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.

[

]

[

]

[

] 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 **[**

]

[

]

[

]

[

]

[

]

[

]

[

]

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.
[

]

[

]

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*)).

2. Sung, Y., and Singh, S., "WCAP-18240 – Westinghouse Thermal Design Procedure (WTDP)," Revision 0, August 2018 (ADAMS Accession No. ML18242A240 (*Non-Proprietary. Publicly Available*), ADAMS Accession No. ML18242A242 (*Proprietary. Non-Publicly Available*)).

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*)).

5. Lenning, E., U.S. Nuclear Regulatory Commission, letter to Zozula C., Westinghouse Electric Company, "Request for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-18240-P/WCAP-18240-NP, Revision 0, 'Westinghouse Thermal Design Procedure (WTDP)' (EPID: L-2018-TOP-0033)," May 14, 2019, ADAMS Accession No. ML19108A135 (*Cover Letter Available*)," ADAMS Accession No. ML19108A144 (*Proprietary Attachment. Non-Publicly Available*), ADAMS Accession No. ML19108A140 (*Non-Proprietary Attachment. Publicly Available*).

6. Lenning, E., U.S. Nuclear Regulatory Commission, memorandum to Morey, D.C., U.S. Nuclear Regulatory Commission, "Summary of April 1-5, 2019, Closed Audit of the Westinghouse Electric Company Topical Report WCAP18240-P/ WCAP-18240-NP, Revision 0, 'Westinghouse Thermal Design Procedure (WTDP)' (EPID: L-2018-TOP-0033)," July 9, 2019 (ADAMS Accession No. ML19175A160 (*Publicly Available*)).

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*)).

8. U.S. Nuclear Regulatory Commission, "Topical Report Process," LIC-500, Revision 7, October 18, 2018 (ADAMS Accession No. ML18227A063 (*Publicly Available)*).

9. Friedland, A.J., Ray., S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989 (ADAMS Accession No. ML080650330 (*Proprietary. Non-Publicly Available*)).

10. Chelemer, H., Boman, L.H., and D.R. Sharp, "Improved Thermal Design Procedure," WCAP-8567-P-A, February 1989 (ADAMS Accession No. ML081770447 (*Proprietary. Non-Publicly Available*)).

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*)).

13. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A, Rev.01-P-A, ABB Combustion Engineering, May 1988 (ADAMS Accession No. ML18248A129 (*Proprietary. Non-Publicly Available*)).

14. Palo Verde Nuclear Generation Station Units 1, 2 and 3 Updated Final Safety Analysis Report (FSAR), Revision 19

15. 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 (ADAMS Accession No. ML103510190 (*Proprietary. Non-Publicly Available)*).

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