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LTR-NRC-20-29

April 13, 2020

Subject: Submittal of WCAP-18240-P-A/WCAP-18240-NP-A, "Westinghouse Thermal Design Procedure (WTDP)," (Proprietary/Non-Proprietary)

Enclosed are the proprietary and non-proprietary versions of WCAP-18240, "Westinghouse Thermal Design Procedure (WTDP)," dated April 2020, submitted for review and approval under the NRC's licensing topical report program for referencing in licensing applications.

This submittal contains proprietary information of Westinghouse Electric Company LLC ("Westinghouse"). In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Nuclear Regulatory Commission's ("Commission's") regulations, we are enclosing with this submittal an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of this submittal or the Westinghouse Affidavit should reference AW-20-5036 and should be addressed to Korey L Hosack, Manager, Licensing, Analysis, and Testing, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 1, Suite 133, Cranberry Township, PA 16066.

Korey L. Hosack, Manager
Licensing, Analysis, and Testing

cc: Ekaterina Lenning (NRC)
Dennis Morey (NRC)

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Korey L. Hosack, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting that the WCAP-18240-P-A enclosure to LTR-NRC-20-29 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission 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 and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

AFFIDAVIT

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters

AFFIDAVIT

refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 2020 04 09

Korey L. Hosack, Manager
Licensing, Analysis, and Testing

WCAP-18240-NP-A
Revision 0

April 2020

Westinghouse Thermal Design Procedure (WTDP)

WCAP-18240-NP-A
Revision 0

WESTINGHOUSE THERMAL DESIGN PROCEDURE (WTDP)

Authors*
Yixing Sung
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April 2020

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PWR Core Methods

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
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Table of Contents

Section A

Final Safety Evaluation

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

January 14, 2020

Ms. Camille Zozula, Manager
Infrastructure & Facilities Licensing
Westinghouse Electric Company
1000 Westinghouse Drive, Building 1, Suite 165
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATION FOR WESTINGHOUSE ELECTRIC COMPANY TOPICAL REPORT WCAP-18240-P/WCAP-18240-NP, REVISION 0, "WESTINGHOUSE THERMAL DESIGN PROCEDURE (WTDP)" (EPID: L-2018-TOP-0033)

Dear Ms. Zozula:

By letter dated August 27, 2018 (Agencywide Documents Access Management System Accession No. ML18242A238), Westinghouse Electric Company (Westinghouse) submitted to the U.S. Nuclear Regulatory Commission (NRC) a request for review and approval of Topical Report (TR) WCAP-18240-P/WCAP-18240-NP, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)." By letter dated May 14, 2019 (ADAMS Accession No. ML19108A135), the NRC issued its request for additional information (RAI) questions for the review of WCAP-18240-P/WCAP-18240-NP, Revision 0.

The enclosed final SE addresses the applicability of WCAP-18240-P/WCAP-18240-NP, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)."

The NRC staff has found that WCAP-18240-P/WCAP-18240-NP, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)," is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the TR and the enclosed SE.

Our acceptance applies only to material provided in the subject TRs. In accordance with the guidance provided on the NRC website, we request that Westinghouse publish accepted proprietary and non-proprietary versions of these TRs within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC RAI questions and your responses. The accepted versions shall include an "-A" (designating accepted) following the TRs identification symbol.

As an alternative to including the RAI questions and RAI responses behind the title page, if changes to the TRs were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAI questions:

1. The RAI questions and RAI responses can be included as an Appendix to the accepted version.

NOTICE: Enclosure 2 transmitted herewith contains SUNSI. When separated from Enclosure 2 this transmittal document is decontrolled.

C. Zozula

2. The RAI questions and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TRs. The table should reference the specific RAI questions and RAI responses which resulted in any changes, as shown in the accepted version of the TRs.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Westinghouse will be expected to revise the TR appropriately or justify its continued applicability for subsequent referencing. Licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

Sincerely,

/RA/

Dennis C. Morey, Chief
Licensing Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 99902038

Enclosures:
As stated

SUBJECT: FINAL SAFETY EVALUATION FOR WESTINGHOUSE ELECTRIC COMPANY TOPICAL REPORT WCAP-18240-P/WCAP-18240-NP, REVISION 0, “WESTINGHOUSE THERMAL DESIGN PROCEDURE (WTDP)” (EPID: L-2018-TOP-0033) DATED JANUARY 14, 2020

DISTRIBUTION:
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***via e-mail** **NRR-106**

OFFICE	NRR/DORL/LLPB	NRR/DORL/LLPB/LA*	NRR/DSS/SFNB*	NRR/DORL/LLPB/BC
NAME	ELenning	DHarrison	RLukes	DMorey
DATE	12/03/19	1/06/20	1/06/20	1/14/20

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U. S. NUCLEAR REGULATORY COMMISSION

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-18240-P/WCAP-18240-NP, REVISION 0

"WESTINGHOUSE THERMAL DESIGN PROCEDURE"

WESTINGHOUSE ELECTRIC COMPANY

EPID L-2018-TOP-0033

Enclosure 1

TABLE OF CONTENTS

1.0 INTRODUCTION

By letter dated August 27, 2018 (Ref. 1), Westinghouse Electric Company (Westinghouse) submitted topical report (TR) WCAP-18240-P/WCAP-18240-NP, Revision 0, “Westinghouse Thermal Design Procedure (WTDP)” (Ref. 2) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The purpose of this TR was to describe a new methodology for determining the statistical departure from nucleate boiling ratio (DNBR) limit for anticipated operational occurrences (AOOs) and calculating the number of rods that experience departure from nucleate boiling (DNB) for postulated accidents. The methodology is intended to be applicable to pressurized water reactors (PWRs), including those with Combustion Engineering (CE)- and Westinghouse-designed nuclear steam supply systems (NSSSs).

The complete list of correspondence between the NRC and Westinghouse is provided in Table 1 below. This includes Requests for Additional Information (RAIs), responses to RAIs, audit documentation, and any other correspondence relevant to this review.

Table 1: List of Key Correspondence

Sender	Document	Document Date	Reference
Westinghouse	Submittal Letter	August 27, 2018	1
Westinghouse	Topical Report	August 27, 2018	2
NRC	Acceptance Letter	November 5, 2018	3
NRC	Audit Plan	March 18, 2019	4
NRC	Round 1 RAIs	May 14, 2019	5
NRC	Audit Summary	July 9, 2019	6
Westinghouse	Round 1 RAI Responses	July 12, 2019	7

A brief summary of the RAIs is provided in Table 2 below.

Table 2: Listing of RAIs

RAI	Subject
RAI-WTDP-01	Clarification of mathematical method
RAI-WTDP-02	Epistemic Uncertainties
RAI-WTDP-03	∆DNBR Spatial Sensitivity
RAI-WTDP-04	Criteria for case exclusion

This review was performed within the guidelines of LIC-500 (Ref. 8). Additionally, the NRC staff chose to use a tiger team approach to perform the review. This approach has been previously suggested by various stakeholders, including industry representatives. Due to the NRC staff familiarity with the subject matter and the short length of the TR, the NRC staff determined the tiger team approach was appropriate for this review.

2.0 REGULATORY EVALUATION

The WTDP TR describes a method for calculating a statistical limit on the DNBR, below which fuel failure may occur. TR also describes a method for using the statistical DNBR limit to determine the number of rods that would be expected to be damaged due to DNB during an accident. These two aspects of the WTDP methodology, though related, are reviewed separately because they relate to different regulatory criteria.

2.1 Statistical DNBR Limit

General Design Criterion (GDC) 10 from Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," requires licensees to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including the effects of AOOs. In pressurized water reactors (PWRs), departing from the nucleate boiling regime could significantly reduce the ability to transfer heat from the fuel rods to the coolant, resulting in an excessive increase in cladding temperature that could cause cladding failure. As such, prevention of departure from nucleate boiling is typically identified as a SAFDL for PWRs. The ratio of the heat flux at which DNB is expected to occur, also known as the critical heat flux (CHF), to the actual heat flux is known as the DNBR. Departure from nucleate boiling is generally prevented by ensuring that the reactor remains above a specified DNBR limit during operation.

The NRC staff reviews thermal-hydraulic analyses using the guidance contained in NUREG-0800, "Standard Review Plan" (SRP), Section 4.4, "Thermal and Hydraulic Design." SRP 4.4 provides criteria for ensuring the requirements of GDC 10 are met. SRP Acceptance Criterion 1 discusses the use of a limit on the DNBR that provides assurance that there is a 95-percent probability at a 95-percent confidence level that the hot rod in the core does not experience DNB during normal operation or AOOs – this is commonly known as a 95/95 DNBR limit.

The regulation at 10 CFR 50.34, "Contents of Applications; Technical Information," provides requirements for nuclear reactor licensees to provide in a final safety analysis report (FSAR) an evaluation of the design and performance of structures, systems, and components of the facility, including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility. In practice, PWRs include analyses of normal operation and transient conditions in their FSARs that evaluate margin to the DNBR limit.

2.2 Rods-In-DNB Evaluation

As discussed above, GDC 10 requires that SAFDLs not be exceeded for normal operation and AOOs. However, certain postulated accidents have been identified which have the potential to fail fuel. For these accidents, the radiological release must be evaluated and is subject to regulatory limits, either by evaluating margin to the 10 CFR 100 dose limits or by performing an accident source term analysis in accordance with 10 CFR 50.67.

As part of the radiological consequence analysis for a given transient, a fraction of the fuel rods in the core is presumed to fail. The rods-in-DNB evaluation proposed as part of WTDP is used to evaluate the number of rods expected to experience DNB during the transient. All rods that experience DNB are assumed to fail. The number of rods that fail in this manner are counted and compared to the number of failed rods in the radiological consequence analysis to ensure acceptability. SRP 15.3.3-15.3.4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break," and SRP 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," provide examples of additional, more detailed guidance on the review of the DNBR criterion and the failed rod census.

2.3 Regulatory History

The NRC staff has reviewed and approved similar methods for performing the statistical DNBR limit analysis discussed in WTDP. The following is a list of the most pertinent methods for DNBR limit analysis:

- WCAP-11397-P-A, "Revised Thermal Design Procedure" (Ref. 9)
- WCAP- 8567-P-A, "Improved Thermal Design Procedure" (Ref 10)
- CEN-283(S)-P, "Statistical Combination of Uncertainties" (Refs. 11 and 12)
- CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties" (Ref. 13)
- WCAP-16500-P-A Supplement 1, Revision 1, "Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF) (Ref. 15)

The following is a list of the most pertinent methods for statistical rods-in-DNB analysis:

- CENPD-183-A, "Loss of Flow" (Ref. 16)
- Palo Verde Nuclear Generation Station Units 1, 2 and 3 Updated Final Safety Analysis Report (FSAR), Revision 19 (Ref. 14)

For further details on the history, see Section 1.2 of the WTDP TR.

2.4 Criteria for this Review

As there are currently no formal frameworks to assess uncertainty quantification methodologies such as WTDP, the NRC staff used portions of the framework described in NUREG/KM-0013 (Ref. 17) as well as their own knowledge and experience to ensure that the estimate of the DNBR limit as well as rods-in-DNB was acceptable. This included ensuring that there was evidence to support the common assumptions made by uncertainty quantification methods such as assuming a set of values is from a normal distribution, assuming a set of values is independent of specific parameters, and assuming certain epistemic uncertainties can be treated as aleatory uncertainties.

3.0 TECHNICAL EVALUATION

The NRC staff considered three separate areas of evaluation for WTDP: the calculation of the DNBR limit, the calculation of the rods-in-DNB, and the replacement of CETOP-D. Each area is discussed below.

3.1 DNBR Limit

Westinghouse proposed to use a Monte Carlo approach to determine the statistical 95/95 DNBR limit. This approach samples operating conditions to determine DNBR sensitivities to fuel parameters and instrument uncertainties. The DNBR sensitivity is then combined with the uncertainty in the prediction of DNBR to calculate the overall DNBR uncertainty distribution.

The statistical DNBR limit is the 95/95 upper tolerance limit of this distribution. This limit is considered one of the GDC 10 SAFDLs. This approach is described in Section 2 of the TR with further details provided in response to RAI-WTDP-01.

The approach proposed by Westinghouse to determine a statistical DNBR limit is based on the existing Statistical Combination of Uncertainties or Modified Statistical Combination of Uncertainties (SCU/MSCU) methodologies that have been approved by the NRC for CE plants. However, WTDP is intended to stand on its own as a replacement for the current statistical DNBR limit methodologies for both CE plants and Westinghouse plants (e.g., SCU/MSCU, Improved Thermal Design Procedure (ITDP), and Revised Thermal Design Procedure (RTDP) – see Refs. 9 through 14). Therefore, the NRC staff considered the prior approval of the SCU/MSCU methodology as a context for the review of the WTDP DNBR limit method, and reviewed WTDP as a standalone methodology.

3.1.1 Input Selection

As discussed in Section 2.1 of the TR, the inputs to WTDP include uncertainties in fuel parameters, uncertainties associated with reactor state parameters, and the range of operating space to be covered by the WTDP calculation. The fuel-related parameters (which Westinghouse refers to as the "system" parameters in the TR) include those associated with fuel manufacturing as well as those associated with the DNB correlation and the subchannel code. The operating state of the reactor is defined by the reactor power and its associated power distribution, the coolant temperature, flow rate, bypass fraction, and the reactor pressure.

Westinghouse provided a list of typical system and state parameters in Section 2.1 of the TR. However, Westinghouse also stated that statistical DNBR limits for a specific plant may or may not include all the uncertainties listed and that the uncertainty inputs will be justified on a plant-specific basis. For parameters whose uncertainty is not included in the DNBR limit calculation, Westinghouse specified that conservative values with respect to DNBR will be used. The NRC staff reviewed the parameters proposed for the uncertainty analysis and found them to be consistent with the existing RTDP and SCU methodologies. The NRC staff also expects that use of conservative values for any of the parameters listed will be more conservative than including the parameter in the uncertainty analysis. Thus, the NRC staff finds the approach for determining the uncertainties to include in the analysis to be acceptable.

The operating space of the reactor is defined by the set of operating states that occur in the transient and accident analysis. To select input for the Monte Carlo runs, Westinghouse randomly samples over the entire operating space, consistent with the NRC-approved SCU methodology. The state parameters are sampled from a uniform distribution. By using a uniform distribution, Westinghouse assumes that all statepoints are equally likely. In reality, all statepoints are not equally likely, and there is an unknown set of statepoints corresponding to the actual operation of the reactor.

When actual parameter values are unknown, it is common to assume that all values are equally likely and therefore to sample them from a uniform distribution. This is because, while there are many methods to analyze aleatory uncertainties (e.g., Monte Carlo analysis), there are few methods to analyze epistemic uncertainties, such as the case of unknown probability distributions. However, this assumption may or may not be appropriate. Therefore, the NRC staff asked RAI-WTDP-02.

In response, Westinghouse stated that the use of a uniform distribution results in higher DNBR limit than would a normal distribution. The NRC staff agrees that in many cases, the use of a uniform distribution will result in a conservative analysis compared to a normal distribution due to the increased weight given at the extremes of the distribution and the common situation in which the most extreme values of the distribution result in the most conservative cases. The NRC staff does note that this may not always be true, and that the use of a uniform distribution is not inherently conservative. However, the NRC staff does find that Westinghouse is using reasonable distributions for the sampled parameters.

Because Westinghouse demonstrated that its method adequately samples over the operating space, the NRC staff determined that it was acceptable.

3.1.2 Generation of the DNBR Sample

The Monte Carlo procedure samples multiple statepoints, with a nominal case and a sensitivity case for each statepoint. The nominal case is based on a random sample of a statepoint, with all parameters at their expected values. The sensitivity case perturbs the statepoint from the nominal case based on the uncertainties in each parameter. The parameter uncertainties accounted for in the sensitivity case include those in state parameters due to measurement, those in fuel-related parameters due to manufacturing, and those in the assembly inlet flow distribution. The specific uncertainties included in the analysis are dependent on the fuel type and plant, as discussed in additional detail in Sections 4.1 and 4.2 of the TR. An approved subchannel code (e.g., VIPRE-W) is then used to calculate the minimum DNBR from both the nominal and sensitivity cases, and the difference between the two at each statepoint is termed the ΔDNBR.

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] Because the methodology that Westinghouse is using would result in a higher variance for the ΔDNBR than expected and that higher variance will (on average) result in a higher DNBR limit than expected, the NRC considered this approach acceptable.

Once the ΔDNBR is calculated, a DNBR value is then randomly sampled based on the CHF correlation statistics to account for the uncertainty in the approved CHF correlation. Prior to the sampling, the mean associated with the CHF correlation is adjusted to account for any biases added to the correlation during the NRC approval process (including the small rounding bias), and the standard deviation is increased to account for the fact that it is based on a sample. The sampled DNBR value based on the correlation statistics is then increased to account for subchannel code uncertainty.

The resulting DNBR value is then added to the ΔDNBR to obtain a single realization of the Monte Carlo process. This entire process is repeated for a minimum of **[ ]** statepoints.

The NRC staff is aware that in some cases the code may fail or produce an error, resulting in that case not being used. The NRC staff therefore asked Westinghouse in RAI-WTDP-04 what criteria were used to ensure that the code failure or error was reasonable (e.g., the randomly selected statepoint was not physically achievable) and was not the result of a code bug or input error.

In its response, Westinghouse provided the criteria used to determine if a case could be excluded. The NRC staff reviewed these criteria and found them to be acceptable because (a) they did not allow the case to be excluded simply because it provided unfavorable results, (b) they provided an objective basis for excluding a case, and (c) they would result in a robust (i.e., consistent) calculation of the DNBR limit.

Because Westinghouse demonstrated that cases that were not used in the statistical analysis were those in which a physically unrealistic combination of state parameters were chosen, the NRC staff found Westinghouse's approach to be acceptable.

The NRC staff reviewed the process for generating the DNBR samples to determine the DNBR uncertainty and found that it would result in a representative sample set of the DNBR population over the operating space.

3.1.3 Development of the Statistical DNBR Limit

Westinghouse proposed different approaches for determining the statistical DNBR limit from the DNBR sample set obtained from the process discussed above. A parametric approach is used if the data can be shown to be from a normal distribution (e.g., Owen's table). A non-parametric approach is used if it can not be shown that the data is from a normal distribution (i.e., Wilks method). The D' test is used to determine if the data is from a normal distribution.

If the D' test shows that the data can be approximated as normal, Westinghouse will use a parametric approach to determine the 95/95 upper tolerance limit. The 95/95 upper tolerance limit is given by the formula:

$$95/95 \text{ Upper Tolerance Limit} = \mu + k\sigma$$

Westinghouse has two different options for implementing the parametric approach. The first option **[**

] The second option uses the formula above, with mean and standard deviation from the DNBR dataset and a *k* factor from Owen's tables based on the sample size. The NRC staff performing this review were familiar with the second option, but not the first; however, test calculations performed by the staff showed that the first option provided conservative results relative to the second.

If the DNBR distribution cannot be described as normal based on the results of the D' test, a non-parametric approach based on order statistics is used to determine the 95/95 upper tolerance limit.

The statistical test and the methods used for determining the DNBR limit applied are among those commonly applied in such instances or were otherwise found by the NRC staff to be conservative relative to these commonly-used methods. Thus, the NRC staff finds the proposed methods for determining the statistical DNBR limit based on the DNBR sample population distribution to be acceptable.

3.2 Rods-in-DNB

The rods-in-DNB methodology proposed in the WTDP TR is similar to that previously reviewed and approved for CE NSSS analysis (Reference 15). In the WTDP TR, Westinghouse described the method in more detail and asked for its extension to Westinghouse-designed NSSS plants.

Westinghouse uses an NRC-approved subchannel code (e.g., VIPRE-W) to calculate a table of DNBR versus fuel rod power at the limiting thermal-hydraulic statepoint from the transient analysis. Since the table is generated from the limiting thermal-hydraulic statepoint, it provides the minimum DNBR expected for a given rod power for a given transient.

Next, Westinghouse generates a table providing the probability of fuel damage as a function of DNBR at a 95% confidence level (termed the DNBR probability distribution in the TR). For example, if a rod were at the 95/95 DNBR limit value, the rod would have a 5% chance of experiencing DNB and therefore a 5% probability of failure. However, after reviewing the WTDP topical report and the previously-approved CE methodology, the NRC staff was unsure as to how this failure probability table was calculated and asked for additional details in RAI-WTDP-01. In its response, Westinghouse provided additional explanation on the process for calculating the probability of failure of a fuel rod. The NRC staff found the explanation provided a logical process, but the staff did question why **[**

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For the next step in the process, Westinghouse then generates a fuel rod census table that contains the fraction of the core greater than or equal to a given fuel rod power. It was not clear to the NRC staff how this table was used and asked for additional clarification in RAI-WTDP-01.
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After a review of the methodology, including the further details provided in response to RAI 1, the NRC staff agrees with the staff's prior conclusion that this technique is acceptable for calculating fuel rod failures caused by DNB, and that it is acceptable for use in PWR analysis.

3.3 Replacement of CETOP-D

In the CE setpoint methodology, Westinghouse uses a simplified thermal-hydraulic code known as CETOP-D to determine correction factors for the online monitoring and protection systems at CE plants. In 1981, the CETOP-D code was chosen for this use instead of a higher-fidelity subchannel code due to the large number of cases needed for the CE setpoints analysis and the relative speed of execution of the code.

In the WTDP TR, Westinghouse stated that plants may replace CETOP-D with a different NRC-approved subchannel code (e.g., VIPRE-W) to perform the same analysis. The NRC staff reviewed the evaluation model applying CETOP-D and agrees with Westinghouse that other NRC-approved subchannel codes are acceptable to perform the same evaluations, provided that they are able to use approved CHF correlations applicable to the fuel type being modeled and have adequately characterized the code and correlation uncertainties. The NRC staff notes that this change is primarily driven by the greatly increased speed of computation since the evaluation model including CETOP-D was originally implemented in the 1980s.

4.0 <u>LIMITATIONS AND CONDITIONS</u>

The use of the WTDP methodology is subject to the following limitations and conditions:

1. In application to a given plant, WTDP shall be used with a subchannel code and CHF correlation combination that has been approved for the plant type and the fuel type in use at the plant.

2. Parameter uncertainties used in the 95/95 DNBR limit calculation must be justified on a plant-specific basis.

3. The DNBR distribution used to determine the statistical DNBR limit shall be based on a minimum of **[ ]** samples from the operating space.

4. The use of an approved subchannel code (e.g., VIPRE-W) in lieu of CETOP-D must be consistent with the CE-NSSS setpoint methodology as defined in WCAP-16500-P-A, Supplement 1, “Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel,” Revision 1 (Ref. 15).

5.0 <u>CONCLUSIONS</u>

The NRC staff concluded that the WTDP methodology described in WCAP-18240-P/WCAP-18240-NP, Revision 0, “Westinghouse Thermal Design Procedure (WTDP),” describes an acceptable methodology for determining a DNBR limit that provides assurance at a 95-percent probability and 95-percent confidence level that the hot rod in the core will not experience DNB during normal operation or AOOs. The limit derived from the WTDP analysis adequately accounts for the appropriate plant uncertainties and will be applicable across the allowable operating space of the plant. For accidents in which some fuel damage is anticipated, WTDP’s rods-in-DNB method provides an acceptable method for evaluating the number of fuel rods that would be expected to experience damage due to DNB.

6.0 REFERENCES

1. Mercier, E. J., Westinghouse Electric Company, letter to U.S. Nuclear Regulatory Commission, “Submittal of WCAP-18240-P/WCAP-18240-NP, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)" (Proprietary / Non-Proprietary),” LTR-NRC-18-59, August 27, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18242A238 (Cover Letter. *Publicly Available*), ADAMS Accession No. ML18242A239 (*Request for Withholding. Publicly Available*)).

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3. Morey, D.C., U.S. Nuclear Regulatory Commission, letter to Gresham, J.A., Westinghouse Electric Company, “Acceptance for Review of Westinghouse Electric Company Topical Report WCAP-18240-P/WCAP-18240-NP, Revision 0, ‘Westinghouse Thermal Design Procedure (WTDP)’ (EPID: L-2018-TOP-0033),” November 5, 2018 (ADAMS Accession No. ML18288A103 (*Publicly Available*)).

4. Closed Audit for the Review of Westinghouse Electric Company Topical Report WCAP-18240-P/WCAP-18240-NP, Revision 0, “Westinghouse Thermal Design Procedure (WTDP),” March 18, 2019 (ADAMS Accession No. ML19077A106 (*Publicly Available*)).

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7. Hosack, K.L., Westinghouse Electric Company, letter to U.S. Nuclear Regulatory Commission, “Transmittal of Responses to the NRC Request for Additional Information for WCAP-18240-P, ‘Westinghouse Thermal Design Procedure (WTDP)’ (Proprietary/NonProprietary),” LTR-NRC-19-36, July 12, 2019 (ADAMS Accession No. ML19192A268 (*Submittal Letter and Non-Proprietary Attachment. Publicly Available*)/ML19192A275 (*Proprietary Attachment. Non-Publicly Available*)).

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11. "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 (ADAMS Accession No. ML19214A154 (*Proprietary. Non-Publicly Available*), ML19214A144 (*Non-Proprietary. Publicly Available)*).

12. "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 (ADAMS Accession No. ML19214A156 (*Proprietary. Non-Publicly Available*), ML19214A145 (*Non-Proprietary. Publicly Available*)).

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16. "Loss of Flow C-E Methods for Loss of Flow Analysis," CENPD-183-A, ABB Combustion Engineering, June 1984.

17. NUREG/KM-0013, "Credibility Assessment Framework for Critical Boiling Transition Models – Draft for Comment," March 2019 (ADAMS Accession No. ML19073A249).

Attachment: Comment Resolution

Principal Contributor: J.S. Kaizer
R. Anzalone

Date: January 14, 2020

RESOLUTION OF COMMENTS ON DRAFT SAFETY EVALUATION FOR

TOPICAL REPORT WCAP-18240-P/WCAP-18240-NP, REVISION 0,

"WESTINGHOUSE THERMAL DESIGN PROCEDURE (WTDP)"

WESTINGHOUSE ELECTRIC COMPANY

By letter dated November 15, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19319B971), Westinghouse Electric Company (Westinghouse) provided comments on the draft safety evaluation (SE) for Topical Report (TR) WCAP-18240-P/WCAP-18240-NP, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)." Westinghouse stated that there is proprietary information in the draft SE. The following is the U.S. Nuclear Regulatory Commission (NRC) staff's resolution of these comments:

Draft SE Comments for TR WCAP-18240-P/WCAP-18240-NP, Revision 0:

1. Last sentence of the fifth paragraph of Section 3.1.2, "Generation of the DNBR Sample," reads in the draft SE: "The DNBR value sampled from based on the correlation statistics is then increased to account for subchannel code uncertainty."

   Westinghouse suggested that the last sentence of the fifth paragraph of Section 3.1.2, "Generation of the DNBR Sample," should be re-worded to read: "The sampled DNBR value based on the correlation statistics is then increased to account for subchannel code uncertainty."

   NRC Resolution for Comment 1 on Draft SE

   The NRC staff has reviewed the Westinghouse comment and agrees that proposed wording provides additional clarification. The NRC staff has updated the last sentence of the fifth paragraph of Section 3.1.2, "Generation of the DNBR Sample."

   Last sentence of the fifth paragraph of Section 3.1.2, "Generation of the DNBR Sample," reads now: "The sampled DNBR value based on the correlation statistics is then increased to account for subchannel code uncertainty."

2. Westinghouse provided proprietary markings on the draft SE.

   NRC Resolution for Comment 2 on Draft SE:

   The NRC staff reviewed the Westinghouse markings and incorporated them into the final SE.

3. Westinghouse provided editorial comments.

   NRC Resolution for Comment 3 on Draft SE:

   The NRC staff reviewed the Westinghouse comments and finds them acceptable because the changes are editorial in nature.

Attachment

Section B

Listing of Changes to Approved Version of Topical Report

Updates for WCAP-18240-NP-A

Content Changes

No.	Section/Page(s)	Description/Reference
1.	Entire document	Changed document name to WCAP-18240-NP-A and document date.
2.	Front page	Changed copyright year.
3.	1.2/p. 1-5	Added ADAMS Accession Number to Reference 10 (Palo Verde UFSAR Revision 19).
4.	1.2/p. 1-5	Revised text to reflect that Reference 10 was submitted to the NRC, not that there was an NRC approval, implied or otherwise.

Section C

Submittal of Topical Report

Westinghouse Electric Company
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LTR-NRC-18-59

August 27, 2018

Subject: Submittal of WCAP-18240-P / WCAP-18240-NP, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)" (Proprietary / Non-Proprietary)

Reference: LTR-NRC-18-49 dated July 11, 2018, "Transmittal of the Pre-Submittal Meeting Slides for Topical Report WCAP-18240-P, 'Westinghouse Thermal Design Procedure (WTDP)' (Proprietary / Non-Proprietary)," ADAMS Accession No. ML18194A644

Enclosed are proprietary and non-proprietary versions of WCAP-18240-P, "Westinghouse Thermal Design Procedure (WTDP)," dated August 2018, submitted for review and approval under the NRC's licensing topical report program for referencing in licensing actions. Approval for this topical report is requested by March 2020, as discussed during the pre-submittal meeting conducted on July 18, 2018 (see the reference above).

This submittal contains proprietary information of Westinghouse Electric Company LLC ("Westinghouse"). In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Nuclear Regulatory Commission's ("Commission's") regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference AW-18-4787 and should be addressed to Edmond J. Mercier, Manager, Fuel Licensing and Regulatory Support, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 256, Cranberry Township, Pennsylvania 16066.

Edmond J. Mercier, Manager
Fuel Licensing and Regulatory Support

Enclosures

cc: Ekaterina Lenning
Dennis Morey

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AW-18-4787

August 27, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-18240-P, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)" (Proprietary)

Reference: Letter from Edmond J. Mercier to the Document Control Desk, LTR-NRC-18-59, dated August 27, 2018

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit AW-18-4787 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of this Application for Withholding or the accompanying Affidavit should reference AW-18-4787 and should be addressed to Edmond J. Mercier, Manager, Fuel Licensing and Regulatory Support, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 256, Cranberry Township, Pennsylvania 16066.

Edmond J. Mercier, Manager
Fuel Licensing and Regulatory Support

AW-18-4787

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Edmond J. Mercier, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 8/24/2018

Edmond J. Mercier, Manager
Fuel Licensing and Regulatory Support

(1) I am Manager, Fuel Licensing and Regulatory Support, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission 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 and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-18240-P, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)" (Proprietary), dated August 2018, for submittal to the Commission, being transmitted by Westinghouse Letter LTR-NRC-18-59. The proprietary information as submitted by Westinghouse is that associated with Westinghouse's upcoming request for NRC review and approval of WCAP-18240-P, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to obtain NRC approval for the application of the WTDP methodology for DNBR Limit calculations and Rods-in-DNB calculations, as documented in WCAP-18240-P, "Westinghouse Thermal Design Procedure (WTDP)."

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of obtaining NRC approval to use the WTDP methodology for DNBR Limit calculations and Rods-in-DNB calculations.

(ii) Westinghouse can sell support and defense of the WTDP methodology for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are the proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACKNOWLEDGMENT

The authors would like to thank Sumit Ray and Zeses E. Karoutas for guiding this project, Jerome C. Galiszewski for the project management, Mitch Nissley, Scott Sidner, Sharon K. Erwin, Guoqiang Wang, and Jun Liao for suggestions on the method integration and improvements, and Jeffery A. Brown, Pete A. Hilton, Hans Van de Berg, Samuel A. Eddinger, Alan MacDonald, George G. Yates, Parvez N. Khambatta, and Edmond J. Mercier for reviews of the document. The authors especially appreciate contributions of Steven F. Grill and Howard Fields to the method development and the topical report.

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 (ADAMS Accession Number ML17193A048).

The plant FSAR changes were submitted to 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:

$$DNBR_{95/95} = \mu_{95/95} + 1.645 * \sigma_{95/95} \quad (2\text{-}13)$$

Or by Method 2:

$$DNBR_{95/95} = \mu_{DNBR\ Distribution} + k_{95/95} * \sigma_{DNBR\ Distribution} \quad (2\text{-}14)$$

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:

[ ] a,c (2-17)

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.

[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/

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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			
Inlet Temperature, ºF	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 WRB-2 DNBR Limit of 1.17	1.0 [ ]a,c [ ]a,c			

a,c

Table A-2 – Statepoint Conditions for Sample RTDP Calculation

Condition Description	Core Power (Fraction)	Inlet Temperature (ºF)	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, ºF	
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, ºF	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

Section D

Submittal of Response to Request for Additional Information

Westinghouse Electric Company
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USA

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Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

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LTR-NRC-19-36

July 12, 2019

Subject: Transmittal of Responses to the NRC Request for Additional Information for WCAP-18240-P, "Westinghouse Thermal Design Procedure (WTDP)" (Proprietary / Non-Proprietary)

References: 1. LTR-NRC-18-59 dated August 27, 2018, "Submittal of WCAP-18240-P / WCAP-18240- NP, Revision 0, 'Westinghouse Thermal Design Procedure (WTDP)' (Proprietary / Non-Proprietary)," ADAMS Accession Number ML18242A237 (package)

2. Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-18240-P/WCAP-18240-NP, Revision 0, "Westinghouse Thermal Design Procedure (WTDP)," E. Lenning (NRC) to C. Zozula (Westinghouse), dated May 14, 2019 (EPID: L-2018- TOP-0033)

Reference 1 transmitted Proprietary and Non-Proprietary versions of WCAP-18240 to the NRC for review. Reference 2 requested additional information to support the NRC review. Enclosed are Proprietary and Non-Proprietary versions of the responses to the request for additional information (RAI) for WCAP-18240-P, "Westinghouse Thermal Design Procedure (WTDP)."

This submittal contains proprietary information of Westinghouse Electric Company LLC ("Westinghouse"). In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Nuclear Regulatory Commission's ("Commission's") regulations, we are enclosing with this submittal an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the this submittal or the Westinghouse Affidavit should reference AW-19-4916 and should be addressed to Camille T. Zozula, Manager, Infrastructure & Facilities Licensing, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 1, Suite 165, Cranberry Township, PA 16066.

Korey L. Hosack, Manager
Product Line Regulatory Support

cc: Ekaterina Lenning (NRC)
Dennis Morey (NRC)

Enclosures:

1. Affidavit AW-19-4916
2. Proprietary Information Notice and Copyright Notice
3. LTR-NRC-19-36 P-Attachment, Responses to the Request for Additional Information for WCAP-18240-P, “Westinghouse Thermal Design Procedure (WTDP)” (Proprietary)
4. LTR-NRC-19-36 NP-Attachment, Responses to the Request for Additional Information for WCAP-18240-NP, “Westinghouse Thermal Design Procedure (WTDP)” (Non-Proprietary)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

COUNTY OF BUTLER:

(1) I, Korey L. Hosack, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of LTR-NRC-19-36 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission 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 and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

AFFIDAVIT

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

AFFIDAVIT

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 20190240

Korey L. Hosack, Manager
Product Line Regulatory Support

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are the proprietary and non-proprietary versions of a document, furnished to the NRC in connection with the review of WCAP-18240-P / NP, "Westinghouse Thermal Design Procedure (WTDP)."

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (5)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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Responses to the Request for Additional Information for WCAP-18240-NP, "Westinghouse Thermal Design Procedure (WTDP)"

(Non-Proprietary)

July 2019

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

Response to Request for Additional Information (RAI) on WCAP-18240-P, "Westinghouse Thermal Design Procedure"

1) RAI-WTDP-01

Clarification

Provide the following clarifications:

a) *Section 2*
   a. *Rewrite the equations in a style more consistent with the previous topical reports. The equations should use common mathematical notation and each term in the equation (e.g., variables, functions, indices) should be fully defined and consistent among all of the equations.*
   b. *For random variables that are defined by probability distributions that use the mean and standard deviation or upper and lower bound, discuss how each parameter of the distribution is determined.*
   c. *Clarify the equations to specify what set of inputs are used to generate the nominal case and how that set of inputs is changed to generate the perturbed case (i.e., how the conditions for the second sub-case are determined).*

b) *Section 3*
   a. *Provide additional detail on how the fuel damage probability table (denoted the DNB probability distribution in the topical report) is defined. [*

      ] ^{a,c}
   b. *Provide additional detail on how the fuel census table, the DNBR versus fuel rod power table, and the fuel damage probability table are combined to generate the expected number of rods experiencing fuel damage due to DNB. Provide a sample calculation showing the entire process for one power interval.*

Response to each question of RAI-WTDP-01 is provided below.

a) Section 2
   a. *Rewrite the equations in a style more consistent with the previous topical reports. The equations should use common mathematical notation and each term in the equation (e.g., variables, functions, indices) should be fully defined and consistent among all of the equations.*

Response:

The process of the 95/95 DNBR limit calculation is rewritten below.

Step 1 – Sampling of State Parameters from Uniform Distributions for Core Condition

The state parameters include [ ]^{a,c} The parameter ranges are plant specific and cover normal operation and DNB-limiting conditions in the non-LOCA accident analysis for which the statistical DNBR limit is applied. The state parameters are sampled [ ]^{a,c} from their respective ranges to obtain a reactor core condition:

[ ]^{a,c}

A DNBR is calculated using the sampled core condition and nominal values for the system parameters. The sampled core condition is not used in subsequent calculations if [ ]^{a,c}

Step 2 - Parameter Sampling from Uncertainty Distributions

The system parameters can be [ ]^{a,c} engineering enthalpy rise hot channel factor, fuel rod pitch, fuel rod diameter, engineering heat flux hot channel factor, guide thimble tube diameter and grid spacer loss coefficients. A perturbed either system or state parameter value is obtained by sampling from either a uniform or a normal distribution of its uncertainty.

If it is a uniform distribution, the perturbed value is calculated by sampling a uniformly distributed random number and combining it with the difference between the upper and lower ranges of the parameter as follows:

[ ]^{a,c}

If it is a normal distribution, the perturbed value is calculated by sampling a normally distributed random number and combining it with the mean and the standard deviation for the parameter:

[ ]^{a,c}

a,c

Based on the DNBR at the reactor core condition with the perturbed parameter values and the DNBR from Step 1, a ΔDNBR for Case “i” is obtained as follows:

a,c

Equation R-4 is the equivalent of Equations 4.1 and 5.1 of Reference R-1, using explicitly calculated DNBR values in place of response surface DNBR values.

Step 3 - Sampled DNBR

A sampled DNBR value is obtained from a normal distribution based on the CHF correlation statistics that consists of a DNBR mean value and a standard deviation. A sampled CHF DNBR for Case "i" is calculated:

a,c

Effects of the system parameter uncertainties ($\Delta DNBR_i$, Equation R-4) are combined with the sampled CHF DNBR value (CHF $DNBR_i$, Equation R-5) and the subchannel code uncertainty to obtain a DNBR value for Case “i”:

a,c

Equation R-6 is the equivalent of Equation 5.2 of Reference R-1. The Subchannel Code $Uncertainty_i$ is a sampled multiplicative subchannel code uncertainty factor for case "i", consistent with the code uncertainties applied in the statistical DNBR calculations (References R-1 and R-2):

[ ]a,c

Step 4 - DNBR Limit Calculation

A DNBR limit calculation is performed based on the distribution of the DNBR values obtained from Equation R-6. The limit is determined as either normal or non-parametric upper 95/95 tolerance limit of the $DNBR_i$ distribution.

Normal DNBR Distribution

The assessment of normality is based on the probability of the D-Prime (D') test statistic (Reference R-3) for a normal distribution. To perform this assessment, probability regions for the D-Prime statistic probability are defined as shown pictorially in Figure R-1. [

]a,c

When the $DNBR_i$ distribution has been determined to be a normal distribution, the following two 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 samples in the DNBR distribution:

[ ]a,c

[ ] a,c

The 95/95 DNBR limits from the two methods above are similar. The "raw" DNBR limit is further adjusted to account for any deterministic DNBR penalty not considered in the statistical DNBR limit calculation, such as the rod bow DNBR penalty as illustrated below:

[ ] a,c

Any other deterministic adjustment can be applied in a similar manner.

Non-Normal DNBR Distribution

If the $DNBR_i$ distribution from Equation R-6 cannot be determined to be a normal distribution using the D' test statistic, non-parametric statistics are used to obtain a non-parametric upper 95/95 tolerance limit. As described in Section 2.3.2 of Reference R-2, one-sided 95/95 tolerance limits are calculated by using non-parametric techniques based on order statistics and the binomial probability distribution. The DNBR distribution is first ordered from the smallest to the largest value. The binomial distribution is used to calculate a locator "L" from the ordered DNBR distribution which estimates the one-sided tolerance limit at a 95/95 probability / confidence level. The one-sided upper 95/95 tolerance limit, $UTL_{95/95}$, is obtained by selecting the DNBR value (from the ordered DNBR distribution) corresponding to the locator "L".

A non-parametric "kσ" can be obtained by using the mean of DNBR distribution, the one-sided upper 95/95 tolerance limit, and rearranging the following equation:

[ ] a,c

Similar to Equation R-14, the DNBR limit value is further adjusted to account for any deterministic DNBR penalty, such as the fuel rod bow penalty:

[ ] a,c

Indeterminate DNBR Distribution

If the $DNBR_i$ distribution is indeterminate [ ] a,c both normal and non-parametric 95/95 DNBR limits are calculated using the above methods. The more conservative (larger) of the two 95/95 DNBR limits is then chosen as the 95/95 DNBR limit.

Compliance to Approved CHF Correlation DNBR Limit

As described in Reference R-4, a CHF correlation DNBR limit is typically obtained by the ratios of measured CHF values to predicted CHF values (M/P) using the CHF correlation. The mean and standard deviation of the resulting M/P distribution are calculated based on the CHF database. Using an Owen's k-value for the number of measurements, a 95th percentile at the 95th confidence level of the normal distribution, or the 95/95 DNBR limit of the distribution, can be calculated as a predicted to measured CHF (P/M) or DNBR limit as follows:

$$CHF\ DNBR_{95/95}\ Limit\ = \frac{1}{Mean_{M/P} - (k * Standard\ Deviation_{M/P})} \qquad \text{(Eq. R-17)}$$

The approved CHF correlation DNBR limit may include additional conservative adjustments as a bias, so that the resultant limit is more conservative than the value obtained from Equation R-17.

As input to the statistical DNBR limit calculation, the DNBR (P/M) mean and standard deviation values of the CHF correlation must preserve the approved CHF correlation DNBR limit:

$$CHF\ DNBR_{95/95}\ Limit_{Approved}\ =\ CHF\ DNBR\ Mean + 1.645 * CHF\ DNBR\ Sigma \qquad \text{(Eq. R-18)}$$

In the above equation, the CHF DNBR Sigma is defined as [

]a,c

[ ]a,c

The CHF DNBR mean value is obtained by subtracting 1.645*(CHF DNBR sigma) from the approved 95/95 correlation DNBR limit:

$$CHF\ DNBR\ Mean =\ CHF\ DNBR_{95/95}\ Limit_{Approved}\ - 1.645 * CHF\ DNBR\ Sigma \qquad \text{(Eq. R-20)}$$

Summarizing, CHF DNBR values are sampled from a normal distribution based on the standard deviation and the mean values in Equations R-19 and R-20, respectively.

b. *For random variables that are defined by probability distributions that use the mean and standard deviation or upper and lower bound, discuss how each parameter of the distribution is determined.*

Response:

Uncertainties in the system and state parameters as input to the 95/95 DNBR limit calculation are discussed in Section 2.1 of the topical report. Determination of the parameters and their uncertainties in the sample calculation for the Westinghouse-NSSS plant in Attachment A, including its mean, range, and standard deviation is described in the table below. Additional parameter uncertainties are incorporated in Combustion Engineering CE-NSSS plant applications and their values are plant specific. A one-sided normal distribution was conservatively assumed for some parametric uncertainties. The numerical values in the table below are consistent with those listed in Table A-1.

Parameter	Mean	Uncertainty Distribution	Random Uncertainty Range	Standard Deviation
Engineering Enthalpy Rise Hot Channel Factor ($F^E_{\Delta H}$)	1.0			
DNB Correlation	1.031 Adjusted (Eq. R-20)			
Subchannel Code and Modeling	1.0			
Reactor Core Power	1.0			
Reactor Power Radial Peaking Factor ($F^N_{\Delta H}$)	1.635			
Reactor Core Inlet Temperature	556.6			
Reactor Core Inlet Flow (Fraction)	1.0			
Reactor Core Bypass Flow (Fraction)	0.924			
Reactor Pressure	2270			

a,c

c. *Clarify the equations to specify what set of inputs are used to generate the nominal case and how that set of inputs is changed to generate the perturbed case (i.e., how the conditions for the second sub-case are determined).*

Response:

The following parameters are sampled from [ ] a,c in plant-specific ranges for generating the nominal case, or the first sub-case in the DNBR limit calculation:

[ ] a,c

The plant-specific ranges cover the plant DNB-limiting accident statepoints for which the DNBR limit is applied.

Uncertainties in the parameters listed in Section 2.1 of the topical report can be sampled from the uncertainty distributions for generating a perturbed case, or the second sub-case, from the nominal case.

Not all the uncertainties in Section 2.1 of the report are incorporated into all the DNBR limit calculations. The uncertainty input is justified on a plant-specific basis. A ΔDNBR is obtained from the DNBR difference between the perturbed case and the nominal case.

b) Section 3

a. *Provide additional detail on how the fuel damage probability table (denoted the DNB probability distribution in the topical report) is defined. [*

] a,c

Response:

The DNB probability distribution is described below.

The probability of a fuel rod experiencing DNB is calculated as a function of DNBR. The probability density function can be represented by the standard normal (Gaussian) distribution as follows:

$$F(Z) = \frac{1}{\sqrt{2\pi}} e^{-\left(\frac{Z^2}{2}\right)} \quad \text{(Eq. R-21)}$$

where Z = (DNBR - μ)/σ
μ = DNBR mean
σ = Standard deviation.

Integration of the above equation from Z to +∞ gives the probability of a fuel rod experiencing DNB corresponding to Z or DNBR value of the fuel rod. An integrated probability from -∞ to 0 or from 0 to +∞ is 0.5. The DNBR mean (μ) is selected [

]a,c The DNBR mean is greater than or equal to the value obtained using Equation R-16.

Two different standard deviations were used for conservatively maximizing the number of fuel rods in DNB. [

] a,c

a. **The rods-in-DNB calculation in Attachment C is further explained below.**

a,c

a,c

a,c

2) RAI-WTDP-02

Epistemic Uncertainties

What is the impact on the DNBR limit of assuming a [ ] a,c

How does Westinghouse ensure that the DNBR SAFDL is satisfied for every statepoint that the reactor operates at during a cycle, and is not simply satisfied based on the initially assumed set of possible conditions?

Response:

Measurement uncertainties of the reactor design parameters should follow normal distributions. For the DNBR limit determination, a uniform distribution of a parameter is assumed [ ] a,c As compared to the standard deviation of a normal distribution derived from the uncertainty range, which is typically defined as the absolute value of the uncertainty divided by 2 or 1.96, the standard deviation of the uniform distribution would be the uncertainty divided 1.732 (√3) as input to the DNBR limit calculation. The resultant 95/95 DNBR limit is slightly higher with the uniform distribution of the parameter uncertainties, and therefore is more conservative.

A conservative DNBR SAFDL is satisfied for a domain of the core parameters [ ] a,c

3) RAI-WTDP-03

Spatial Sensitivity

Westinghouse's current method for determining the DNBR sensitivity is to use a [

] a,c

Response:

As described in Section 2 of the topical report and supplemental information in response to RAI-WTDP-01, the DNBR sensitivity is determined from the value of ΔDNBR at each condition sampled from [

]a,c in the plant design domain. The ΔDNBR is applied to a sampled DNBR from [

] a,c for determining the 95/95 DNBR limit.

A method sensitivity study was performed by selecting 50 cases [

] a,c where the $DNBR_i$ values were greater than the 95/95 DNBR limit. In each case, additional DNBR calculations were performed to determine the 95/95 DNBR sensitivity based on about 5000 ΔDNBR values from sampling of the parameter uncertainties. The standard deviations of the uncertainties were the same as those in the original calculations in Table A-1, except for some parameters [

] a,c The resultant 95/95 DNBR values of the sensitivity cases are listed in Table R-2 for comparison with the original values. A comparison of the $DNBR_i$ distributions is shown in Figure R-2.

The comparison shows similar $DNBR_i$ distributions between the 95/95 ΔDNBR sensitivity result and the result from the sample calculation in Attachment A. [

] a,c

4) RAI-WTDP-04

Criteria for case exclusion

What criteria are used to ensure that code cases which fail to execute or produce an error are reasonable to exclude from the statistical analysis?

Response:

In the statistical analysis using the Westinghouse Thermal Design Procedure (WTDP), the following criteria are used:

- DNBR calculations are performed within the approved parameter range of a CHF correlation. [

  ] a,c

- Any case not converged in the DNBR calculation, if ever occurred, is not used for generating a ΔDNBR for the $DNBR_i$ distribution. [

  ] a,c

References:

R-1 "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, Westinghouse Electric Company, LLC, June 1984.

R-2 "Statistical Combination of Uncertainties Part 2: Uncertainty Analysis of Limiting Safety System Settings for San Onofre Nuclear Units 2 and 3," CEN-283(S)-P Revision 0, Westinghouse Electric Company, LLC, October 1984.

R-3 "American National Standard Assessment of the Assumption of Normality (Employing Individual Observed Values)," ANSI N15.15-1974, American National Standard Institute, Inc., October 1973.

R-4 "Fuel Safety Limit Calculation Inputs Were Inconsistent with NRC-Approved Correlation Limit Values," Information Notice 2014-1, the United States Nuclear Regulatory Commission, February 2014.

Table R-1 - Result Summary of Locked Rotor Rods-in-DNB Case in Attachment C

a,c

Fuel Rod Power Interval	Number of Fuel Rods	DNBR	DNB Probability	Number Fuel Rods in DNB

Table R-2 – Comparison of ΔDNBR of Sample Case in Attachment A

Sampled DNBR from CHF Correlation Statistics	ΔDNBR of Sample Case in Attachment A	DNBRs from Case in Attachment A	95/95 ΔDNBR Of 5000 Sampled Cases of Uncertainties	DNBRs from 95/95 ΔDNBR of 5000 Cases

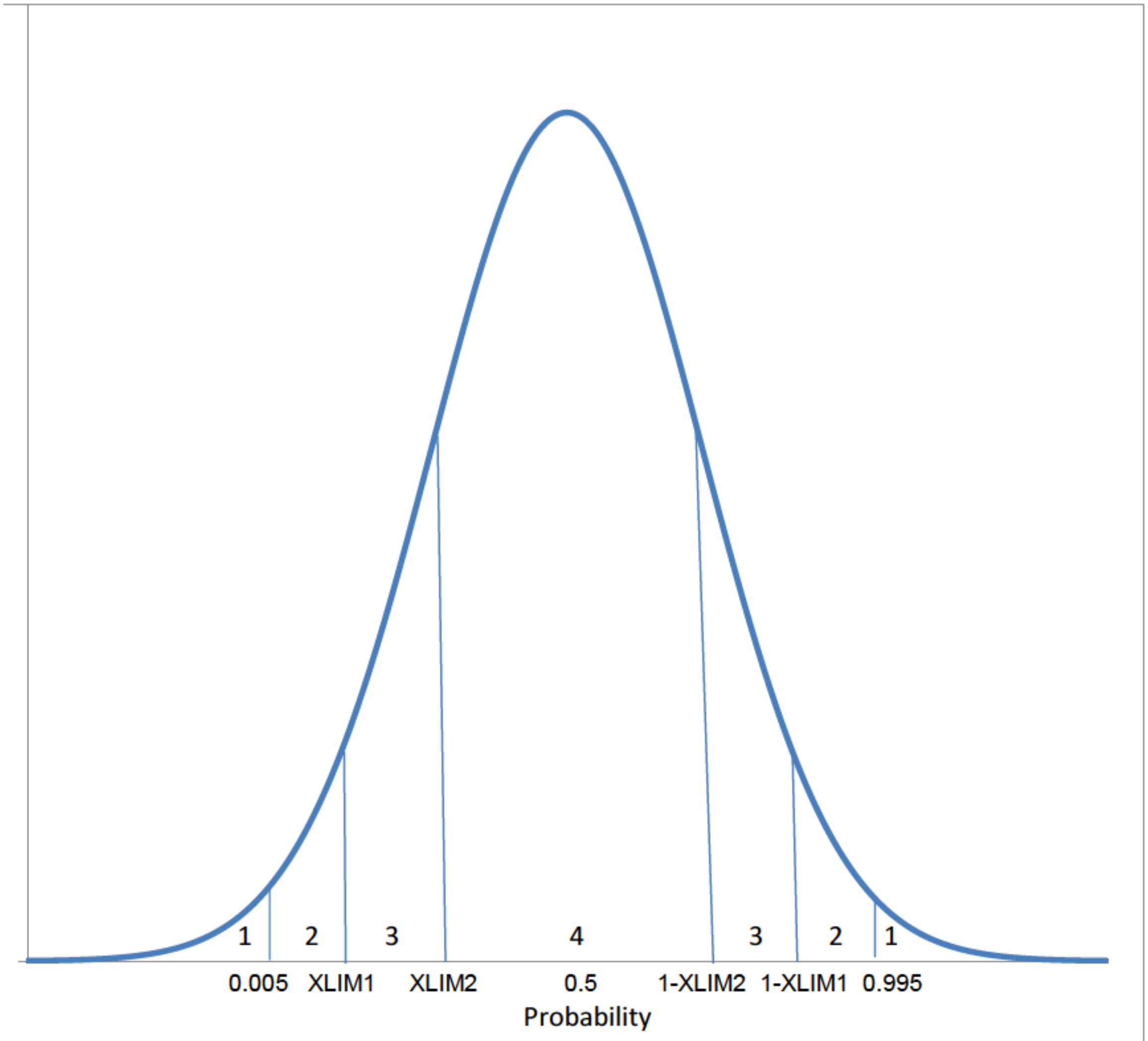
a,c

Table R-2 (continued)

Sampled DNBR from CHF Correlation Statistics	ΔDNBR of Sample Case in Attachment A	DNBRs from Case in Attachment A	95/95 ΔDNBR Of 5000 Sampled Cases of Uncertainties	DNBRs from 95/95 ΔDNBR of 5000 Cases

a,c

Figure R-1 - D-Prime Statistics Probability Regions



[ ]^{a,c}

Figure R-2 – Comparison of $DNBR_i$ Distributions from the Sensitivity Study

a,c