ANF-524(NP)(A)
REVISION 2

ANF-524(NP)(A)
SUPPLEMENT 1
REVISION 2

ANF-524(NP)(A)
SUPPLEMENT 2

ADVANCED NUCLEAR FUELS CORPORATION

# ADVANCED NUCLEAR FUELS CORPORATION CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

NOVEMBER 1990



A Siemens Company

**ADVANCED NUCLEAR FUELS** CORPORATION

ANF-524(NP)(A)
Revision 2

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Supplement 1
Revision 2

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| ANF-524(NP)(A)<br>Revision 2 | ADVANCED NUCLEAR FUELS CORPORATION CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS |
| ANF-524(NP)(A)<br>Supplement 1<br>Revision 2 | ADVANCED NUCLEAR FUELS CORPORATION CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS: METHODOLOGY FOR ANALYSIS OF ASSEMBLY CHANNEL BOWING EFFECTS |
| ANF-524(NP)(A)<br>Supplement 2 | NRC CORRESPONDENCE |

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

August 8, 1990

Mr. R. A. Copeland, Manager
Reload Licensing
Advanced Nuclear Fuels Corporation
P. O. Box 130
Richland, Washington 99352-0130

Dear Mr. Copeland:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT ANF-524(P) REVISION 2, "ANF CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS"

The staff has completed its review of the Topical Report ANF-524(P) Revision 2, "ANF Critical Power Methodology for Boiling Water Reactors," submitted by the Advanced Nuclear Fuels Corporation (ANF) by letters dated April 11 and November 30, 1989. Additional information was submitted by letters March 17, April 9, May 3, and June 7, 1990.

ANF-524(P) Revision 2 describes the Advanced Nuclear Fuels Corporation (ANF) critical power methodology that was developed to assess compliance with minimum critical power ratio (MCPR) operating limits in boiling water reactors (BWRs).

The report includes the MCPR calculational procedure and the system measurement and calculational uncertainties to determine a MCPR safety limit that protects 99.9 percent of the fuel rods from boiling transition. The MCPR safety limit with a delta-CPR from transient analyses establishes a value on the range of reactor operating parameters consistent with established criteria for nominal and transient reactor operation. This methodology utilizes a Monte Carlo procedure to account for the measurement and calculational uncertainties. A design-basis power distribution is assumed that conservatively represents the expected reactor power distribution.

The proposed methodology includes the determination of the effect of channel box bowing on the thermal margin. This procedure is also statistical in nature and is benchmarked to actual reactor data.

We find the application of the ANF critical power methodology acceptable for use for BWR fuels under the limitations delineated in the associated U.S. Nuclear Regulatory Commission (NRC) technical evaluation. The evaluation defines the basis for acceptance of this topical report.

We do not intend to repeat our review of the matters found acceptable as described in ANF-524(P) Revision 2 when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the application of ANF-524(P) Revision 2.

In accordance with procedures established in NUREG-0390, we request that the Advanced Nuclear Fuels Corporation publish an accepted version of this topical report within 3 months of receipt of this letter. The accepted version shall include an "A" (designating "accepted") following the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, the Advanced Nuclear Fuels Corporation and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

AC Thadani

Ashok C. Thadani, Director
Division of Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
ANF-524(P) Revision 2
Evaluation

ENCLOSURE

SAFETY EVALUATION FOR THE TOPICAL REPORT ANF-524(P) REVISION 2,
"ANF CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS"

## 1.0 INTRODUCTION

By letters dated April 11 and November 30, 1989, the Advanced Nuclear Fuels Corporation submitted the Topical Report ANF-524(P) Revision 2 for NRC review (Refs. 1-3). Additional information was submitted by letters dated March 19, May 3, and June 7, 1990 (Refs. 4-6). The critical power ratio (CPR) methodology is the approach used by ANF to determine the margin-to-thermal limits for boiling water reactors (BWRs). Revision 2 includes the modifications required for consistency with the recently proposed (Ref. 4) ANF critical heat flux correlation. Revision 2 also includes the application of CASMO-3G/MICROBURN-B (Ref. 8). (NRC review of the ANF MICROBURN-B code system is outside the scope of the present review.) ANF-524(P) Revision 2 documents the details of the revisions associated with ANF and MICROBURN-B and the methodology used to incorporate the effect of fuel channel bowing on thermal margin.

Brookhaven National Laboratory (BNL) has been the staff consultant in this review under FIN No. A-3868.

## 2.0 SUMMARY OF THE TOPICAL REPORT

The topical report provides the basis for the ANF methodology for determining the operating safety limit for minimum critical power (SLMCPR) which ensures that 99.9 percent of the fuel rods are protected from boiling transition. This determination is carried out by a series of Monte Carlo calculations in which the variables affecting the probability of boiling transition are varied randomly and the total number of rods experiencing boiling transition is determined for each Monte Carlo trial. The ANFB CPR correlation depends on the core coolant pressure, channel mass velocity, planar enthalpy, local peaking function (FEFF), radial and axial power, and channel geometry. The major calculational uncertainties are determined by the local power and channel flow. The power distribution uncertainty is given in Reference 8 and the channel flow calculational uncertainty is given in Chapter 3 of the subject topical report. The determination of the channel flow calculation uncertainty and the Monte Carlo determination of the SLMCPR are the principal elements of the CPR methodology and are summarized in the following sections.

## 2.1 Core Flow Distribution

The core flow distribution and individual bundle flows are required for input to the ANFB correlation and the determination of the bundle critical power. The calculation of the core flow distribution assumes a set of flow resistances connected in series and parallel linear network. The core and the core bypass are represented by a set of parallel resistances from the lower plenum to the upper plenum. The pressure drop across each of the parallel flow paths is calculated by the channel flow model using the hydraulic resistance of the channel and the channel flow. The bundle-wise flow distribution is determined iteratively by imposing the condition that the parallel flow paths have a common pressure drop.

In the channel flow model, the bundle pressure drop is calculated (by the momentum equation) as the sum of the fuel component hydraulic losses. The two-phase component losses are determined by two-phase friction multipliers. The void fraction model used in the channel flow model is based on a mechanistic description of two-phase separated flow and includes a subcooled void model.

The ANF two-phase flow model has been validated by comparison to pressure drop measurements for BWR prototypic rod bundles. Comparisons were made for a range of axial power shapes, mass velocity, pressure, inlet enthalpy, quality and assembly power. The ANF models reproduced the measured pressure drops to within about 3 percent (Ref. 9).

## 2.2 Determination of the Safety Limit MCPR

The safety limit maximum critical power ratio (SLMCPR) ensures that 99.9 percent of the fuel rods are protected from boiling transition. The SLMCPR is determine' by a statistical procedure in which the important measurement and modeling parameters are varied randomly and the number of fuel rods experiencing boiling transition $N_{BT}$ for each random statepoint is used to determine the distribution of $N_{BT}$. The SLMCPR is the core MCPR which results in a 50-percent probability (at the 95-percent confidence level) of 0.1 percent of the rods experiencing boiling transition. The SLMCPR for a given state is determined iteratively with parameters that are not varied randomly at their most limiting values.

The statistical distributions for the uncertainties in the feedwater temperature and flow, core pressure and flow, assembly flow and power, and the ANFB CPR correlation are used to determine the random variations in these variables. A nonparametric procedure is used to determine the distribution-free value for the SLMCPR.

## 2.3 Evaluation of the Effects of Channel Bowing

The ANF procedure for determining the effects of fuel channel bowing consists of three steps: (1) the prediction of the channel deflection for a given fuel burnup, (2) calculation of the increase in local rod power as a function of bowing displacement, and (3) the determination of the reduction in thermal margin CPR for a given channel bow. Available channel bow data is used to determine the batch-average bow and its standard deviation as a function of fuel exposure.

The calculation of the increase in local powers is performed in a typical four-bundle geometry with CASMO-3G (Ref. 8). The bowed and unbowed power maps are calculated versus fuel exposure for each lattice design. The calculated bowed power maps are used directly in the calculation of the reduction in the core MCPR as a result of the nominal channel bow. The statistical uncertainty component of the effect of channel bowing is estimated using the bowing uncertainty together with conservative maximum power-to-bow sensitivities. This uncertainty is then included in the standard ANF Monte Carlo safety limit analysis (Section 2.2).

## 3.0 TECHNICAL EVALUATION

The critical power methodology described in ANF-524(P) is the procedure used by ANF to determine the thermal margin SLMCPR for boiling water reactors. Revision 2 of ANF-524(P) documents (1) the changes that will be required for consistency with the new ANFB critical power correlation and (2) the power distribution uncertainties that result from the application of MICROBURN-B. Supplement 1 of Revision 2 describes the assessment of fuel channel bowing. The review of the initial report resulted in a series of questions which were transmitted to ANF in References 7-9. The evaluation of the CPR methodology topical report ANF-524(P) Revision 2, and Supplement 1 and the responses to staff questions (Refs. 4-6 and 10-12) are summarized in the following sections.

### 3.1 Uncertainties Used in the Determination of the SLMCPR

In the new ANF critical power methodology the values for calculational and measurement uncertainties have been revised and are given in Table 5.1 of ANF-524(P) Revision 2. In the new CPR analysis, the inlet enthalpy is determined from a core heat balance, but the inlet temperature uncertainty is not required. The inlet enthalpy uncertainty is determined by randomly varying the core flow and feedwater temperature and flow input to the heat balance in the SLMCPR calculation. As in the previous XN-3 analysis, the uncertainties in the flow areas and hydraulic parameters are included in the flow uncertainty, and the uncertainty in the heated and wetted perimeters is included in the ANFB correlation uncertainty.

The uncertainty in the additive Q-constants that account for the CPR dependence on spacer design is evaluated in ANF-1125(P) Supplement 1. The uncertainties depend on fuel design and have been evaluated for the ANF-8x8, ANF-9x9, KWU-9x9 and KWU/ANF-9X9-IX fuel designs. The ANF-8X8 and ANF-9x9 fuel requires the largest ℓ-constant uncertainty ±0.02 in units of local peaking.

The MICROBURN-B uncertainties are described in detail in Reference 8, which is presently under separate NRC review. The MICROBURN-B uncertainties used in a specific SLMCPR application should be based on the NRC approved values. The application of the new SLMCPR methodology to mixed cores will be accommodated by increasing the bundle flow uncertainty, as is presently done for the XN-3 correlation.

The new CPR methodology includes both the reduced ANFB CPR correlation and MICROBURN-B power distribution measurement uncertainties. ANF has indicated in Response 11 (Ref. 4) that no plants will be licensed that use the new MICROBURN-B methodology together with the XN-3 CPR correlation. However, the new ANFB correlation may be used with the XTGBWR power distribution methodology and, in this case, the radial bundle power uncertainty will be increased to the presently approved value for XTGBWR.

It is concluded that the treatment of uncertainties in the new ANF CPR methodology is acceptable.

### 3.2 ANFB CPR Correlation Bias

Comparison of the predicted and measured critical powers indicates that ANFB predicts a database mean of 1.003. In Response 4 (Ref. 9) ANF indicates that this nonconservative bias will be offset by two additional conservatisms in the ANF CPR methodology. First, a 0.01 CPR conservatism has been included as a result of a systematic ANF overprediction (by a factor of two) of the number of rods experiencing boiling transition. Second, the bundle power is increased by 10 percent in the calculation of transient delta-CPR.

It is concluded that sufficient conservatism is available in the ANF CPR methodology to offset the mean bias in the ANFB correlation.

### 3.3 Selection of Statepoint Variables

The SLMCPR is determined by the condition that 99.9 percent of the rods avoid boiling transition. The calculated number of rods that satisfy this condition is sensitive to the initial reactor statepoint variables (power level, pressure, flow, power and flow distribution, etc.). In Response 7 (Ref. 4), ANF has stated that worst-case conditions that put the maximum number of rods closest to the SLMCPR have been assumed in the ANF calculation of the SLMCPR.

The ANF selection of the reactor statepoint variables is, therefore, acceptable.

### 3.4 ANFB Correlation Uncertainty

The uncertainty in the ANFB CPR correlation is introduced into the SLMCPR calculation as an uncertainty in the local peaking additive constants. The calculated sensitivity of the ANFB predicted CPR to variations in the constants is minus two; that is, an increase in  of 0.01 results in a decrease in CPR of 0.02. The standard deviation between the ANFB calculated CPR and the measured database is 0.03 for ANF-8x8 fuel, for example, which is less than the 0.04 CPR uncertainty resulting from the assumed ± 0.02 $\ell$-constant uncertainty (Table 5.1). In Response 10 (Ref. 4), ANF has stated that the CPR correlation uncertainty is always bounded by the CPR uncertainty introduced by the additive $\ell$-constants.

The uncertainty in the $\ell$-constants is assumed to have a normal distribution. The relationship between the additive $\ell$-constant and CPR is linear (to a very good approximation), and the observed normal distribution of the CPRs given in Response 3 (Ref. 4) demonstrates that the $\ell$-constant distribution may be accurately represented with a normal distribution.

It is, therefore, concluded that the treatment of the $\ell$-constant uncertainty is acceptable.

### 3.5 Assessment of the Effects of Fuel Channel Bowing

The ANF assessment of the effect of channel bowing on the SLMCPR is given in Supplement 1 of ANF-524(P). The evaluation of the major concerns associated with the bowing of BWR fuel channels and the assessment provided in Supplement 1 is summarized in the following sections.

#### 3.5.1 Determination of Channel Bow

As the fast neutron fluence exposure to the zircaloy fuel channels increases with fuel burnup, the channels undergo irradiation growth and are deflected from their nominal core positions. The increased growth of the channel walls adjacent to the narrow water gaps (regions of relatively high fast flux) deflects the fuel channels away from the narrow-narrow gap and toward the wide-wide gap. Fuel channels in fast flux gradients (e.g., on the core periphery) also experience channel deflection; however, this bowing may be either diagonal or parallel to the channel faces.

The distribution of channel bow (mean and standard deviation) depends on the cycle core loading. This dependence results from (1) the dependence of the channel strain on the bundle fluence and initial bow and (2) the geometrical dependence of the bowing on the arrangement of the fuel bundles in the core. ANF calculates the core mean channel bow using a conservative bounding core-loading analysis. This analysis employs a four-bundle supercell in which the fresh CPR limiting bundle is adjacent to two exposed and highly bowed fuel bundles and is diagonal to a once-or twice-burned fuel bundle. The locations of the bundles in the four-bundle supercell are determined by the expected (i.e., nominal) increase in the four control rod water gaps. The uncertainty because of deviations from this nominal geometry is determined using a Monte Carlo procedure (Response 1, Ref. 6).

The effect of channel bowing is considered statistically and is described by the expected nominal value of channel bow and its standard deviation. These statistical parameters are determined using one of two base methods. When a large amount of exposure-dependent channel bowing data is available, a regression fit versus exposure is constructed and the required mean and standard deviation are determined. When sufficient data is not available to determine a reliable correlation versus exposure, the channel data is grouped by exposure and a groupwise mean and standard deviation are determined.

Since reused second-lifetime channels are expected to have substantially larger bow than first-lifetime channels (and most likely represent a different underlying population), the applicability of these statistical procedures to second-lifetime channels will require additional validation. Also, the methodology assumes in justifying the applicability of the XN-3 and ANFB correlations that the CPR limiting bundle does not have large bow (Response 3, Ref. 5). This is not generally true for second-lifetime fuel channels.

### 3.5.2 Effect of Fuel Channel Bowing on Local Power Peaking

The effect of channel bowing, and the resulting increased water gaps, on the pin power peaking is calculated with the CASMO-3G lattice physics code (Ref. 8). The CASMO-3G model includes the four fuel bundles surrounding the control rod channel water gaps. The boundary conditions imposed on the outer boundary of this supercell affect the sensitivity of the power peaking to changes in the water gap thickness. ANF uses a periodic boundary condition on the four-bundle cell which is conservative (Response 7, Ref. 5).

The sensitivity of the local power peaking to increased water gap thickness depends on exposure and void fraction. The void dependence is conservatively accounted for by calculating the power sensitivity at high in-channel void fraction (typically 70 to 75 percent) where the sensitivity is maximum (Response 3, Ref. 6). ANF neglects the flattening of the bowing perturbation with increased fuel burnup and calculates the effect assuming an instantaneous increase in water gap thickness at selected exposure points. This method is correct for fresh fuel bundles and is conservative for bundles with non-zero fuel burnup.

### 3.5.3 Effect of Fuel Channel Bowing on the Critical Power Ratio

The core fuel loadings for certain plants include fuel from multiple fuel vendors. In this case, the determination of the core MCPR requires the calculation of the bowing penalty for non-ANF fuel. In Response 10 (Ref. 5) ANF has indicated that conservative estimates of the effect of channel bowing on local power peaking will be used in determining the CPR penalty for non-ANF fuel.

### 3.5.4 Fuel Misloading/Misorientation

Both the core average channel bow and the sensitivity of CPR to bow may be affected by a fuel misloading or misorientation. The sensitivity of CPR to bowing, however, is to a good approximation determined by the location of the fuel rod in the bundle and is independent of the fuel bundle orientation. In addition, in the fuel channel analysis the channels are oriented to undergo the maximum average bow and, consequently, a misorientation will result in an increase in CPR.

In the fuel misloading, the maximum delta-CPR penalty occurs when a highly exposed low-powered fuel bundle is replaced by a fresh high-powered bundle. In this case, therefore, the cell-average bow and associated CPR bowing penalty will be less than the values determined in the standard bowing analysis. It is assumed that the fresh high-powered fuel bundle is not contained in a reused second-lifetime channel.

### 3.5.5 Effect of Fuel Channel Bowing on Power Distribution Monitoring

Fuel channel bowing reduces the water gap dimensions at the location of the traversing incore probe (TIP) detectors used to monitor the core power distribution. The geometry change may be considered as a core-wide reduction in the gap thickness and a random or statistical variation in the gap about this average gap reduction. The effect of the core-wide reduction in the gap is removed by the normalization of the power distribution to the core thermal power. The statistical variation in the gap dimensions, however, introduces a random spatial variation into the TIP response. This detector response variation contributes to the measured TIP asymmetry and is included in the SLMCPR analysis as part of the XTGBWR (Ref. 13) and MICROBURN-B (Ref. 8) radial power distribution uncertainties.

The ANF methodology for evaluating the effects of channel bowing is therefore considered acceptable.

## 4.0 TECHNICAL POSITION AND LIMITATIONS

The ANF critical power methodology described in the ANF-524(P) Revision 2 and the channel bowing Supplement 1 and in the ANF responses provided in References 4 and 5 has been reviewed. The ANF methodology has been found to be acceptable for performing safety limit MCPR analyses under the following limitations:

1. The NRC-approved MICROBURN-B power distribution uncertainties should be used in the SLMCPR determination (Section 3.1).

2. Since the ANFB correlation uncertainties depend on fuel design, in plant-specific applications the uncertainty value used for the ANFB additive constants should be verified (Section 3.4).

3. The CPR channel bowing penalty for non-ANF fuel should be made using conservative estimates of the sensitivity of local power peaking to channel bow (Section 3.5.3).

4. The methodology for evaluating the effect of fuel channel bowing is not applicable to reused second-lifetime fuel channels (Sections 3.5.1 and 3.5.4).

## 5.0 REFERENCES


1. Letter, R.A. Copeland (ANF) to Director of NRR, "Transmittal of ANF-524(P), Revision 2," dated April 11, 1989.

2. Macduff, R.B., and N.F.Fausz, "EXXON Nuclear Critical Power Methodology for Boiling Water Reactors," XN-NF-524(P), (A) Revision 1, dated November 1983.

3. Letter, R.A. Copeland (ANF) to Director of NRR, "Transmittal of Supplement 1 to ANF-524(P), Revision 2," dated November 30, 1989.

4. Letter, R.A. Copeland, ANF to R.C. Jones, NRC, "Transmittal of Additional Information on Topical Report ANF-524(P), Revision 2," dated March 19, 1990.

5. Letter, R.A. Copeland, ANF to R.C. Jones, NRC, "Transmittal of Additional Information on Topical Report ANF-524(P), Revision 2," May 3, 1990.

6. Letter, R.A. Copeland, ANF to R.C. Jones, NRC, "Additional Information Regarding Loss of Thermal Margin Caused by Channel Box Bow," June 7, 1990.

7. Zerressen, K.W. and R.B. Macduff, "ANFB Critical Power Correlation," ANF-1125(P), Supplement 1, dated April 1989.

8. Brown, O.C., and T.H. Timmons, "Advanced Nuclear Fuels Methodology for Boiling Water Reactors - Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," XN-NF-80-19(P), Volume 1, Supplement 3, dated February 1989.

9. "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," XN-NF-79-59, November 1983.

10. Letter, R.C. Jones, NRC to R.A. Copeland, ANF, "Request for Additional Information Regarding Topical Report ANF-524(P), Revision 2," March 5, 1990.

11. Letter, R.C. Jones, NRC to R.A. Copeland, ANF, "Request for Additional Information Regarding Topical Report ANF-524(P), Revision 2," March 30, 1990.

12. Letter, R.C. Jones, NRC to R.A. Copeland, ANF, "Request for Additional Information Regarding the Topical Report ANF-524(P), Revision 2, Supplement 1," June 6, 1990.

13. "Exxon Nuclear Methodology for Boiling Water Reactors Neutronic Methods for Design Analysis," XN-NF-90-19(P) (A), Volume 1, Supplements 1 and 2, Exxon Nuclear Company, Richland, Washington 99352, March 1983.

**ADVANCED NUCLEAR FUELS** CORPORATION

ANF-524(NP)(A)
Revision 2
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# ADVANCED NUCLEAR FUELS CORPORATION CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

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April 1989

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## Summary of Changes for Revision 2

| Section | Description of Changes |
|---|---|
| 1.0 | Updated to reflect use of ANFB correlation. |
| 2.0 | Minor typographical |
| 3.0 | Minor typographical |
| 4.0 | Updated to reflect change from XN-3 CPR correlation to the ANFB CPR correlation |
| 5.0 | Updated to reflect use of ANFB correlation in safety limit analysis and to define new uncertainties related to updated neutronics methodology |
| 6.0 | No changes |
| 7.0 | Updated reference list with current document revisions |

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# ADVANCED NUCLEAR FUELS CORPORATION CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

## 1.0 INTRODUCTION

This document describes the Advanced Nuclear Fuels Corporation (ANF) methodology used for determination of thermal margin of a boiling water reactor. Revision 2 supercedes the previously accepted Revision 1 of this report. This methodology has been revised to incorporate the new ANFB critical heat flux correlation in determination of the core safety limit minimum critical power ratio (MCPR). The methodology for evaluating operating limits is also presented. The objective of establishing operating limits is the preservation of the fuel clad integrity. The methodology uses a series of conservative assumptions which over estimate the probability of a breach of fuel clad integrity. Therefore, the reactor operating limit provides a level of protection in excess of established requirements.(1,2)*

The thermal margin determination depends upon hydraulic and thermal calculations. Reactor coolant flow distribution is calculated from a set of experimentally or calculationally determined assembly hydraulic characteristics and an experimentally verified two-phase flow model. Following the calculation of core flow distribution, a critical power correlation can be used to determine if fuel rods experience boiling transition. The safety limit is derived by statistically convolving hydraulic and thermal calculational uncertainties with measurement uncertainties associated with reactor instrumentation. The safety limit provides an appropriate level of core protection from boiling transition. The incremental change in margin due to reactor system transients is added to the safety limit to establish the limit for normal reactor operations.

For purposes of establishing the reactor operating limit, damage of the fuel rod clad is conservatively assumed to occur if the fuel rod experiences

*Numbers in brackets refer to references.

boiling transition. Considerable data exist to show cladding integrity can be maintained for an extended period of time in boiling transition.[3,4] Boiling transition is characterized by a degradation of rod surface heat transfer and a subsequent rise in clad operating temperatures. Because boiling transition is not a directly measurable quantity in an operating reactor, it is quantified in terms of the critical power ratio (CPR) which is derived from a critical power correlation. The critical power correlation is an empirical representation of the assembly coolant conditions at which boiling transition has been experimentally detected. The critical power ratio is defined as the assembly power required to produce boiling transition divided by the operating assembly power. The safety and operating limits of a reactor core are expressed by the allowable MCPR.

The reactor system transients and events which are plausible for a BWR are classified according to expected or observed frequency of occurrence in accordance with established standards.[5] These transients and events are analyzed with methodology described elsewhere[6],[17] to determine their impacts upon fuel rod performance, which are characterized by a change in the MCPR (ΔCPR) from steady-state during the transient. The largest ΔCPR due to the reactor system transients or events is added to the MCPR safety limit to establish the MCPR operating limit. Reactor operation is restricted such that the observed MCPR is always greater than or equal to the MCPR operating limit.

The level of core protection which has been established for BWRs[15] is that 99.9% of the fuel rods in the reactor core are expected to avoid boiling transition when the reactor core is operating at the MCPR safety limit. Derivation of the MCPR safety limit is performed with a design basis power distribution which conservatively represents expected reactor power distributions for normal reactor operation and as a consequence of reactor system transients. With ANF's revised methodology, a deterministic approach is incorporated in evaluating the expected number of rods in boiling transition for a specified safety limit MCPR.

In summary, the procedure used to determine the MCPR of a BWR and to establish the MCPR safety limit is described within this document. The determination of MCPR includes a calculation of the distribution of reactor coolant flow which provides data for the critical power calculation. The MCPR safety limit is established with a design basis power distribution and a statistical convolution of the measurement and calculational uncertainties associated with the determination of MCPR. The MCPR safety limit, in conjunction with reactor transient and event analyses, establishes an operating limit on MCPR which in turn limits the range of reactor operation. The MCPR limit on reactor operation provides for the maintenance of fuel rod cladding integrity during normal operation and reactor system transients or events.

## 2.0 SUMMARY

This document describes the methodology used by ANF to establish and to assess compliance with MCPR operating limits in a boiling water reactor (BWR). The steps of the MCPR calculational procedure are presented and verification is provided as appropriate. The reactor system measurement uncertainties are statistically convolved with MCPR calculational uncertainties to determine a MCPR safety limit which protects 99.9% of the fuel rods in the reactor core from boiling transition (See Figure 5.1). The MCPR safety limit, incorporated with a ΔCPR from transient analyses described elsewhere,(6),(17) establishes a limit on the range of reactor operating parameters which is consistent with established criteria for nominal and transient reactor operation.

A MCPR safety limit is generated for a set of reactor system measurement and calculational uncertainties by a Monte Carlo procedure. The generation of the MCPR safety limit is based upon a design basis power distribution which conservatively represents expected reactor power distributions. Hence, the MCPR safety limit presents a conservative limit with regard to protection of the reactor core from boiling transition. The MCPR operating limit may be reactor core specific and hence is established on a core/cycle specific basis.

## 3.0 CORE FLOW DISTRIBUTION

The calculation of the core flow distribution determines the flow to each assembly and to the bypass region, and provides the hydraulic information necessary for calculating the assembly critical power ratios. The core flow distribution is calculated from a hydraulic model of the reactor core. The physical components of the reactor core (support plates, assemblies, and assembly components, etc.) are represented in the hydraulic model by flow resistances connected in series and in parallel. The hydraulic model provides a mathematical representation of the pressure and coolant flow distributions which result from the physical configuration of the reactor core.

The flow resistances in the reactor core are determined by analytical techniques or by experimental programs or a combination of both. For example, the single-phase flow resistances of the orifice, lower tie plate, bare rod region, spacers, and upper tie plate of the ANF fuel, have been determined by an experimental program. The two-phase flow resistances of appropriate components are determined from the single-phase loss coefficients and a set of two-phase flow models.(11) The prediction of pressure drop by a combination of single-phase loss coefficients and two-phase flow models has been experimentally verified.(11)

Because the assembly flow is constrained by the placement of metal liners (channels) around each fuel assembly, the flow through each assembly depends upon the resistance to flow encountered. The core is hydraulically comprised of a number of parallel flow paths with an equal pressure drop existing across all paths between points of common communication. Since the fuel assemblies communicate only at the upper and lower plenum, the pressure drop across each assembly is equal from the lower plenum to the upper plenum. The recirculating flow rate and the assembly hydraulic resistance, in conjunction with the hydraulic resistance of the bypass region, determines the core pressure drop. A schematic diagram of the flow resistances of the core hydraulic model is shown in Figure 3.1. The core is comprised of parallel resistances across the core support plate, the bypass region, and from the lower plenum to the upper plenum.

The results of the calculation of core flow distribution are the bypass flow fraction and the distribution of coolant flow and enthalpy throughout the reactor core. For the determination of the safety limit, the relationship between assembly flow rate and assembly power is determined for each fuel type.

3.1 Channel Flow Model

The channel flow model is used to determine the pressure drop across each flow path identified in the core hydraulic model and is used to determine the core pressure drop.

The calculation of pressure drop is based upon the momentum equation for separated flow[11] and may be written as:

The pressure gradients defined by relation (3.1) are numerically integrated over the fuel length to determine the overall pressure drop. The numerical integration procedure which is used reduces the sensitivity of the calculated pressure drop to the nodalization and thereby results in an accurate calculation of the pressure drop as described in Reference 11.

The pressure drop and, therefore, the flow rate in each assembly is dependent upon the hydraulic losses, operating power, and axial power distribution present in that assembly. The hydraulic losses in the assembly are characterized by single-phase hydraulic tests or analytical models. The assembly power is manifested in the pressure drop through the two-phase flow models.

A negative axial offset is indicative that greater than 50% of the assembly power is deposited in the lower half of the assembly, and is the usual situation for an unrodded assembly. The variation of assembly flow as a function of the axial offset is shown in Figure 3.2 for assemblies in a typical central orifice zone at typical BWR operating conditions.

### 3.2 Two-Phase Flow Models

Because the void fraction model is used to determine an average fluid density, which in turn is used to determine gravitational and other pressure drop components, it is an implicit part of the methodology for calculating pressure drop.

Experimental pressure drop data was acquired in tests conducted for ANF by Columbia University.(8,9,11) The data represents pressure drop data taken during diabatic two-phase flow conditions with rod bundles prototypic of BWR fuel designs. The pressure drop in assemblies with both uniform and non-uniform axial heat flux profiles has been determined over a wide range of operating conditions

3.2.1 Void Fraction

The void fraction correlation used in the pressure drop calculation is based upon a mechanistic description of two-phase separated flow and incorporates the effects of integral and relative phase slip. The void fraction correlation is a function of the pressure, mass velocity, flow quality, and rod surface heat flux within an assembly. A subcooled void model is included in the void fraction correlation to include the effects of thermal non-equilibrium.

3.2.2 Two-Phase Friction Multipliers

### 3.3 Hydraulic Test and Analysis

The single-phase fuel assembly hydraulic loss coefficients are determined by analytical procedures or an experimental test program. In the case that hydraulic characteristics are determined experimentally, a portable hydraulic test facility (PHTF) is used to measure the single-phase pressure losses associated with both ANF fuel and existing fuel. This eliminates the potential for experimental uncertainty due to the use of different test facilities and testing procedures.

Pages 11-14 have been deleted.

## 4.0 CRITICAL POWER CALCULATION

The calculation of assembly thermal margin is based upon the core flow distribution analysis and is completed by the assembly critical power calculation. The assembly critical power corresponding to a particular reactor operating state is determined from the ANFB critical power correlation.(10) The ANFB correlation is an empirical representation of the set of assembly coolant conditions at which boiling transition has been experimentally detected. The figure of merit in the assessment of thermal margin is the critical power ratio (CPR). Thus, an assembly with an absolute CPR of 1.30 could experience a 30% increase in power before it is expected that boiling transition will occur on the most limiting rod within that assembly.

### 4.1 ANFB Critical Power Correlation

The ANFB critical power correlation is used to determine the assembly power required to produce boiling transition for a particular reactor and fuel assembly operating state. The correlation was developed from a large body of experimental data encompassing a wide variety of coolant conditions and assembly geometry.

The ANFB correlation has also been compared to transient boiling transition data. Assembly power and flow were varied in a manner typical of anticipated transients until boiling transition occurred. It was determined that the ANFB correlation conservatively predicted the time to boiling transition, indicating that use of the ANFB correlation to predict critical power during transient operating conditions is conservative.

The ANFB correlation has also been used to predict the number of rods experiencing boiling transition for representative test data. The number of rods in an assembly calculated to be in boiling transition, as predicted by the ANFB correlation, is found to be a conservative prediction of the total number of rods in boiling transition for a particular data point.

### 4.2 Critical Power Analysis

The calculation of assembly thermal margin is performed following a thermal hydraulic calculation which determines the flow distribution within the core. The flow distribution is determined by the core flow analysis described in Section 3.0.

Pages 17-18 have been deleted.

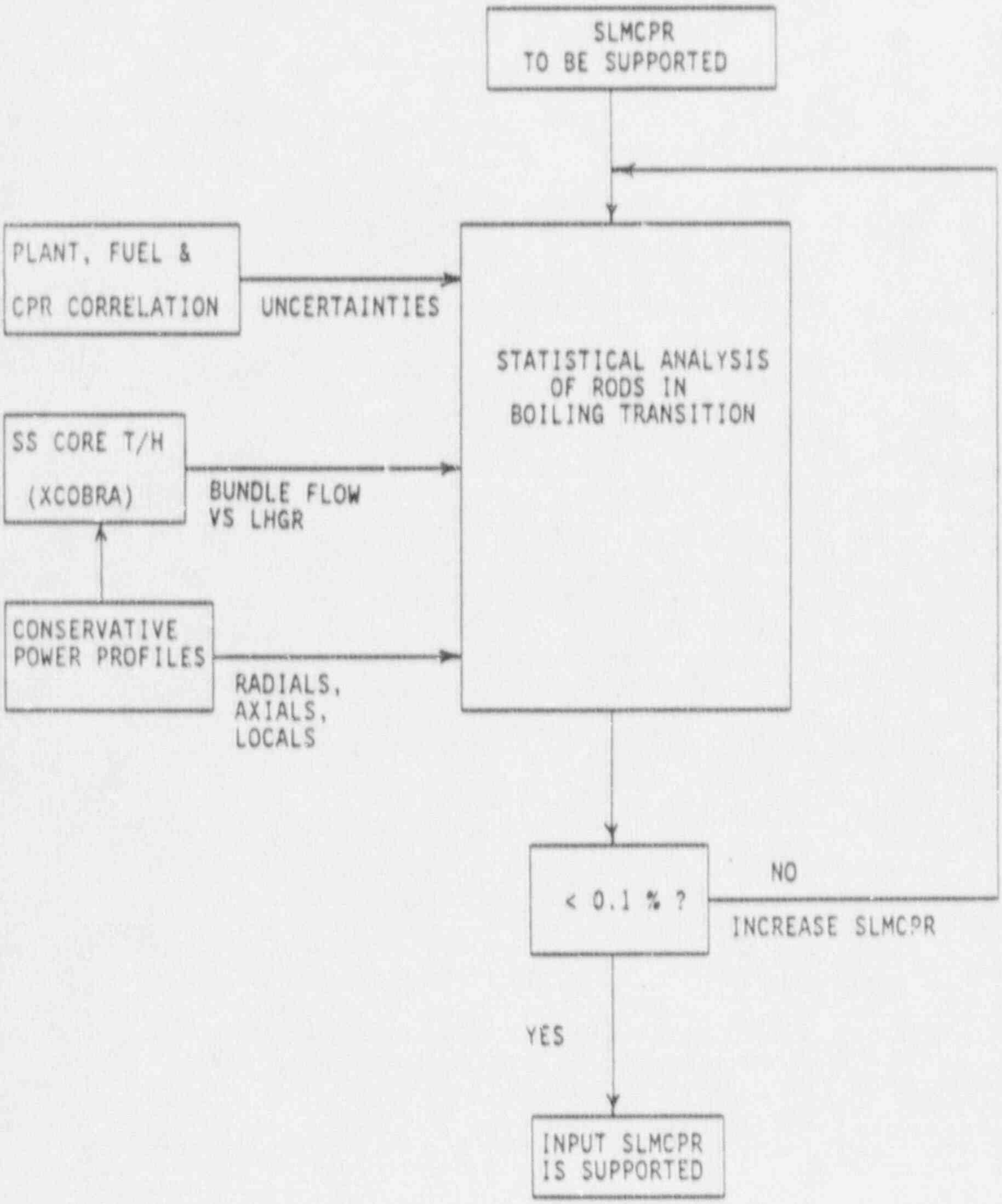
## 5.0 GENERATION OF THE MINIMUM CPR SAFETY LIMIT

The minimum CPR (MCPR) safety limit is established to protect the core from boiling transition during both normal operation and anticipated operational occurrences. When the reactor core is operating at or above the MCPR safety limit, at least 99.9% of the rods in the core are expected to

The MCPR safety limit established by this procedure is an appropriate limit for protecting the core during normal operating conditions and anticipated operational occurrences. The MCPR safety limit derived by the procedure presented provides a credible limit for MCPR monitoring, because the MCPR monitoring procedure was simulated in generating the safety limit.

TABLE 5.1 TYPICAL UNCERTAINTIES USED TO GENERATE MCPR SAFETY LIMIT

FIGURE 5.1 FLOW DIAGRAM FOR SAFETY LIMIT ANALYSIS



## 6.0 GLOSSARY

AO = axial offset

$-\frac{dP}{dZ}$ = pressure gradient

D = hydraulic diameter

f = bare rod friction factor

$g, g_c$ = gravitational constants

G = mass velocity

$K_C$ = component loss coefficient

$K_{exp}$ = irreversible loss coefficient for sudden expansion

L = fraction of power generated in lower half of assembly

U = fraction of power generated in upper half of assembly

$\Delta Z$ = calculational increment

$\alpha$ = void fraction

$\nu_m$ = specific volume for momentum transfer

$\bar{\rho}$ = average density

$\rho_g$ = density of saturated vapor

$\rho_f$ = density of saturated fluid

$\rho_\ell$ = density of liquid phase

$\sigma$ = area ratio

$\phi^2_{BR}$ = bare rod two-phase friction multiplier

$\phi^2_C$ = component two-phase friction multiplier

## 7.0 REFERENCES


1. Title 10, Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities," Government Printing Office, Washington, D.C.

2. Title 10, Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation," Government Printing Office, Washington, D.C.

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

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10. ANF-1125(P), "ANFB Critical Power Correlation," Revision 0, April 1989.

11. XN-NF-79-59, "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," November 1983.

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13. XN-NF-80-19(P), Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors, Volume 1, Neutronics Methods for Design and Analysis, Supplement 3, February 1989.

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**ADVANCED NUCLEAR FUELS** CORPORATION

ANF-524(NP)(A)
Supplement 1
Revision 2
Issue Date: 11/30/89

# ADVANCED NUCLEAR FUELS CORPORATION CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

## METHODOLOGY FOR ANALYSIS OF ASSEMBLY CHANNEL BOWING EFFECTS

Prepared by

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November 1989

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# TABLE OF CONTENTS

# ADVANCED NUCLEAR FUELS CORPORATION CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

## METHODOLOGY FOR ANALYSIS OF ASSEMBLY CHANNEL BOWING EFFECTS

### 1.0 INTRODUCTION

This supplement to ANF-524 describes the application of Advanced Nuclear Fuels Corporation (ANF) safety limit methodology for determining the thermal margin impacts of channel bowing. ANF has developed a method that accurately assesses the thermal margin impacts of channel bow through inclusion in the core safety limit calculation (Reference 1). This report describes the calculation procedure, results from a benchmark calculation, and results of four example calculations.

Channel bowing is best modeled with a statistical procedure. For a given reactor core, the mean channel bow for a given batch of fuel can be determined based on the exposure history of the fuel channels. Channel deflection data can also be used to determine the uncertainty (standard deviation) in the expected (mean) value of channel bowing.

## 2.0 SUMMARY

ANF has developed a calculational procedure to determine the thermal margin impacts of BWR channel bowing. This procedure is an extension of the standard safety limit methodology and is based on the statistical nature of channel bowing. A benchmark to the Oskarshamn-2 fuel failure event and four example cases comparing channel bow effects based on differing input assumptions are presented.

## 3.0 CALCULATION PROCEDURE

The calculation procedure used to determine the thermal margin impacts of channel bowing can be broken into three areas:

1) Determination of expected channel bowing and uncertainty for a given channel exposure using channel deflection data.
2) Calculation of local peaking effect caused by channel bowing by employing a lattice physics code such as CASMO-3G.
3) Assessment of the thermal margin (critical power) effects of channel bowing with a statistical procedure within the standard ANF safety limit analysis technique (Reference 1).

These procedural steps will now be described.

### 3.1 Determination of Expected Channel Bowing

Channel bow data is available from several sources and covers GE, CARTECH, and ASEA channels. EPRI (References 2, 3, 4) has published data for GE and CARTECH channels, KWU has an extensive database of CARTECH channel deflection measurements, and the Swedish Regulatory Authority has issued a table of channel bow as a function of exposure for ASEA channels. This information is used to determine the expected channel bow throughout the cycle of operation that is being evaluated.

### 3.2 Calculation of Local Peaking Effects

After tabulating the channel bow versus exposure data, it is necessary to develop a lattice physics model to simulate the channel bow phenomenon. The CASMO-3G code is currently used for this calculation (Reference 5).

3.3 Statistical Analysis of Thermal Margin Impacts

The safety limit statistical analysis reflects how the core is monitored by the core monitoring system and includes the channel bow effect.

## 4.0 CALCULATION RESULTS

The ANF procedure for analyzing channel bow includes the nominal magnitude of bowing expected based on the exposure history of the channels

The safety limit for the reactor may be increased slightly due to the bow and the transient delta-CPR is unaffected, so the operating limit may be increased the same magnitude as the safety limit increase.

### 4.1 Example Calculations

The results of four example calculations requested by the NRC staff are presented below for a reference D-lattice plant using ANF fuel in ASEA channels. The channel bow data used for these calculations was prescribed by the Swedish Regulatory Authority. The four cases considered are:

1) Nominal case without effects of channel bow.

2) Typical channel bow case with two twice burned and one once burned assemblies in a super-cell with a fresh bundle.

3) Case with two twice burned and one once burned assemblies in a super-cell with a fresh bundle without accounting for bowing depletion effects. A standard depletion calculation is performed for nominal geometry and restart cases with expected channel bow are run at each exposure point of interest. This is more severe than case 2 because rods with high peaking due to channel bow are not allowed to undergo accelerated depletion as the core is burned.

4) Case with three twice burned assemblies in a super-cell with one fresh bundle without bow depletion effects. This is an atypical worst case and produces the largest MCPR effect.

All of the example cases assume that fuel channels are not being reused. The results of the calculations are presented in Table S1.

TABLE S1 CHANNEL BOW CALCULATION RESULTS

| Case Number | Channel Bow MCPR Effect |
|---|---|
| 1 | |
| 2 | |
| 3 | |
| 4 | |

Channel bow and the associated MCPR effect are slightly smaller for a typical C-lattice plant due to smaller bundle flux gradients resulting from the uniformity in core water gaps. The results in Table S1 therefore bound similar results for a C-lattice plant. These results indicate the current approach is acceptable.

4.2 Methodology Benchmark

A benchmark to the Oskarshamn-2 fuel failures was carried out to insure that the ANF methodology is appropriate and conservative. To perform the benchmark, the Oskarshamn reactor was modeled with the ANF methodology

The results are based on standard safety limit modeling procedures

## 5.0 REFERENCES


1. ANF-524(P), "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," Revision 2, April 1989.

2. NP-2483, "An Assessment of BWR Fuel Channel Lifetimes," Electric Power Research Institute, July 1982.

3. NP-3937M, "Extending BWR Fuel Channel Lifetimes: Development of Predictive Deformation Models, Electric Power Research Institute, April 1985.

4. NP-4225M, "Fuel Channel Lifetime: Expansion of Statistical Predictive Models," Electric Power Research Institute, September 1985.

5. XN-NF-80-19(P), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors," Volume 1, Supplement 3, February 1989.

6. "Analysis of the Dryout Incident in the Oskarshamn 2 Boiling Water Reactor," Paper presented at the European Two-Phase Flow Group Meeting, Paris, May 1989.

7. BX-89-20, "Oskarshamn 2 Fuel Failures," ABB Atom Incorporated, June 1989.

**ADVANCED NUCLEAR FUELS** CORPORATION

ANF-524(NP)(A)
Supplement 2
Issue Date: 11/30/90

NRC CORRESPONDENCE

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RAC:023:90
March 19, 1990

Mr. Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Jones:

Subject: Transmittal of Additional Information on Topical Report ANF-524(P), Revision 2

Ref:

1. Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "Request for Additional Information Regarding the Topical Report ANF-524, Rev. 2," dated March 5, 1990.

2. ANF-524(P), Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, April 1989.

3. Letter, R. A. Copeland (ANF) to Director of NRR (USNRC), "Transmittal of ANF-524(P), Revision 2," April 11, 1989, RAC:023:089.

Attached is the additional information that was requested (Reference 1) on Advanced Nuclear Fuels Corporation's critical power methodology for BWRs (Reference 2). ANF considers the information contained in the Reference 2 topical report to be proprietary. The affidavit included with the Reference 3 letter provides the information required by 10 CFR 2.790(b) to support withholding of this topical report from public disclosure.

If there are questions, or if further information is needed, please contact Larry Nielsen at (509)375-8358.

Sincerely,

Bob Copeland

R. A. Copeland
Manager, Reload Licensing

/skm
cc: Dr. L. Lois (USNRC)
Attachment

# ATTACHMENT

## RESPONSES TO NRC QUESTIONS ON SAFETY LIMIT METHODOLOGY

PREPARED BY: S. E. STATE

March 1990

Question 1

In addition to (1) the use of the ANFB Critical Power Correlation and associated uncertainties, (2) the reduced MICROBURN-B power distribution measurement uncertainties, and (3) the inclusion of the CPR correlation uncertainties vs uncertainties in the additive constants, what other changes in the SLMCPR methodology have been introduced in Revision 2 of ANF-524(P)?

Answer 1

Question 2

Provide a specific reference for the 0.02 uncertainty in Table 5.1 for the additive constants.

Answer 2

The 0.02 uncertainty found in Table 5.1 of ANF-524(P), Revision 2 comes from Table 6.2 of document ANF 1125(P), Supplement 1. This uncertainty is for ANF 8x8 and 9x9-2 BWR fuel assemblies. ANF-1125(P), Supplement 1 gives additional additive constant uncertainties for other fuel types. The NRC has generically approved ANF-1125(P), Supplement 1.

Question 3

What distributions were used for the uncertainties in the MICROBURN-B power distribution measurement and the ANFB additive constants, *l*? Provide justification for these distributions. If normal distribution were assumed, justify this selection with statistical tests.

Answer 3

The distribution for the MICROBURN-B power distribution measurement and the ANFB additive constant uncertainties are

Pages 5 - 6 have been deleted.

Question 4

Does MICROBURN-B and/or ANFB non-conservatively over predict the limiting CPR and, if so, how are the biases accounted for in the SLMCPR determination?

Answer 4

Question 5

Do the ANFB and MICROBURN-B uncertainties depend on the local conditions, fuel design, core loading or the specific plant? If so, how is this dependence accounted for in the SLMCPR methodology? For example, the ANFB uncertainties seem to be larger for low mass fluxes and for higher pressure and enthalpy (Figures 5.3-5.5).

Answer 5

The MICROBURN-B uncertainty is selected to be appropriate for all plant and fuel types and applied in a manner consistent with the currently approved methodology.

The ANFB uncertainties depend on fuel type because the additive constants vary for different fuel types. The additive constant uncertainty is derived from the CHF tests based on the fuel type being tested.

The observations made regarding the higher uncertainties at low mass flux, high pressure, and high enthalpy are correct. In applying the conservatisms in the analysis, the values used are conservatively drawn from the entire database. These uncertainties are conservative based on the following discussion.

The safety limit is established to protect 99.9% of the fuel rods in the core during an operational transient initiated when the reactor is in normal operation mode. The extreme values of mass flux, pressure, and burnout enthalpy are at core conditions outside of the normal operating domain of a boiling water reactor. Since these higher uncertainties are combined with the data that is in the range of normal operations, the resulting uncertainty values are conservatively bounding for normal operating conditions.

The new safety limit methodology based on ANFB addresses the effects of core loading just as current methodology based on the XN-3 correlation does through the use of a slightly higher

Question 6

To what specific fuel/spacer designs and plant types will the ANFB and MICROBURN-B uncertainties be applied? Justify this application if it is not included in the associated verification data base.

Answer 6

In conformance with the SER issued for the ANFB correlation, the use of the ANFB correlation and uncertainties will be limited to assessments with the additive constants given in ANF-1125(P), Supplement 1. If a new fuel design results in additive constants outside the range of ANF-1125(P), Supplement 1, the new additive constants and uncertainties will be justified.

The MICROBURN-B uncertainties will be applied in the U.S. for General Electric BWR 2,3,4,5, and 6 reactors. In general, the MICROBURN-B uncertainties are being applied just as the XTGBWR uncertainties were in the previously approved methodology.

Question 7

Have the worst-case plant parameters (e.g., power and flow distributions) which put the maximum number of rods closest to the SLMCPR been selected for the SLMCPR calculation?

Answer 7

Yes. The worst case plant parameters are used for the SLMCPR calculation.

Question 8

Has the number of trials been reduced in the new CPR methodology? If so, justify this reduction.

Answer 8

No. The number of trials is not reduced.

Question 9

Why has the core inlet temperature uncertainty been eliminated form Table 5.1?

Answer 9

Question 10

In the new methodology, the CPR correlation uncertainty has been reduced substantially (from 4.1% to 2.5%) and this uncertainty is now introduced by varying the additive constants - $\ell$. There are several questions concerning the magnitude and the simulation of this uncertainty:

a. How are the random variations in the additive constants introduced? Are all constants for all rods in the core changed by the same amount, or are the $\ell$'s for different fuel bundle designs varied independently? What is the basis for the procedure used?

b. Varying the additive constants - $\ell$, rather than the calculated CPR, gives physically artificial pressure and flow dependence to the ANFB correlation uncertainty. Does this procedure reproduce the observed 2.5% standard deviation between the measurement and calculation of the verification test data? If not, justify the variation of the additive constants - $\ell$ rather than the calculated CPR.

c. While the introduction of the fuel and spacer design dependent additive constants - $\ell$ results in a reduced CPR calculational uncertainty, this lower uncertainty has only been verified for the dependent test data that was used to determine the

ANFB CPR correlation. Therefore, verify the lower ANFB correlation uncertainty using independent data, by partitioning the test data base and using different parts for the construction and the verification of the correlation.

Answer 10

- Part a

The additive constants are varied for each individual rod in the core by a random process.

- Part b

As discussed in the response to question 5, the range of operating conditions present during normal reactor operations would produce smaller uncertainties than are used in the safety limit analysis.

- Part c

The uncertainty in the ANFB correlations is accounted for through the additive constant uncertainty.

Question 11

The XN-NF-524(P) methodology assumes the MCPR monitoring calculation is performed by the POWERPLEX. The ANF-524(P) assumes the MCPR Monitoring calculation will be performed with MICROBURN-B. How will the change from POWERPLEX to MICROBURN-B be implemented? Will any plants use the MICROBURN-B/XN-3 or POWERPLEX/ANFB combination?

Answer 11

The change from XTGBWR based POWERPLEX to MICROBURN-B based POWERPLEX is implemented through the use of the appropriate Radial Bundle Power Uncertainty factor. The hydraulic models in the two monitoring codes have not changed.

No plants will be licensed with a methodology that uses a MICROBURN-B/XN-3 combination. If a plant is upgraded to ANFB but retains an XTGBWR based version of POWERPLEX, the appropriate Radial Bundle Power Uncertainty factor for XTGBWR based POWERPLEX will be used.

In general, the ANF-524(P) methodology can be used with any core monitoring code if sufficient information is available about uncertainties associated with the core monitoring code and ANFB is used for CPR calculations.

Question 12

How are the uncertainties in the bundle geometry factors (e.g., bundle flow areas, heated area, etc.) accounted for in the statistical analysis?

Answer 12

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April 9, 1990
RAC:030.90

Mr. Robert C. Jones, Chief
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Jones:

Subject: Loss of Thermal Margin Caused by Channel Box Bow

Reference:
1. Telecon: D. Fieno (NRC) S.E. State (ANF), and L.A. Nielsen (ANF), Same Subject, April 2, 1990.
2. NRC Bulletin No, 90-02: "Loss of Thermal Margin Caused by Channel Box Bow," dated March 20, 1990.
3. Letter, R.A. Copeland (ANF) to Director of NRR (USNRC), "Transmittal of ANF-524(P), Revision 2," April 11, 1989, RAC:023:89.

Attached is the additional documentation that was discussed in the telephone call of April 2, 1990 (Reference 1). The attachment provides data that shows the preliminary values of the channel bow MCPR penalty identified in the Bulletin (Reference 2) is bounded by conservatisms in the XN-3 critical heat flux correlation.

Please consider the information in these responses to be proprietary to ANF. The affidavit supplied with the original submittal (Reference 3) provides the necessary information as required by 10 CFR 2.790(b) to support the withholding of the attachment from public disclosure.

If there are questions, or if I can be of further help, please contact me.

Sincerely:

RA Copeland

R. A. Copeland
Manager, Reload Licensing

/skm
cc: D. Fieno (NRC)

The XN-3 correlation was approved for use in developing design limits for boiling water reactors by the NRC in July of 1982. In March of 1990, NRC approved the ANFB correlation for use in developing design limits for boiling water reactors. In the eight years between approval of XN-3 and ANFB, ANF has added more than 1300 critical heat flux data points. This additional data made it possible to quantify the magnitude of known conservatisms in XN-3 when applied to ANF 8x8-2 and 9x9-2 fuel designs.

The technical basis for determining the XN-3 conservatism comes from analysis of the ANF critical heat flux data base. The additional data added since the approval of XN-3 includes 27 full array tests. The XN-3 correlation was developed using conservative part array tests. By adding full array tests to the critical heat flux data base, it became possible to quantify and remove the excess conservatism associated with the part array tests. This was done in developing ANFB.

ANF believes that the approach outlined above is the most prudent way to address the channel box bow issue at this time. This approach will not result in a reduction in safety margin for those plants using ANF 8x8-2 and 9x9-2 fuel assemblies licensed with the XN-3 correlation. This approach also permits sufficient time for NRC review and approval of the ANF channel bow methodology.

Table 1 MCPR Comparison of in Reactor Data for ANF 9x9-2

Table 2 Critical Power for ANF 8x8-2 as a Function of Flow and Inlet Enthalpy

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May 3, 1990
RAC:045:90

Mr. Robert C. Jones, Chief
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Additional Information Regarding Loss of Thermal Margin Caused by Channel Box Bow

Reference:

1. Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "Request for Additional Information Regarding the Topical Report ANF-524 Rev. 2, Supplement 1," dated March 30, 1990.
2. Letter, R.A. Copeland (ANF) to Director of NRR (USNRC), "Transmittal of ANF-524(P), Revision 2," April 11, 1989, RAC:023:89.

Dear Mr. Jones:

Attached is the additional information requested in the Reference 1 letter. The responses provide information regarding the ANF methodology used to calculate the loss of thermal margin caused by channel box bow.

Please consider the information in these responses to be proprietary to ANF. The affidavit supplied with the original submittal (Reference 2) provides the necessary information as required by 10 CFR 2.790(b) to support the withholding of the attachment from public disclosure.

If there are questions, or if I can be of further help, please contact me.

Sincerely:

RA Copeland

R. A. Copeland
Manager, Reload Licensing

/skm
cc: D. Fieno (NRC)
L. Lois (NRC)

A Siemens Company

ADDITIONAL INFORMATION REQUIRED FOR THE
REVIEW OF THE ADVANCED NUCLEAR FUELS CHANNEL
BOWING SUPPLEMENT-1 TO ANF-524[P], REVISION 2

1. **How is the effect of channel bowing on the bundle $K_\infty$, core reactivity and shutdown margin and bundle power distribution accounted for?**

2. **What bundle displacements are calculated? Are displacements perpendicular to the bundle face and diagonal bundle displacements analyzed? Is this selection of bundle displacements consistent with the channel bowing measurements? Justify the use of a reduced set of bowed geometries to represent all expected bundle displacements.**

Ans: Bundle displacements that are both perpendicular and diagonal to the bundle face have been analyzed. The bundle displacements are consistent with the channel bowing measurements. A reduced set of geometries is used to simplify the overall calculation process and at the same time conservatively bound the expected in core geometries.

Attachment

3. **Discuss the applicability of XN3 and/or ANFB to the radially and axially bowed channel geometry? Is the internal rod-to-rod pitch preserved?**

Ans: XN-3 and ANFB can be conservatively applied to the bowed channel geometry. The rod-to-rod pitch is assumed to be preserved in the region where CHF typically occurs due to the mechanical constraint of the spacer.

4. **What is the effect of channel bowing on the bypass flow distribution?**

5. **What procedures are used in the fuel loading/orientation which affect the channel-to-channel spacing? How are these procedures accounted for in the determination of the effects of channel bowing?**

Ans: In the ANF evaluation of channel bow, utility procedures which define fuel loading/orientation are not considered and therefore have no impact on channel-to-channel spacing. The largest potential channel bowing is assumed in the analysis methodology.

Attachment

6. How does channel bowing and the associated fuel loading/orientation procedures affect the calculation of the DMCPR for a fuel misloading/misorientation error?

7. What boundary conditions are assumed in the CASMO-3G calculations and how is the sensitivity to the boundary conditions accounted for in determining the power increase?

8. Describe in detail the calculation of the R-factor uncertainty resulting from the channel bow uncertainty. Justify any approximations or assumptions in this determination.

9. What cycle-dependent input will be used to determine the channel bow? Will the bow be estimated using bundle-specific values of initial bow, fluence, etc.?

10. How will the bowing and R-factor sensitivity be determined for mixed cores of ANF and non-ANF fuel?

11. Are all uncertainties in the determination of the cycle-specific gap spacing and the bowed power maps included in the R-factor uncertainty. For example, how is the uncertainty allowance determined and accounted for:

(1) the strain dependence on bundle-specific fluences, initial bow, etc.,

(2) the dependence of the gap spacing on the cycle-specific fuel loading,

(3) the dependence of the bowed power maps on the neighboring fuel bundles (rather than those assumed in the four-bundle cell calculations), and

(4) the four-bundle cell bow model - i.e., representing all possible bowed geometries by the selected set of calculated displacement patterns.

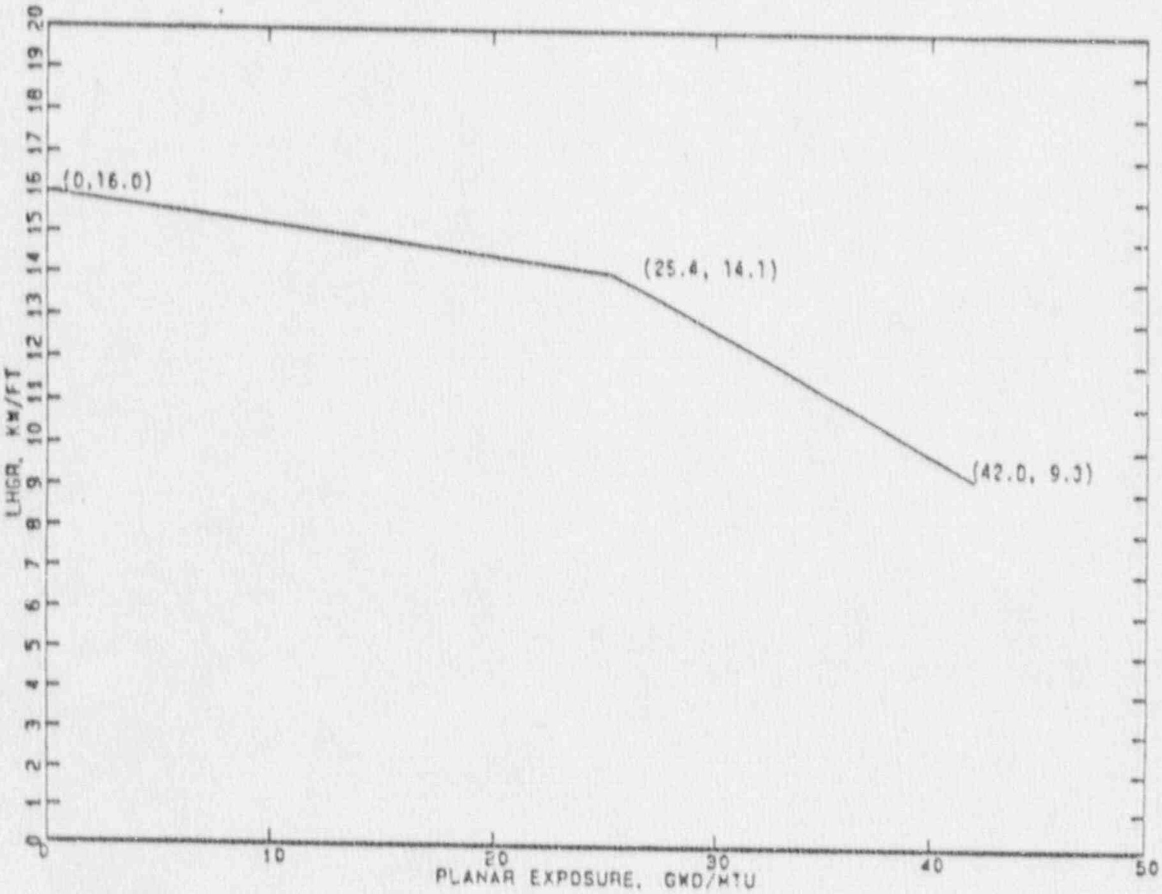
Ans: The uncertainty allowance is accounted for as follows:

12. What conservatism has been incorporated into the calculation of the channel-to-channel spacing and the bowed channel power maps?

13. How will the effect of channel bowing on the LHGR be accounted for?

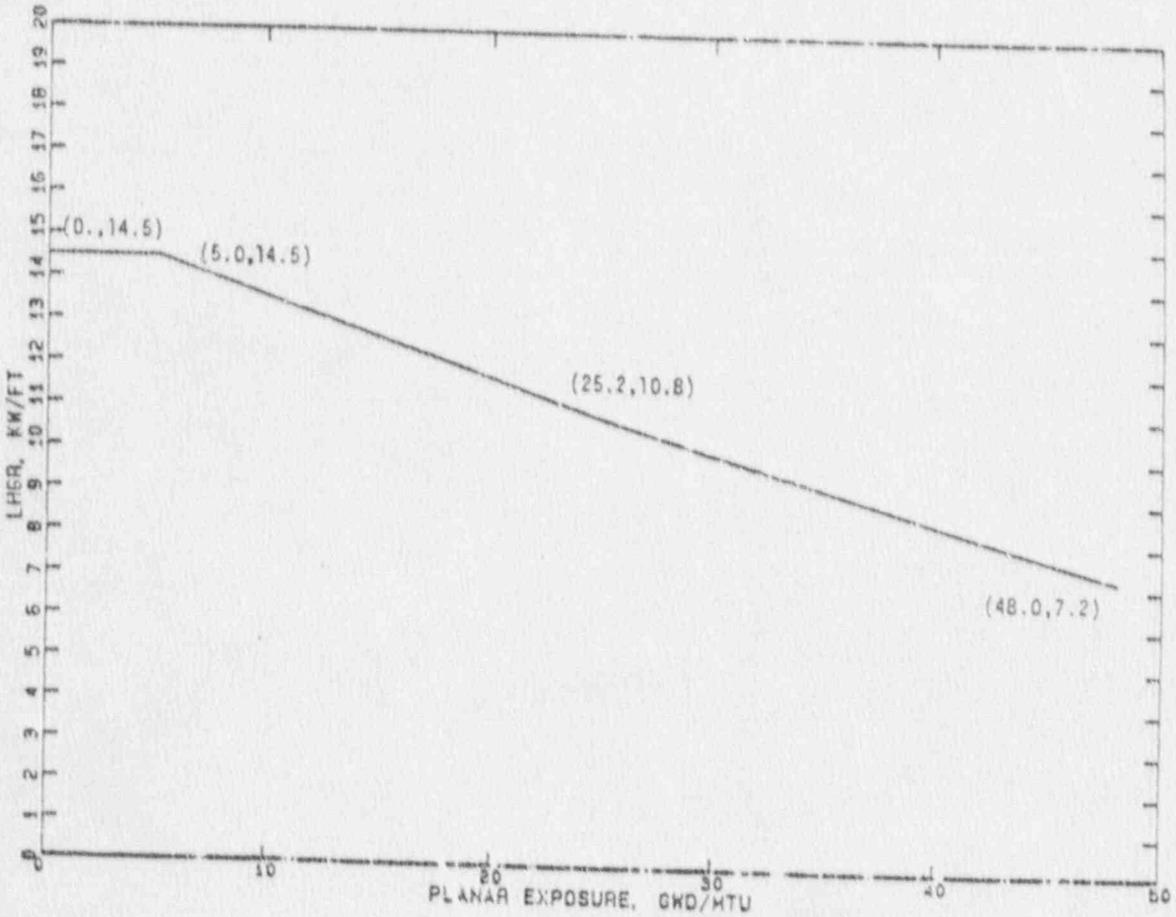
Attachment

FIGURE 13-1 LHGR LIMIT FOR 8X8 FUEL



Attachment

FIGURE 13-2 LHGR LIMIT FOR 9X9 FUEL



14. Discuss in detail how the channel-to-channel spacings are determined for a specific core and fuel loading. How do you account for (1) the spacing dependence on the fuel loading and the number of exposed bundles in a four-bundle cell and (2) the dependence of the power maps on the neighboring fuel bundles? How are C and D lattices distinguished?

15. How is the exposure and void dependence of the bundle pin power maps accounted for?

16. What, if any, geometrical variables (e.g., radial bundle displacement, total gap spacing, etc.) are used to correlate the CASMO-3G calculated power maps as a function of channel bowing?

17. How is it assured that the selected four-bundle cell calculation will conservatively represent the typical four-bundle cell in the reactor?

18. Demonstrate that the ANF TIP and power distribution measurement uncertainties are sufficient to account for the effects of channel bowing.

19. How is the fuel channel bowing and the associated standard deviations determined?

Ans: ANF has two base methods that are used to determine expected channel bowing.

Attachment

20. Channel bowing data for GE, CARTECH and ASEA channels has been obtained from several sources (including EPRI, KWU and the Swedish Regulatory Authority) and is used to determine the cycle-dependent expected channel bow. In order to demonstrate the adequacy of this data to represent all intended channel suppliers, core loadings and fuel exposures, please provide the data to be used in the determination of the mean channel bow.

21. The staff found no information in this submittal which addresses second channel box lifetime, thus, we assume that only single lifetime is intended.

**ADVANCED NUCLEAR FUELS** *CORPORATION*

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June 7, 1990
RAC:059:90

Mr. Robert C. Jones, Chief
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Additional Information Regarding Loss of Thermal Margin Caused by Channel Box Bow

Reference:
1. Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "Request for Additional Information Regarding the Topical Report ANF-524 Rev. 2, Supplement 1," dated June 6, 1990.
2. Letter, R.A. Copeland (ANF) to Director of NRR (USNRC), "Transmittal of ANF-524(P), Revision 2," April 11, 1989, RAC:023:89.
3. Letter, R. A. Copeland (ANF) to R. C. Jones (NRC), "Additional Information Regarding Loss of Thermal Margin Caused by Channel Box Bow," dated May 3, 1990, RAC:045:90.

Dear Mr. Jones:

Attached is the additional information requested in the Reference 1 letter. The responses provide information regarding the ANF methodology used to calculate the loss of thermal margin caused by channel box bow.

Please consider the information in these responses to be proprietary to ANF. The affidavit supplied with the original submittal (Reference 2) provides the necessary information as required by 10 CFR 2.790(b) to support the withholding of the attachment from public disclosure.

If there are questions, or if I can be of further help, please contact me.

Sincerely:

Bob Copeland
R. A. Copeland
Manager, Reload Licensing

/skm
cc: D. Fieno (NRC)
L. Lois (NRC)

## Request for Additional Information on the Assessment of the Effects of Fuel Channel Bowing

1. Fuel Channel Bowing Displacement

The response to Question 2 (Reference 3) does not indicate what specific fuel bundle displacements are calculated in the determination of the bowing effect on local power peaking. Demonstrate that the selected displacements are conservative or account for the additional uncertainty introduced by this selection in the uncertainty analysis.

Since only one displacement configuration of the four bundles in the four bundle cell is calculated, demonstrate that this configuration is a worst-case or bounding arrangement of the four-bundle geometry.

2. Uncertainty Analysis

In the determination of the uncertainty in the water gap thickness, is the variability of both neighboring fuel channels accounted for?

Ans: Yes, the uncertainty in the water gap thickness for both neighboring channels is considered. In the calculation, this variability is accounted for independently on the two sides.

3. Sensitivity of Power Peaking to Channel Bow

The sensitivity of the bundle pin power distribution to channel bowing has a substantial dependence ($\leq$ 50%) on the in-channel void fraction. This dependence must be accounted for in determining the power sensitivity to bow (Response-15).

ANF-524(NP)(A)
Revision 2

ANF-524(NP)(A)
Supplement 1
Revision 2

ANF-524(NP)(A)
Supplement 2

Issue Date: 11/30/90

# ADVANCED NUCLEAR FUELS CORPORATION CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

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