods in DNB convolution, respectively. The intended applications are described in Section 4. Sample calculations for different plant designs are described in the attachments.

A WTDP application to a plant, as an alternative to either RTDP or SCU, will be based on the following conditions:

- WTDP shall be used with an approved subchannel code and DNB correlation for the plant application;
- Input of parameter uncertainties to the 95/95 DNBR limit calculation shall be justified on a plant specific basis;
- Input of DNBR limit to the rods-in-DNB evaluation shall be justified on a plant specific basis;
- The plant application shall reference this report for the statistical DNBR limit method or rods-in-DNB calculation method;
- For CE-NSSS plant using the VIPRE-W code in replacement of the CETOP-D code, the WTDP application shall be within the limits and conditions of the CE-NSSS setpoint methodology as defined in WCAP-16500-P-A Supplement 1 Revision 1.

6 REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 50, “Domestic Licensing of Production and Utilization Facilities,” https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/

2. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/

3. A. J. Friedland and S. Ray, “Revised Thermal Design Procedure,” WCAP-11397-P-A, Westinghouse Electric Corporation, April 1989.

4. “Statistical Combination of Uncertainties Part 1; Combination of System Parameter Uncertainties in Thermal Margin Analyses for San Onofre Nuclear Units 2 and 3,” CEN-283(S)-P Revision 0, ABB Combustion Engineering, June 1984.

5. "Statistical Combination of Uncertainties Part 2; Uncertainty Analysis of Limiting Safety System Settings San Onofre Nuclear Generating Station Units 2 and 3," CEN-283(S)-P Revision 0, ABB Combustion Engineering, October 1984.

6. “Modified Statistical Combination of Uncertainties,” CEN-356(V)-P-A, Rev.01-P-A, ABB Combustion Engineering, May 1988.

7. "Fuel Safety Limit Calculation Inputs Were Inconsistent With NRC-Approved Correlation Limit Values," NRC Information Notice 2014-1, February 21, 2014.

8. “Assessment of the Assumption of Normality (Employing Individual Observed Values),” N15.15-1974, American National Standard Institute (ANSI), October 1973.

9. “Loss of Flow C-E Methods for Loss of Flow Analysis,” CENPD-183-A, ABB Combustion Engineering, June 1984.

10. Palo Verde Nuclear Generation Station Units 1, 2 and 3 Updated Final Safety Analysis Report, Revision 19, June 2017.

11. Y. Sung, P. Schueren and A. Meliksetian, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” WCAP-14565-P-A/WCAP-15306-NP-A, Westinghouse Electric Company LLC, October 1999.

12. L. David Smith III, et al, “Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids,” WCAP-15025-P-A, Westinghouse Electric Company LLC, April 1999.

13. Paul F. Joffre, et al, “Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes,” WCAP-16523-P-A, Westinghouse Electric Company LLC, August 2007.

14. M. A. Book and W. L. Greene, “Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel,” WCAP-16500-P-A Supplement 1 Revision 1, December 2010.

15. “CETOP-D Code Structure and Modeling Methods for San Onofre Nuclear Generating Station Units 2 and 3,” CEN-160(S)-P, ABB Combustion Engineering, September 1981.

16. “TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core,” CENPD-161-P-A, ABB Combustion Engineering, April 1986.

ATTACHMENT A – SAMPLE CALCULATION OF 95/95 DNBR LIMIT FOR WESTINGHOUSE-NSSS PLANT DESIGN

Sample Calculation of 95/95 DNBR Limit for Westinghouse-NSSS Plant Design

A sample calculation of the 95/95 DNBR limit using the WTDP method for a Westinghouse-NSSS plant design is described in this attachment. The WTDP calculated results are compared to the values based on the RTDP method (Reference A-1).

The sample calculation was performed for a Westinghouse-NSSS 4-loop PWR loaded with the 12-foot Westinghouse 17x17 VANTAGE-5 (V-5) fuel assemblies, also referred to as the 17x17 V-5 fuel design. The V-5 fuel rod outside diameter is 0.360 inches. The 17x17 V-5 fuel design is comprised of six mixing vane and three intermediate flow mixer (IFM) grid spacers across the active length where DNBR is predicted using the WRB-2 CHF correlation, Reference A-2, and the VIPRE-W code, Reference A-3.

The WTDP method is described in Chapter 2 of the report. The WTDP sample calculation is described below.

A.1 Parameter Uncertainty Input

Uncertainties in the following parameters were input to the WTDP DNBR limit calculation:

- Reactor power
- Reactor coolant inlet temperature
- Reactor flow rate
- Core bypass flow
- Reactor system pressure
- Nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$
- Engineering enthalpy rise hot channel factor, $F^E_{\Delta H}$
- Computer codes
- WRB-2 CHF correlation

The parameter uncertainties are summarized in Table A-1. The approved WRB-2 CHF correlation DNBR limit of 1.17, Reference A-2, was preserved [

]^{a,c} The code and modeling uncertainties were the same as those used for the RTDP calculations.

A.2 Parameter Sensitivities

In the WTDP calculation, the entire design space was sampled [

]^{a,c} The range of core design conditions and the DNB correlation parameter range used in the sampling are presented in Table A-3.

A.3 VIPRE-W Model

DNBR calculations were performed using the VIPRE-W code and the reactor core modeling approach described in Reference A-3. The model represented the one-eighth core that consists of [ ]a,c as shown in Figure A-1.

A.4 WTDP DNBR Limit

The WTDP statistical treatment involved combining the reactor core and fuel parameter uncertainties with the WRB-2 CHF correlation uncertainties using the Monte Carlo sampling techniques. The 95/95 DNBR limit was determined from the resultant DNBR distribution. Each delta-DNBR (ΔDNBR) was based on running a pair of VIPRE-W cases. [ ]a,c The ΔDNBR between the two cases was then applied to the DNBR sampled [ ]a,c to obtain DNBR sample. This process was repeated for [ ]a,c times through the Monte Carlo sampling process.

The resultant DNBR distribution of the collected DNBR samples, Figure A-2, was checked for normality using the D-Prime test. The D-Prime testing results for the data distribution in Figure A-2 passed the normality test at a 5% significant level. For a normally distributed DNBR data samples, the WTDP 95/95 DNBR value was [ ]a,c.

A sensitivity study was performed by increasing the Monte Carlo Sampling process to [ ]a,c times. The D-Prime testing results for the data distribution of the DNBR samples in Figure A-3 failed the normality test at a 5% significant level. For distribution-free DNBR data samples, the WTDP 95/95 DNBR value was [ ]a,c using the non-parametric statistics.

A.5 Comparison with RTDP DNBR Limit

The RTDP method (Reference A-1) combined plant and fuel parameter uncertainties with CHF correlation and code uncertainties to determine the 95/95 DNBR limit. It was based on the [ ]a,c that accounted for DNBR sensitivity to the plant and fuel parameters to obtain [ ]a,c

In the RTDP calculation, sensitivities of DNBR to changes in the parameters were determined from several sets of the reactor statepoints consisting of the power, flow, temperature, and pressure. The statepoints covered the [ ]a,c as shown in Table A-2. For each statepoint, a DNBR value was calculated by combining the DNBR variances in the reactor core and fuel parameters with the correlation uncertainty. The 95/95 DNBR design limit was obtained from the most limiting statepoint at which the DNBR sensitivities due to the parameter uncertainties resulted in the highest DNBR value. The 95/95 RTDP limit was [ ]a,c.

A comparison between the WTDP and RTDP calculations is summarized in Table A-4. [ ]a,c

A.6 References

A-1. WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

A-2. WCAP-10444-P-A, “Reference Core Report - VANTAGE 5 Fuel Assembly,” September 1985.

A-3. WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

Figure A-1 – VIPRE-W Model for 4-Loop PWR, 1/8th Core with 1/8th Hot Assembly

a,c

Figure A-2 – Histogram of WTDP Sampled DNBR Distribution of [ ]a,c Data Points

a,c

Figure A-3 – Histogram of WTDP Sampled DNBR Distribution of [ ]a,c Data Points

a,c

Table A-1 – Parameter Uncertainty Input to Sample WTDP and RTDP Calculations

Parameter	Mean	Uncertainty	Standard Deviation	Distribution
Core Power, Fraction	1.0	[		]a,c
Inlet Temperature, °F	556.6			
Pressure, psia	2270			
Flow Rate, percent	100			
Core Bypass Flow, Fraction	0.924			
$F^N_{\Delta H}$	1.635			
$F^E_{\Delta H}$	1.0			
Computer Codes	1.0			
WRB-2 DNBR	[ ]a,c			
Limit of 1.17	[ ]a,c			

Table A-2 – Statepoint Conditions for Sample RTDP Calculation

Condition Description	Core Power (Fraction)	Inlet Temperature (°F)	Inlet Flow Rate (Fraction)	Core Pressure (psia)	$F^N_{\Delta H}$	Axial Offset (%)
[						]a,c

Table A-3 – Design Space for Sample WTDP Calculation

Parameter	Range
Core Power, % of Rated[1]	[ ] a,c
Core Inlet Temperature, °F	
Core Inlet Flow, % of MMF[2]	
Pressurizer Pressure, psia	
Local Flow at WRB-2 MDNBR Location, Mlbm/hr-ft^2	
Maximum Local Quality at WRB-2 MDNBR, Location, Fraction	
DNB-Limiting Axial Power Shapes	

[1] Rated core power was 3648 MWt.

[2] MMF (Minimum Measured Flow) was 380,900 gpm.

Table A-4 – Comparison of WTDP and RTDP Calculations

Parameter	RTDP	WTDP
DNB Limiting Conditions	Table A-2	Table A-3
Uncertainty Input	Table A-1	Table A-1
Calculating Method	Root Sum Square	Monte Carlo
Calculated 95/95 DNBR Limit	[ ]^{a,c}	[ ]^{a,c} [ ]^{a,c}

ATTACHMENT B – SAMPLE CALCULATION OF 95/95 DNBR LIMIT FOR CE-NSSS PLANT DESIGN

Sample Calculation of 95/95 DNBR Limit for CE-NSSS Plant Designs

A sample calculation of the 95/95 DNBR limit using the WTDP method for a CE-NSSS plant with the Combustion Engineering 16x16 Next Generation Fuel (CE16NGF) assemblies is described in this attachment. The WTDP calculated results are compared to the values based on the SCU method (References B-1 and B-2). There was no change [ ]^{a,c} when WTDP is used as an alternative to SCU.

The CE16NGF design is equipped with mixing vane grids and Intermediate Flow Mixing (IFM) grids, as described in Reference B-3. For the DNBR predictions using the VIPRE-W code (Reference B-4), the WSSV CHF correlation (Reference B-5) was applied in the mixing vane and IFM grid regions.

B.1 System Parameter Uncertainties

The system parameters are related to the fuel design, and are characterized by the physical system through which the coolant passes and are inferred while the reactor is operational. Uncertainties in the following system parameters were input to the DNBR limit calculations:

- [ ]^{a,c}
- [ ]^{a,c}
- Engineering enthalpy rise hot channel factor
- Systematic fuel rod pitch
- Systematic fuel rod clad outside diameter
- Engineering heat flux factor
- VIPRE-W Computer code
- WSSV CHF correlation

1. Inlet Flow Factors – The inlet flow factors and uncertainties are presented in Figures B-1 and B-2, respectively.

2. Heat Flux and Enthalpy Rise Factors – The variations and tolerance deviations pertaining to CE16NGF design pellet density, fuel enrichment, pellet diameter, and clad outside diameter were used to determine the bounding values for the heat flux and enthalpy rise engineering factor for CE16NGF design.

3. Systematic Rod Pitch – The uncertainty in the systematic rod pitch accounted for variations in rod-to-rod gaps in the CE16NGF fuel assembly.

4. Systematic Rod OD – The uncertainty in the systematic rod OD of the CE16NGF design accounted for the effect of variations in subchannel flow area.

5. VIPRE-W code – A 5% uncertainty in DNBR was applied to account for the code uncertainty.

6. CHF Correlation – The WSSV CHF correlation uncertainty was obtained from Reference B-5 in measured/predicted (M/P) statistics. For the WTDP analysis, the M/P statistics were converted to the P/M (DNBR) statistics with an adjustment such that the NRC approved 95/95 DNBR limit of 1.12 was preserved.

The parameter uncertainties used as input to the DNBR limit calculations are presented in Table B-1 in terms of mean (μ95) and standard deviation (σ95) at the 95% confidence level.

B.2 State Parameter Range

The state parameters are related to the reactor design, and are measured while the reactor is operational. Their uncertainties are treated separately from the DNBR limit calculation using the MSCU process. The sensitivity of minimum DNBR to system parameter variations was determined [ ]a,c from a range of operating conditions. The range of operating conditions used in the demonstration calculation is presented below.

Parameter	Sampling Range
Inlet temperature, °F	[ ]a,c
System pressure, psia	[ ]a,c
Vessel flow, % design flow*	[ ]a,c
ASI	[ ]a,c

* % of design (445,600 gpm)

[ ]a,c

B.3 VIPRE-W Model

VIPRE-W geometric modeling was based on the single stage or one-pass modeling approach in Reference B-4, where one-eighth of the whole core was modeled using [ ]a,c as shown in Figure B-3. The radial power distribution and the inlet flow distribution for the [ ]a,c model were set to represent or bound the limiting fuel assembly.

B.4 WTDP DNBR Limit

Once the system parameters and their uncertainties, range of state parameters, and VIPRE-W model were established, the Monte Carlo simulations were made by using the parameter inputs in conjunction with the WSSV CHF correlation statistics to generate the DNBR distribution. Through the Monte Carlo simulation, DNBR samples were collected for comparison with the original SCU calculations, References B-1 and B-2. Each ΔDNBR was based on running a pair of VIPRE-W cases. The first case in the pair sampled the state parameter condition [ ]a,c

[ ]^{a,c} The second case in the pair used the same sampled state parameter conditions as the first case but randomly perturbed system parameters [ ]^{a,c} The ΔDNBR from the two cases was then applied to a DNBR sampled [ ]^{a,c} to obtain a DNBR sample for the DNBR limit distribution. This process was repeated [ ]^{a,c} times.

The resultant DNBR distribution, Figure B-4, was checked for normality using the D-Prime test. For this calculation, the D-Prime test results showed that the DNBR distribution in Figure B-4 did not pass the normality test at a 5% significance level. Consequently, the non-parametric statistic technique was applied to obtain the 95/95 DNBR limit of [ ]^{a,c} for the CE16NGF fuel.

B.5 Comparison with SCU DNBR Limit Using TORC Code

The overall SCU analysis for CE-NSSS PWR considers parameter uncertainty treatment in two groups. One group statistically combines system parameter uncertainties with code and CHF correlation uncertainties to arrive at the DNBR limit and its associated probability density function (PDF). The system parameter inputs for the sample calculation are listed in Table B-1. Uncertainties in the other group, the state parameters, are not included in the 95/95 DNBR limit or the DNBR PDF.

The SCU detailed DNBR calculations were performed using the TORC code (Reference B-6). TORC is a subchannel code derived from the COBRA-IIIC code. A two-stage TORC model contains Stage 1, [ ]^{a,c} Figure B-5, and Stage 2, [ ]^{a,c} Figure B-6. The local coolant conditions are used with the WSSV-T DNB correlation (Reference B-5) to determine the minimum DNBR value for the CE16NGF fuel design. The WSSV-T DNB correlation has the same functional form and DNBR limit as WSSV, but the WSSV-T correlation coefficients were optimized with the TORC code.

Due to limitations on computing capabilities at the time, the SCU method used a DNBR response surface process to calculate DNBR values based on a reduced number of subchannel code calculations using the TORC code. The response surface methodology followed the orthogonal center composite experiment design [ ]^{a,c} The response surface [ ]^{a,c} was used to determine SCU 95/95 DNBR limit by combining the system parameter uncertainties with the CHF correlation uncertainties. The SCU 95/95 DNBR limit was [ ]^{a,c} based on the response surface approach.

A comparison between the WTDP and SCU calculations is shown in Table B-2. The WTDP and SCU calculations were based on the same system parameter ranges and uncertainty inputs but different calculation processes. [

]^{a,c} The WTDP 95/95 DNBR limit was [ ]^{a,c} in the sample calculation, as compared to the original SCU 95/95 DNBR limit of [

]^{a,c}

B.6 References

B-1. LD-82-054, Enclosure 1-P, "Statistical Combination of Uncertainties, Combination of System Parameter Uncertainties in Thermal Margin Analyses for SYSTEM 80," May 1982.

B-2. LD-82-054, Supplement 1-P to Enclosure 1-P, "System 80 Inlet Flow Distribution Supplement 1-P to Enclosure 1-P to LD-82-054," February 1993.

B-3. WCAP-16500-P-A, "CE 16x16 Next Generation Fuel Core Reference Report," August 2007.

B-4. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

B-5. WCAP-16523-P-A, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," August 2007.

B-6. "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P-A, ABB Combustion Engineering, April 1986.

Figure B-1 – CE-NSSS Core Inlet Flow Distribution

a,c

Figure B-2 – CE-NSSS Inlet Flow Factor Uncertainties

a,c

Figure B-3 - VIPRE-W Model for CE-NSSS PWR (1/8th Hot Assembly and Core)

a,c

Figure B-4 - Histogram of WTDP Sampled DNBR Distribution with WSSV Correlation

a,c

Figure B-5 - Quarter Core Stage 1 TORC Model

a,c

Figure B-6 - 16x16 Assembly – Stage 2 TORC Model

a,c

Table B-1 – CE-NSSS PWR System Parameter Uncertainties and CHF Data

Parameter	μ95	σ95	Distribution
Box* 3 inlet flow factor			
Box 8 inlet flow factor			
Box 9 inlet flow factor			
Box 10 inlet flow factor			
Box 16 inlet flow factor			
Enthalpy rise factor			
Systematic pitch, inch			
Systematic rod OD, inch			
Heat flux factor			
WSSV CHF Data (M/P)			
VIPRE-W uncertainty			

a,c

* Box number is the fuel assembly number in Figure B-5

Table B-2 – Comparison of WTDP and SCU Calculations

Parameter	SCU	WTDP
Inlet Flow Factors	Table B-1	Table B-1
Enthalpy Rise Factor	Table B-1	Table B-1
Systematic pitch	Table B-1	Table B-1
Systematic rod OD	Table B-1	Table B-1
Heat flux factor	Table B-1	Table B-1
Computer Codes	Table B-1	Table B-1
WSSV M/P (DNBR Limit = 1.12)	[ ]a,c	[ ]a,c
DNB Limiting Conditions	[ ]a,c	[ ]a,c
Calculating Method	Monte Carlo	Monte Carlo
Subchannel Code	TORC	VIPRE-W
Calculated 95/95 DNBR Limit	[ ]a,c	[ ]a,c

ATTACHMENT C – SAMPLE CALCULATION OF RODS-IN-DNB FOR WESTINGHOUSE-NSSS PLANT

Sample Calculation of Rods-in-DNB for Westinghouse-NSSS Plant

A sample evaluation of fuel rods in DNB was performed for a locked rotor event of a Westinghouse-NSSS 4-loop plant using the 17x17 VANTAGE-5 (V-5) fuel assemblies. The locked rotor event was initiated with instantaneous seizure of a reactor coolant pump rotor, resulting in a rapid reduction in the reactor coolant rate. It is classified as a Condition IV event (limiting faults) for which fission product releases must meet the requirements of 10CFR100. In the plant safety analysis, any fuel rod reaching DNB was conservatively assumed to fail for input to the site radiological consequence evaluation.

The previous rods-in-DNB evaluation method, as presented in Reference C-1, was based on the deterministic approach. It assumed that all rods experienced DNB when DNBRs were below a DNBR design limit that was more conservative than the 95/95 criterion. The statistical convolution approach (Reference C-2), as described in Section 3 of this report, considers probability of rod experiencing DNB based on the calculated DNBR value. There is less than 5% probability with 95% confidence that DNB will occur at the 95/95 DNBR limit, since the limit is designed to protect the rods from DNB occurrence.

C.1 Calculation Input

The reactor core and fuel design parameters of the sample calculation are listed in Table C-1. The locked rotor statepoint, or the core boundary condition at the DNB limiting time step, is shown in Table C-2. The DNBR calculations were performed using the VIPRE-W code (Reference C-3) and the WRB-2 CHF (DNB) correlation (Reference C-4). The fuel rod census table is provided in Table C-3. The DNB probability distribution based on the WRB-2 correlation statistics and the plant DNBR SAFDL is shown in Table C-4. [

]^{a,c}

C.2 Statistical Rods in DNB

The statistical convolution method, as described in Section 3.0 of the report, was used to determine rods-in-DNB based on the DNB probability distribution. The fuel census curve for the locked rotor event was used to group the rods experiencing DNB [

]^{a,c} Fuel rod power and DNBR pairs were generated [

]^{a,c} as presented in Table C-5. The fuel census curve, fuel rod power and DNBR pairs, and DNBR distribution statistics were used to compute rods-in-DNB.

The fuel rod power versus DNBR table from the VIPRE-W calculation was used to determine DNBR for each specified fuel rod power interval of the fuel census curve. [

]^{a,c}

[

]a,c

For the given locked rod event statepoints and fuel census curve, the statistical rods in DNB was calculated to be [ ]a,c based on the DNBR probability distribution corresponding to the DNBR limit of [ ]a,c which contained DNBR margin to the 95/95 acceptance criterion. The deterministic rods in DNB value for the same input was calculated to be [ ]a,c, based on the conservative assumption that a fuel rod was in DNB when the calculated minimum DNBR fell below the DNBR of [ ]a,c.

C.3 References

C-1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

C-2. CENPD-183-A, "Loss of Flow C-E Methods for Loss of Flow Analysis", June 1984.

C-3. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

C-4. WCAP-10444-P-A, "Reference Core Report - VANTAGE 5 Fuel Assembly," September 1985.

Table C-1 – Westinghouse NSSS PWR Reactor Core and Fuel Design Parameters

Parameter	Value
Fuel Assembly Type	17x17 V-5
Fuel Rod Outside Diameter, inches	0.360
Nominal Fuel Heated Length, ft	12
Core Power, MWth	3648
Core Inlet Temperature, °F	556.6
Core Inlet Flow, gpm	386,000
Core Bypass Flow, %	7.6
Core Pressure, psia	2270
Radial Peaking Factor	1.635
CHF Correlation	WRB-2

Table C-2 – Locked Rotor Statepoint at DNB Limiting Time Step

Parameter	Value
Time, seconds	[ ]a,c
Pressure, fraction	
Inlet Enthalpy, fraction	
Inlet Flow, fraction	
Nuclear Power, fraction	
Radial Peaking Factor	

Table C-3 – Fuel Rod Census Curve

$F_{\Delta H}$	Fraction of Core (%)	$F_{\Delta H}$	Fraction of Core (%)	$F_{\Delta H}$	Fraction of Core (%)	$F_{\Delta H}$	Fraction of Core (%)

a,c

Table C-4 – DNB Probability Distribution

Parameter	Value
95/95 DNBR Limit	[ ][a,c]
Mean Value of DNB Probability	[ ][a,c]
[ ][a,c]	[ ][a,c]
[ ][a,c]	[ ][a,c]

Table C-5 – Fuel Rod Power and DNBR Pairs

Fuel Rod Power	DNBR
[	] a,c