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AREVA MCPR Safety Limit Methodology for Boiling Water Reactors

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AREVA

AREVA MCPR Safety Limit Methodology for Boiling Water Reactors

AREVA NP Inc.

Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document.

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1.0 Introduction

1.1 *Purpose*

This document describes the AREVA NP Inc. (AREVA) methodology for determining the safety limit minimum critical power ratio (SLMCPR) used in assessing thermal margin during operation of a boiling water reactor (BWR). The methodology described in this document is an update of the SLMCPR methodology previously approved by the NRC [1]* and currently supporting BWR operation. The SLMCPR methodology is being updated to incorporate an improved AREVA critical power methodology [2] and a realistic fuel channel bow model developed for the AREVA fuel rod thermal-mechanical methodology [3]. Both the new critical power methodology and the fuel rod thermal-mechanical methodology have been reviewed and approved by the NRC. Incorporating these mechanistic-based methods improves the predictive capability of the SLMCPR methodology while ensuring that adequate conservatisms are maintained.

1.2 *Background*

Operation of a BWR requires protection against fuel damage due to overheating of the cladding during normal operation and anticipated operation occurrences (AOOs). For the purpose of establishing reactor operating limits, damage of the fuel rod cladding is conservatively assumed to occur if the fuel rod experiences boiling transition. Boiling transition is characterized by a degradation of rod surface heat transfer and a subsequent rise in cladding temperature. Test data from multiple sources demonstrate that cladding integrity can be maintained for an extended period of time in boiling transition [4, 5]. Protection of the fuel against boiling transition is a conservative approach that ensures cladding integrity is maintained during normal operation and AOOs.

Because boiling transition is not directly measurable in an operating reactor, it is quantified in terms of assembly critical power. Assembly critical power is defined as the minimum assembly power that results in onset of boiling transition at any location in the assembly. The degree of protection against boiling transition is expressed quantitatively for an individual fuel bundle as the Critical Power Ratio (CPR). CPR is defined as the assembly critical power divided by the operating assembly power. The most limiting assembly in the core has the minimum CPR

* Numbers in brackets refer to references.

(MCPR). Operating and safety limits for the reactor core are established in terms of allowable MCPR.

The acceptance criterion related to boiling transition used as the licensing basis for currently operating BWRs is that the limiting value of MCPR is to be established such that at least 99.9% of the fuel rods in the core are not expected to experience boiling transition during normal operation or AOOs [6]. The AREVA thermal limits methodology [7] is used to ensure compliance with this criterion. The thermal limits methodology and its relation to the SLMCPR methodology is briefly described below.

The SLMCPR is determined such that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition if the core MCPR is greater than or equal to the SLMCPR. The SLMCPR is determined using a statistical analysis that employs a Monte Carlo process that perturbs key input parameters used in the MCPR calculation. The methodology for calculating the SLMCPR is described in this report.

Potential AOOs are identified and evaluated in the Final Safety Analysis Report (FSAR) for each BWR. The impact on CPR during an AOO is characterized by the change (reduction) from the initial steady-state MCPR to the minimum value of MCPR during the event (ΔCPR). AOO analyses are performed with methodologies approved by the NRC for use with the AREVA thermal limits methodology. Currently approved methodologies include [7, 8, 9, 13, 17, 18, 19, and 20]. The largest ΔCPR from the AOO analyses is added to the SLMCPR to establish the operating limit for MCPR (OLMCPR) (see Figure 1-1). Reactor operation is restricted such that the MCPR predicted by the core monitoring system is always greater than or equal to the OLMCPR.

The SLMCPR methodology uses a critical power correlation to calculate CPR for a fuel assembly based on the thermal hydraulic conditions and power distribution in the assembly. Critical power correlations are based on theoretical models defining the functional dependence of critical power on assembly conditions. The coefficients for the correlation are developed based on extensive fuel assembly critical power testing. A critical power correlation is developed (or confirmed) for new fuel designs based on applicable test data. Critical power correlations developed for AREVA fuel designs and used in the SLMCPR methodology described in this report include the SPCB correlation [10] and the ACE correlation [2, 11]. Other

NRC-approved critical power correlations may be used with the AREVA SLMCPR methodology. Non-AREVA fuel that is co-resident during a fuel design transition is evaluated using a critical power correlation qualified by the process previously approved by the NRC [12].

Because the SLMCPR methodology provides the ability to evaluate the number of rods in boiling transition for a specified MCPR, the method can be used to assess the number of rods that fail due to cladding overheating during infrequent events or accidents.

1.3 ***Summary of New Features***

The improvements incorporated in the SLMCPR methodology described in the document are summarized below. More detailed descriptions are provided in Section 2.0.

- Full Implementation of the ACE Critical Power Correlation

  The ACE critical power correlation uses more detailed power distribution data than required for previous correlations. As described in the approved ACE correlation topical report [2], the correlation was implemented in a conservative manner within the currently approved SLMCPR methodology. The SLMCPR methodology has been revised to obtain more detailed power distribution data from calculations performed using the MICROBURN-B2 computer code [13].

  The ACE correlation uses a [ ] that improves the prediction of critical power. With new fuel designs, the expectation is to continue to use the same form of the ACE correlation with coefficients based on test data applicable for the fuel design. Once ACE is approved for application to a new fuel design, its use within the SLMCPR methodology is easily implemented. The same form of the ACE correlation is retained for the ATRIUM 10XM fuel design [11] that is planned for introduction in the US.

- Incorporate Realistic Fuel Channel Bow Model

  A realistic model to predict fuel channel bow was developed as part of the recently approved AREVA fuel rod thermal mechanical methodology for BWRs [3]. The realistic channel bow model is implemented in the MICROBURN-B2 computer code. The SLMCPR methodology has been revised to use the MICROBURN-B2 computer code [

]

The current SLMCPR methodology [1] conservatively assumed that each and every assembly in the core experienced the most adverse channel bow condition simultaneously and applied the most conservative uncertainty of the most limiting rod of each fuel type to all rods of that fuel type. The use of MICROBURN-B2 [

]

- Expanded Coupling with MICROBURN-B2

The implementation of ACE and the realistic channel bow model required additional data from MICROBURN-B2. The expanded coupling with MICROBURN-B2 facilitates other improvements to the SLMCPR methodology.

[

] within the updated SLMCPR methodology.

[

] This approach provides the best representation of the conditions in each assembly in the core and is consistent with the core monitoring system.

In summary, this document describes the process for the determination of the SLMCPR. The process includes the calculation of MCPR using the same method as used by the core monitoring system. The SLMCPR is established using a design basis power distribution and a statistical convolution of the measurement and calculation uncertainties associated with the determination of MCPR. The impact of fuel channel bow on power distribution is included in the calculations. The SLMCPR, in conjunction with reactor transient and event analyses, establishes an operating limit on MCPR that ensures fuel rod cladding integrity is preserved during normal operation and anticipated operational occurrences.

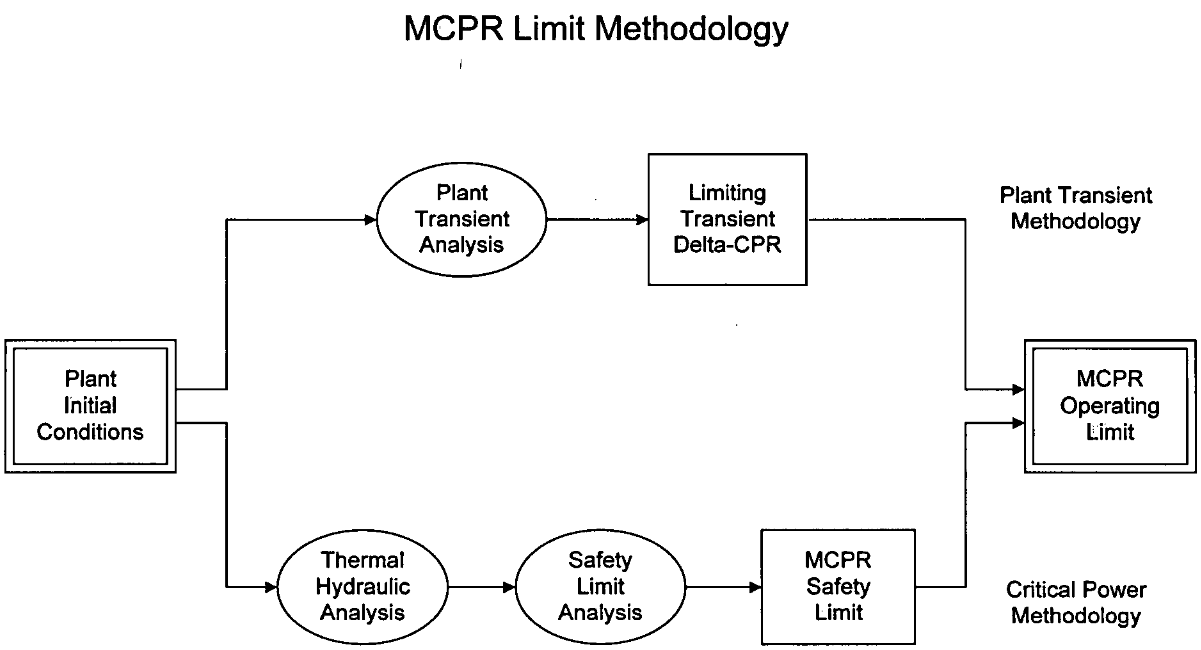

Figure 1-1 Calculation Process for Determination of Operating Limit

2.0 New Methodology Features and Models

2.1 *ACE/ATRIUM-10 Critical Power Correlation*

In addition to the SPCB critical power correlation [10], the AREVA critical power methodology incorporates the ACE/ATRIUM-10 critical power correlation [2]. The ACE/ATRIUM-10 correlation is [

] Application of ACE to new fuel designs will be described in topical reports submitted to the NRC for approval. The ACE correlation application to the ATRIUM 10XM fuel design is currently being reviewed by the NRC [11].

The ACE critical power correlation uses more detailed power distribution data than previous correlations. As described in the ACE topical report [2], the correlation was implemented in a conservative manner within the currently approved SLMCPR methodology. The SLMCPR methodology has been revised to obtain more detailed power distribution data from calculations performed using the MICROBURN-B2 computer code [13].

The ACE correlation uses a [ ] to characterize the local peaking factor effect on the assembly critical power [2]. MICROBURN-B2 performs detailed cycle depletion calculations and uses pin power reconstruction to determine [

]

2.2 *Channel Bow*

The MCPR safety limit is sensitive to power distribution. As identified in Section 2.1, the ACE correlation uses a [

] Because of the dependence of the power distribution on channel bow, the MCPR safety limit is impacted by channel bow.

As channel bow involves the deformation of the fuel channel, [

] Fuel rods closest to the fuel channel experience larger changes in local moderation and power than interior rods. [

]

2.2.1 Channel Bow Model

The channel bow model is presented in Reference [3]. It was implemented in the MICROBURN-B2 core simulator to account for channel bow effects on the realistic power histories used in the statistical fuel rod methodology.

The main cause of bow in the fuel channel is the differential irradiation growth between two opposing sides of the channel. The fuel channel accommodates the differential axial growth by bending such that the convex side is the longer side. Neglecting the elastic stresses and strains, the bending is assumed to be a fully permanent deformation.

[

]

The MICROBURN-B2 core simulator computes the maximum channel bow magnitude for each assembly, consistent with the correlations described in Reference [3]. The assembly maximum bow is then used to determine the axial bow profile as a quadratic function with its peak at the axial mid-plane and with zero values at the top and bottom of the channel. The MICROBURN-B2 calculation [

]

2.2.2 Application of Channel Bow Model for SLMCPR Methodology

The assessment of channel bow within the SLMCPR methodology is accomplished by a

[

]

The MICROBURN-B2 channel bow model and uncertainty described in Section 2.2.1 is applicable for typical reload cycles. For non-typical situations, the channel bow model will be applied in a conservative manner [

] Non-typical situations include abnormal channel bow situations, transition cores, and new channel designs.

Abnormal channel bow does not appear to have a simple cause. Two mechanisms appear to influence abnormal channel bow. Early control blade exposure is responsible for shadow corrosion induced hydride formation and consequently contributes to fuel channel bow during the first cycle of operation. However, this mechanism is insufficient to completely explain abnormal bow, especially for channels with large bows. A second mechanism considers the growth behavior under conditions of high fluence and hydrogen content.

The effects of abnormal channel bow are being addressed through [

]

To address the impact of abnormal channel bow for safety limit analysis, evidence of abnormal channel bow from the utility would be the initiating event. Such evidence could come from direct measurements of channel bow or from indirect indicators such as slow control blade settle times. An evaluation of the information received would provide a basis for determining the method to be used for evaluating the impact on safety limit. [

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Transition cores may introduce co-resident channels that [

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The development of new channels could result in channels that [

]

2.2.3 Power Distribution Uncertainty Due to Channel Bow

Channel bow impacts both the [

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2.2.3.1 [ ]

[

]

2.2.3.2 [ ]

[ ]

2.2.3.3 [ ]

[ ]

[ ]

2.3 *MICROBURN-B2 Interface*

In addition to the interface between MICROBURN-B2 and [ ] The Monte Carlo process is implemented by the SAFLIM3D computer program. This section describes the various interfaces of MICROBURN-B2 with SAFLIM3D.

SAFLIM3D receives MICROBURN-B2 information [ ] The other information obtained by SAFLIM3D from MICROBURN-B2 is described in this section. Figure 2-1 provides a diagram of the interface among the three computer codes.

2.3.1 Core Power Distribution

[ ] For the determination of the nominal operating conditions used in the Monte Carlo process, the [ ] consistent with the monitoring system. Following the determination of the nominal operating conditions, SAFLIM3D uses the [ ].

The core nodal power distribution is the nodal power at each axial node of each assembly in the core. Consequently, the [ ]

2.3.2 Core Flow and Enthalpy Distribution

Through the interface between SAFLIM3D and MICROBURN-B2, [ ]. During the determination of the nominal operating conditions, MICROBURN-B2 is executed using core boundary conditions to determine the assembly conditions that place the core MCPR at the assumed MCPR safety limit. During the Monte Carlo process, the [ ]

2.3.3 [ ]

[ ]

2.4 *Methodology Comparison*

Major features that are different between the currently approved SAFLIM2 methodology and SAFLIM3D are presented in Table 2-1. Each feature is discussed in the following subsections.

2.4.1 [ ]

[

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2.4.2 [ ]

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2.4.3 [ ]

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2.4.4 [ ]

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2.4.5 [ ]

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2.4.6 [ ]

[

]

2.4.7 [ ]

[

]

Table 2-1 Methodology Comparison

[ ]

Figure 2-1 Major Computer Codes for Safety Limit Analysis

3.0 MCPR Safety Limit Calculation Process

The calculation of the MCPR safety limit involves the statistical convolution of reactor system measurement and core monitoring uncertainties using a Monte Carlo procedure. The CPR is calculated using approved critical power correlations and is consistent with the POWERPLEX®-III CMSS calculation of MCPR. Approved critical power correlations are used directly to determine if a rod is in boiling transition and all rods in the core predicted to be in boiling transition are summed. A non-parametric tolerance limit is used to determine the total number of BT rods with 95% confidence. At least 99.9% of the rods in the core are not expected to experience boiling transition when the core minimum CPR is at or above the SLMCPR.

The MCPR safety limit analysis is performed each cycle using core and fuel design specific characteristics for the cycle. Figure 3-1 provides an overview of the MCPR safety limit calculation process.

3.1 *Monte Carlo Technique*

The Monte Carlo analysis is a statistical technique to determine the distribution function of a parameter that is a function of random variables. Each random variable is characterized by a nominal (mean) value, standard deviation, and distribution function. The variables are perturbed using a random number obtained from a pseudo-random number generator. The random number in combination with a stratification method (Section 3.4) and the standard deviation results in a random value to be added to the nominal value. As applied to the determination of the MCPR safety limit, the distribution that is created from the Monte Carlo process is the number of rods in boiling transition. The process to create this distribution is repeated a large number of times.

3.2 *MCPR Calculation Uncertainties*

The MCPR safety limit is determined by a statistical convolution of the uncertainties associated with the calculation of thermal margin. The set of uncertainties which form the basis of the statistical convolution was established by the relative sensitivity of all the parameters used in the

MCPR calculation. These parameters include both reactor system measurement and core monitoring uncertainties.

3.2.1 Reactor System Measurement Uncertainties

Reactor system measurement uncertainties include those associated with the measurement of feedwater flow rate, feedwater temperature, core pressure, and total core flow rate.

Table 3-1 summarizes the reactor system measurement uncertainties and representative values for each parameter. Actual values used in the SLMCPR analysis are provided by the customer.

3.2.2 Core Monitoring Uncertainties

Core monitoring uncertainties include those associated with the calculation of assembly flow rate, radial power, nodal power, nodal rod power, and critical power (correlation uncertainty). The power related uncertainties are derived from MICROBURN-B2 [13]. The process used in validating MICROBURN-B2 as documented in [13] provides the bases for the uncertainties used in the safety limit analysis.

The value for the radial power uncertainty and [ ] used in the SLMCPR analysis depends on the core monitoring system, lattice type (C or D), and the number of recirculation loops operating. Increased values for the radial and nodal power uncertainties are used to support extended LPRM calibration intervals and LPRM/TIP out of service conditions. The uncertainty models described in [13] are used to determine these increased uncertainties.

The assembly flow rate uncertainty depends on the fuel design and the core monitoring system. Components of the assembly flow rate uncertainty include pressure drop, hydraulic compatibility, bypass flow rate, and nuclear peaking. Factors entering the pressure drop calculation of an assembly include the two-phase flow model, assembly single-phase loss coefficients, orifice loss coefficients, and manufacturing and in-service variations. Assembly flow rate uncertainty is related more to flow distribution than to absolute flow. The absolute flow uncertainty for the core is addressed by the total core flow uncertainty.

The CPR uncertainties are derived from the ACE critical power correlation [2, 11], and the SPCB critical power correlation [10].

Table 3-2 presents the core monitoring uncertainties consistent with two-loop operation with POWERPLEX®-III CMSS. The actual values used in the cycle SLMCPR analysis are provided in the cycle specific reload analysis report.

Power distribution uncertainties due to channel bow are obtained [

] as described in Section 2.2.3.

3.3 *Nominal Conditions*

The SLMCPR calculation process starts with the determination of the nominal conditions which provide the "mean" value for each variable perturbed in the Monte Carlo process.

3.3.1 Nominal Operating Conditions

The nominal operating conditions include the system pressure, total core flow, feedwater temperature, and feedwater flow rate. The design basis core nodal power and nodal rod power distributions used in determining the nominal operating conditions [

]

The calculation of the nominal operating conditions is an iterative calculation. The system pressure, total core flow rate, and feedwater temperature are held constant while the feedwater flow rate is adjusted. The system pressure, feedwater flow rate, and feedwater temperature define the total core power and the core inlet enthalpy (inlet subcooling). These boundary conditions with the design basis core power distribution) are input to MICROBURN-B2 [

]

The CPR for high power assemblies (radial peaking factor > 1.0) is calculated and the core minimum CPR (MCPR) is identified. If the core MCPR does not equal the assumed MCPR safety limit, the feedwater flow rate is adjusted and the process is repeated. When the core MCPR equals the assumed MCPR safety limit, the nominal operating conditions (system pressure, total core flow, feedwater flow, and feedwater temperature) are defined.

3.3.2 Nominal Power Distribution

The initial MCPR distribution of the core is a major factor affecting how many rods are predicted to be in boiling transition. The MCPR distribution of the core depends on the fuel assembly design and the power distribution in the core. The safety limit methodology is performed with a design basis power distribution that conservatively represents expected reactor operating states which could both exist at the MCPR operating limit and produce a MCPR equal to the MCPR safety limit during an anticipated operational occurrence.

[

]

3.4 *Monte Carlo Procedure*

3.4.1 Perturbed Conditions

The uncertainty in a measured (or calculated) input is determined by the distribution of the measurements (or calculations) about the nominal or "mean" value. These distributions are assumed to be normal. The uncertainty is characterized by one standard deviation.

The normal distribution is modeled by a statistical stratification method. [

]

3.4.2 Monte Carlo Trials

The Monte Carlo calculation determines the number of rods predicted to be in boiling transition for a set of randomly perturbed parameters associated with the reactor system, core monitoring, and channel bow uncertainties. The calculation is repeated a number of times (trials) to create a probability distribution for the number of rods in BT.

The number of trials performed is sufficient such that each new trial does not cause a significant change in the running average of the number of rods predicted to be in boiling transition. It has been observed that [

]

The Monte Carlo calculations for a single trial are described below. Figure 3-3 provides an overview of the Monte Carlo calculation process. A detailed representation of the process is provided in Appendix B.

3.4.2.1 Core Calculations

The operating conditions that result in the limiting assembly at the SLMCPR (Section 3.3) provide the nominal or "mean" values for total core flow, system pressure, feedwater flow rate and feedwater temperature. The nominal nodal power distribution used in the Monte Carlo calculation is obtained from the MICROBURN-B2 cycle depletion calculation [

]

The nominal operating conditions (total core flow, system pressure, feedwater flow, and feedwater temperature) are randomly varied according to the applicable probability distribution and uncertainty. The nodal power distribution is perturbed to account for the radial power uncertainty, [

] The nodal power distribution is renormalized to maintain total core power.

[

]

3.4.2.2 Assembly Calculations

The [ ] from the MICROBURN-B2 calculation represent nominal or "mean" values for each assembly.

The active channel flow for each assembly is perturbed to account for the assembly flow rate uncertainty. The active channel flow distribution is renormalized to maintain total core active channel flow. The axial enthalpy distribution is recalculated based on the perturbed flow.

[

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3.4.2.3 Rod Calculations

[

]

All of the fuel rods in each assembly are evaluated using the appropriate critical power correlation (ACE or SPCB) to determine the number of rods predicted to be in boiling transition. The rods predicted to be in boiling transition for each assembly are summed over the entire core to determine the total number of rods in boiling transition for each trial.

3.4.3 Evaluation of BT Rods

For each Monte Carlo trial, the total number of rods predicted to be in BT from all assemblies is determined as described above. The probability distribution function for rods predicted to be in BT is based on the results from all of the Monte Carlo trials.

The number of rods predicted to be in BT is determined based on the number of trials and is independent of the probability distribution function because a non-parametric procedure [16] is applied.

The expected number of rods in boiling transition for the base state is defined as the maximum number of rods in boiling transition at the 50% probability level with 95% confidence. As the sample size approaches infinity, the 50/95 value approaches the population median. The median is the preferred measure of central tendency when analyzing expected number of rods in boiling transition since the distribution of rods in boiling transition is strongly skewed.

The basis for the MCPR safety limit is that the expected number of rods in boiling transition be less than 0.1% of the rods in the core with a 95% confidence.

If the expected number of rods in boiling transition is less than the 0.1% criterion, the assumed SLMCPR is acceptable. If the 0.1% criterion is exceeded, the assumed SLMCPR is increased and the calculation is repeated.

Table 3-1 Typical Reactor System Uncertainties

Parameter	Standard Deviation (Percent of Nominal)
Feedwater Flow Rate	1.76
Feedwater Temperature	0.76
Core Pressure	0.5
Total Core Flow Rate	2.5

Table 3-2 Core Monitoring Uncertainties

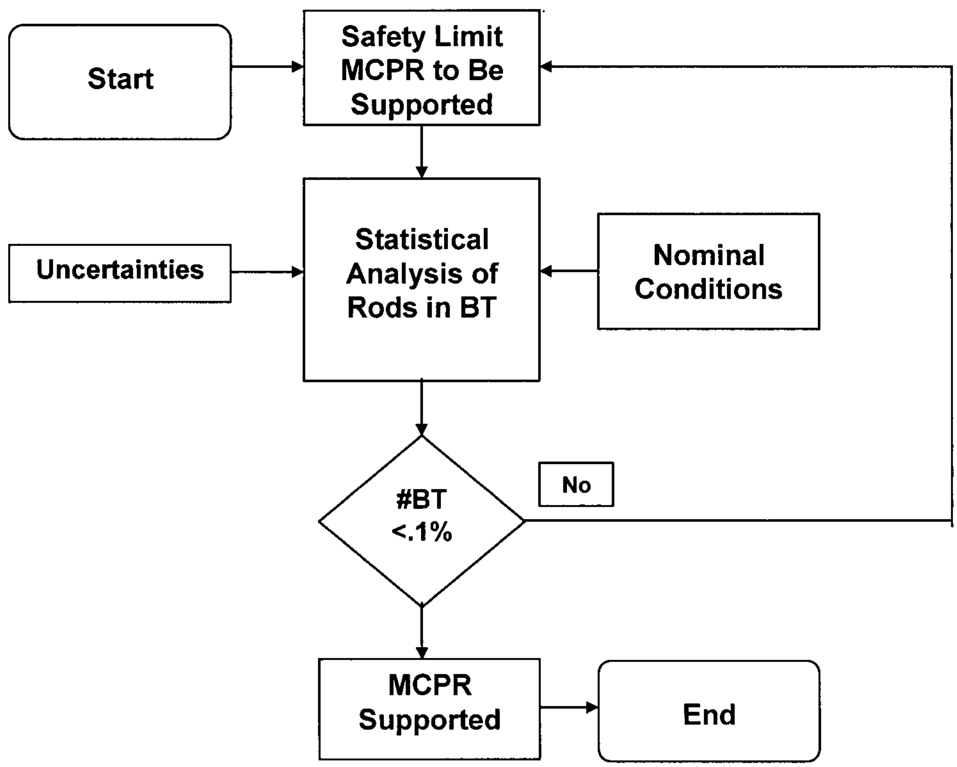

Figure 3-1 Flow Diagram for Safety Limit Analysis

Figure 3-2 Stratified Normal Distribution Curve

Figure 3-3 Monte Carlo Trial

4.0 Methodology Assessment

4.1 *Impact of Channel Bow*

An assessment was performed to determine the impact of channel bow on the SLMCPR. This assessment compares the channel bow impact on the SLMCPR with the current SAFLIM2 methodology [1] and the updated SAFLIM3D methodology described in this report. Three different correlations and both C-Lattice and D-Lattice BWRs were used in this assessment. The assessment used the same statepoint from the cycle (exposure, operating conditions) with both methodologies. SAFLIM3D receives [

] The conservatism present in the SAFLIM2 methodology for the ACE correlation is apparent. The channel bow penalty with the updated SAFLIM3D methodology is slightly improved for SPCB and significantly improved for ACE.

4.2 *Impact of* [ ]

SAFLIM2 uses the core average axial power shape for each assembly and hydraulic demand curves while SAFLIM3D uses [

] . The impact of using [

] The assessment was performed by simulating the current SAFLIM2 methodology with SAFLIM3D. Uncertainties that are not available in SAFLIM2 are set to zero in SAFLIM3D. [

] The same state point – cycle, exposure, core pressure and flow from SAFLIM2 are used in SAFLIM3D. Assessments are made using SPCB and ACE critical power correlations and the resultant SLMCPR values are compared in Table 4-2.

The conclusion drawn from Table 4-2 is that the impact of [

]

4.3 *Sample Applications*

Three sample calculations were performed using the revised methodology described in this report. These calculations with SAFLIM3D use the MICROBURN-B2 generated information, typical uncertainties, [ ] channel bow uncertainties, and cycle-specific depletion calculations. The MCPR safety limits presented in Table 4-3 represent the value that would be obtained for the particular sets of conditions considered. These sample calculations are representative of the results expected when applying the revised SLMCPR methodology described in this report.

Table 4-1 Effect of Channel Bow on Safety Limit

Table 4-2 Effect of Assembly Specific Axial Power Shape and MICROBURN-B2 Thermal Hydraulic Conditions

Table 4-3 Sample Application

5.0 References

1. ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation November 1990.

2. ANP-10249PA Revision 1, "ACE/ATRIUM-10 Critical Power Correlation," AREVA NP Inc., September 2009.

3. BAW-10247PA Revision 0, "Realistic Thermal Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP Inc., April 2008.

4. NUREG/CR-0517 (TREE-1327), S.L. Seiffert and R.R. Hobbins, "Oxidation and Embrittlement of Zircaloy-4 Cladding from Higher Temperature Film Boiling Operation," EG&G Idaho, Inc. April 1979.

5. NUREG-0562, R. Van Houten, "Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout," U.S. Nuclear Regulatory Commission, Washington D.C., June 1979.

6. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, U.S. Nuclear Regulatory Commission.

7. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987.

8. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

9. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.

10. EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," AREVA NP Inc., September 2009.

11. ANP-10298(P) Revision 0, "ACE/ATRIUM 10XM Critical Power Correlation," AREVA NP Inc., December 2008.

12. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-resident Fuel," Siemens Power Corporation, August 2000.

13. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2," Siemens Power Corporation, October 1999.

14. M. Griffiths, et. al., "Accelerated Irradiation Growth of Zirconium Alloys," Zirconium in the Nuclear Industry, Eighth International Symposium, ASTM STP1023, p 658, 1989.

15. Conover, W. J., 1971. "Practical Nonparametric Statistics," John Wiley & Sons, New York, pp 99-105.

16. P.N. Somerville, "Tables for Obtaining Non-Parametric Tolerance Limits," Annals of Mathematical Statistics, Vol 29, No. 2, June 1958, pp 599-601.

17. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983.

18. XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

19. XN-NF-825(P)(A) Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain," Exxon Nuclear Company, October 1986.

20. ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.

Appendix A Nomenclature

Appendix B Monte Carlo Procedure

[ ]