XN-NF-524 (NP) (A)
REVISION 1

EXXON NUCLEAR CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

NOVEMBER 1983

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

XN-NF-524(NP)(A)
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EXXON NUCLEAR CRITICAL POWER METHODOLOGY

FOR BOILING WATER REACTORS

This is the NRC approved version of Document XN-NF-524(NP), Revision 1 and has been prepared in accordance with NRC guidance.

EXXON NUCLEAR COMPANY, Inc.

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

OCT 3 1 1983

Mr. J. C. Chandler
Exxon Nuclear Company
P. O. Box 130
Richland, Washington 99352

Dear Mr. Chandler:

Subject: Acceptance for Referencing of Licensing Topical Report XN-NF-524 (P), "Exxon Nuclear Company Critical Power Methodology for Boiling Water Reactors" - Revision 1

We have completed our review of the subject topical report submitted May 30, 1980 by Exxon Nuclear Company (ENC) letter GFO:096:80. We find this report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable 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 report.

In accordance with procedures established in NUREG-0390, it is requested that ENC publish accepted versions of this report within three months of receipt of this letter. The accepted version should incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted version shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, ENC 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,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure: As stated

SAFETY EVALUATION REPORT

OF THE

EXXON NUCLEAR COMPANY TOPICAL REPORT

XN-NF-524(P)

Revision 1

"EXXON NUCLEAR COMPANY CRITICAL POWER

METHODOLOGY FOR BOILING WATER REACTORS"

May 1983

Core Performance Branch

1.0 INTRODUCTION

The limitations on the total power produced by a boiling water reactor (BWR) are established such that boiling transition will not occur during normal operation and reactor system transients. By preventing boiling transition, adequate heat transfer is maintained between the fuel rod cladding and the reactor coolant. This ensures that the fuel cladding integrity is maintained and a barrier between the reactor fission products and coolant exists.

Since boiling transition is not a measurable quantity, the amount of thermal margin present in a BWR core is expressed in terms of the critical power ratio (CPR). The methodology used by Exxon Nuclear Company (ENC) to determine the minimum critical power ratio (MCPR) of a BWR is presented in topical report XN-NF-524(P), (Ref. 1).

2.0 DESCRIPTION OF REPORT

The initial step in the ENC CPR methodology is the determination of the flow distribution in the reactor core. Since BWR fuel assemblies are surrounded by metal channel boxes, ENC models the core as parallel flow paths having equal pressure drops between the upper and lower plenums. An iterative solution process is used to determine the core flow distribution. The individual assembly flow rates are adjusted until the pressure drops for all of the assemblies are equal and the sum of their flows equals the core flow.

The models and correlations which form the basis of the ENC pressure drop methodology are given in XN-NF-79-59(P), (Ref. 2). Also included in the core flow distribution calculation are the energy deposition rate in the active coolant and the bypass flows.

Once the core flow distribution is determined, the amount of thermal margin is calculated using the XN-3 critical power correlation. The XN-3 correlation was developed from 900 data points obtained from 20 different test assemblies. A complete description of the XN-3 correlation and its development are given in Reference (3).

The MCPR safety limit for the core is determined by statistically convoluting the uncertainties associated with the thermal margin calculation. The MCPR safety limit is established such that 99.9% of the fuel rods are expected to avoid boiling transition. ENC employs a Monte Carlo procedure to determine the MCPR safety limit.

At a given operating state and core-wide power distribution, the critical power ratio for each rod in the core is determined using the XN-3 correlation. These CPRs are then used to calculate the probability of boiling transition for each rod. ENC then determines the number of rods expected to be in boiling transition by summing the rod probabilities over the entire core. By repeatedly applying this Monte Carlo procedure a frequency distribution of the number of rods in transition boiling can be defined. This distribution is statistically analyzed by fitting a Pearson curve to it using the methods described in References (4) and (5).

The number of rods expected in transition boiling, for a particular operating state, are derived from the statistical analyses. The MCPR for this state is considered the safety limit if the number of rods in boiling transition is less than or equal to 0.1%.

The criterion used by Exxon to determine the number of Monte Carlo trains needed in establishing the safety limit is "that number which provides sufficient data for an accurate Pearson curve fitting."

3.0 STAFF EVALUATION

The staff has reviewed the methodology described in XN-NF-524. Our review included the procedure used in calculating the core flow distribution (e.g., pressure drop methodology), the XN-3 correlation, and the method used in determining the safety limit MCPR.

Since the pressure drop methodology used to determine the core flow distribution and the XN-3 correlation have been reviewed and approved by the staff (Refs. 6 and 7) we find their use acceptable in the methodology presented in XN-NF-524. The method of accounting for uncertainties in parameters associated with the thermal margin calculation, the use of a Monte Carlo technique, and the fitting of a Pearson curve to the resultant distribution are also acceptable contingent upon the following restrictions:

(1) Each plant specific application must contain the data used to generate the uncertainties employed in the methodology.

(2) All plant parameters that are not statistically convoluted must be placed at their limiting value.

(3) Each application should demonstrate that the uncertainties in plant parameters are treated with at least a 95% probability at a 95% confidence level in accordance with Acceptance Criterion 1.0 of Standard Review Plan Section 4.4.

(4) Each application must present a goodness-of-fit analysis for the fitting of the Pearson curve in order to insure that the number of Monte Carlo trails used in establishing the safety limit MCPR are sufficient.

4.0 STAFF POSITION

Based on our review and the recommendation of our consultant (Ref. 8), we find XN-NF-524 an acceptable and referential report with the contingencies noted above.

5.0 REFERENCES

1. Patten, T. W., "Exxon Nuclear Company Critical Power Methodology for Boiling Water Reactors," XN-NF-524(P), Exxon Nuclear Company, November, 1979.

2. Adams, F. T. and T. W. Patten, "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, October, 1979.

3. Patton, T. W. and R. B. McDuff, "XN-3 Critical Power Correlation," XN-NF-512, Exxon Nuclear Company, June, 1979.

4. G. J. Hahn and S. S. Shapiro, Statistical Models in Engineering, John Wiley and Sons, New York, NY, 1967.

5. N. L. Johnson, Eric Nixon, D. C. Amos, and E. S. Pearson, "Table of Percentage Points of Pearson Curves for Given , and , expressed in standard measure," Biometrika, 50, 457, 1963.

6. Memorandum. L. S. Rubenstein (NRC) to R. L. Tedesco (NRC), Subject: Staff Evaluation of Exxon Topical Reports, February XX, 1982.

7. XN-3 SER.

8. Carlson, R. W., et al., "Evaluation of Exxon Nuclear Company Critical Power Methodology for Boiling Water Reactors," Georgia Institute of Technology, January, 1982.

XN-NF-524 (NP)(A) Revision 1

Issue Date: 5/26/83

EXXON NUCLEAR CRITICAL POWER METHODOLOGY

FOR BOILING WATER REACTORS

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EXXON NUCLEAR COMPANY, Inc.

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EXXON NUCLEAR CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

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EXXON NUCLEAR CRITICAL POWER METHODOLOGY FOR BOILING WATER REACTORS

1.0 INTRODUCTION

This document describes the Exxon Nuclear Company, Inc. (ENC) methodology used for determination of thermal margin of a boiling water reactor. 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 overestimate 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 distribution, the likelihood of boiling transition can be determined by use of the critical power correlation. The safety limit is derived by statistically convolving hydraulic and thermal calculational

* Numbers in brackets refer to references.

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 assumed to occur if the fuel rod experiences 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 minimum critical power ratio (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] 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 percent 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 envelopes expected reactor power distributions for normal reactor operation and as a consequence of reactor system transients.

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 Exxon Nuclear Company (ENC) 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 percent of the fuel rods in the reactor core from boiling transition. The MCPR safety limit, incorporated with a ΔCPR from transient analyses described elsewhere[6], 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 envelopes expected reactor power distributions. Hence, the MCPR safety limit represents a conservative limit with regard to protection of the reactor core from boiling transition. The CPR operating limit may be reactor core specific and hence is established on a core/cycle specific basis.

3.0 CORE FLOW DISTRIBUTION

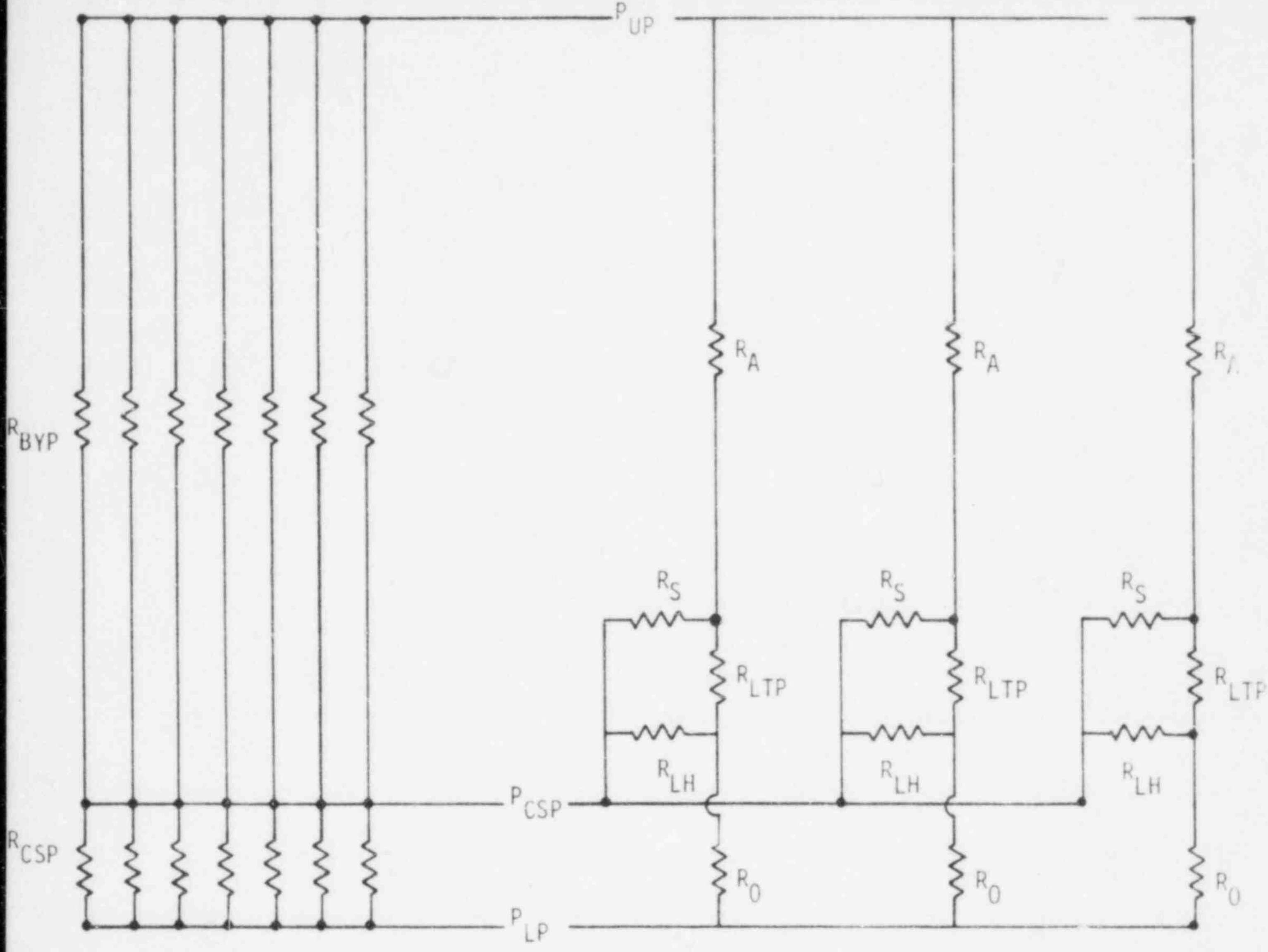
The calculation of the core flow distribution determines the flow to each assembly and the bypass region, and provides the hydraulic information necessary for calculating the assembly critical power ratios. The core flow distribution is calculated from a hydraulic model of the reactor core. The physical 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 ENC 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 pressure drop across each flow path is calculated from the channel flow model which is comprised of a combination of analytically and experimentally determined loss coefficients and incorporates the effects of assembly power through a set of two-phase flow models. The flow through each parallel flow path is adjusted iteratively until the pressure drop is equal for all parallel flow paths. For a given set of reactor operating conditions, the distribution of flow between assemblies and the bypass region is determined from the summation of the pressure losses associated with each flow path.

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.



P_{LP}	Pressure in lower plenum
P_{CSP}	Pressure above core support plate
P_{UP}	Pressure under plenum
R_O	Flow hydraulic resistance through orifice
R_{CSP}	Flow hydraulic resistance across lower core support plate
R_{LH}	Flow hydraulic resistance through holes in lower tie plate
R_{LTP}	Flow hydraulic resistance through lower tie plate
R_{CS}	Flow hydraulic resistance through channel seal
R_A	Flow hydraulic resistance through the assembly and upper tie plate
R_{BPY}	Flow hydraulic resistance in the bypass region

Figure 3.1 Schematic diagram of the reactor core hydraulic model.

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 channel flow model is comprised of analytical expressions for the pressure drop, fuel component loss coefficients, and experimentally verified two-phase flow models. It is general in nature, is not restricted to a particular fuel design or geometry, and is, therefore, applicable to cores loaded entirely with ENC fuel, as well as cores loaded with both ENC fuel and fuel supplied by other vendors. For a specified flow rate the pressure drop across the reactor core is determined from the core axial power distribution and a composite model of the core hydraulic resistance.

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

$$-\frac{dP}{dZ} = \frac{G^2}{g_c}\left\{\frac{f\phi^2_{BR}}{2D\rho_\ell} + \frac{\Delta v_m}{\Delta Z} + \frac{K_C\phi^2_C}{2\Delta Z\rho_\ell}\right\} + \frac{\bar{\rho}g}{g_c} + \frac{G^2}{2g_c}\left\{\frac{v_m\{(1-\sigma^2)-K_{exp}\}}{\Delta Z}\right\} \qquad (3.1)$$

where:

$\bar{\rho}$ is the average density defined by the relation $\bar{\rho} = \alpha\rho_g + (1-\alpha)\rho_f$

and the other quantities are as noted in the Glossary. Inspection of Eq. (3.1) shows that the single-phase component loss coefficients of a particular fuel design are sufficient to determine the two-phase pressure losses. Comparison of experimental two-phase pressure drops with those calculated by this methodology have established the accuracy of this approach.[11]

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. The variation of assembly flow due to the axial distribution of power within that assembly is correlated as a linear function of the axial offset, defined as twice the fraction of power produced in the upper half of the assembly minus one.

$$AO = 2U - 1 = \frac{U - L}{U + L} \tag{3.2}$$

A negative axial offset is indicative that greater than 50 percent 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. For a constant value of assembly axial offset, the relative assembly flow is well

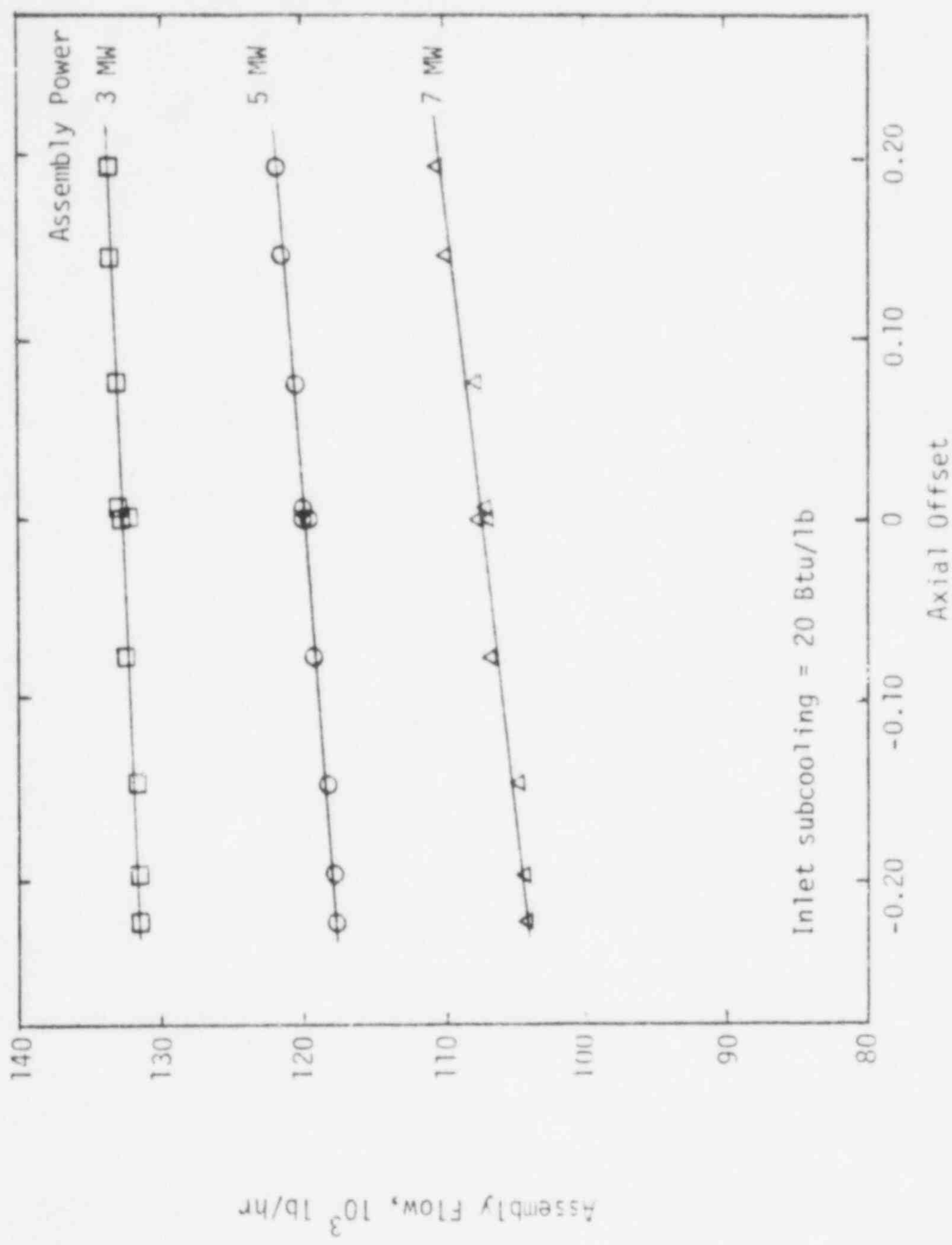


Figure 3.2 Effect of assembly axial offset on assembly flow rate for central orifice zone.

represented by a weakly quadratic relationship with relative power as indicated in Figure 3.3.

Ninety-six percent (96%) of the fission power produced in an operating fuel rod leaves the rod as surface heat flux. Approximately 2 percent of the fission power is directly deposited in the assembly coolant while the remaining 2 percent is directly deposited in the core bypass region. These energy deposition rates are included in the core flow distribution calculation to correctly distribute energy throughout the reactor core and to properly model the two-phase flow effects.

3.2 TWO-PHASE FLOW MODELS

The methodology used for the calculation of two-phase pressure drop is comprised of basic relations representing the various terms of the momentum equation and relies on constitutive relationships (correlations) for void fraction and two-phase friction multiplier. 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.

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. The prediction of this data provides an indication

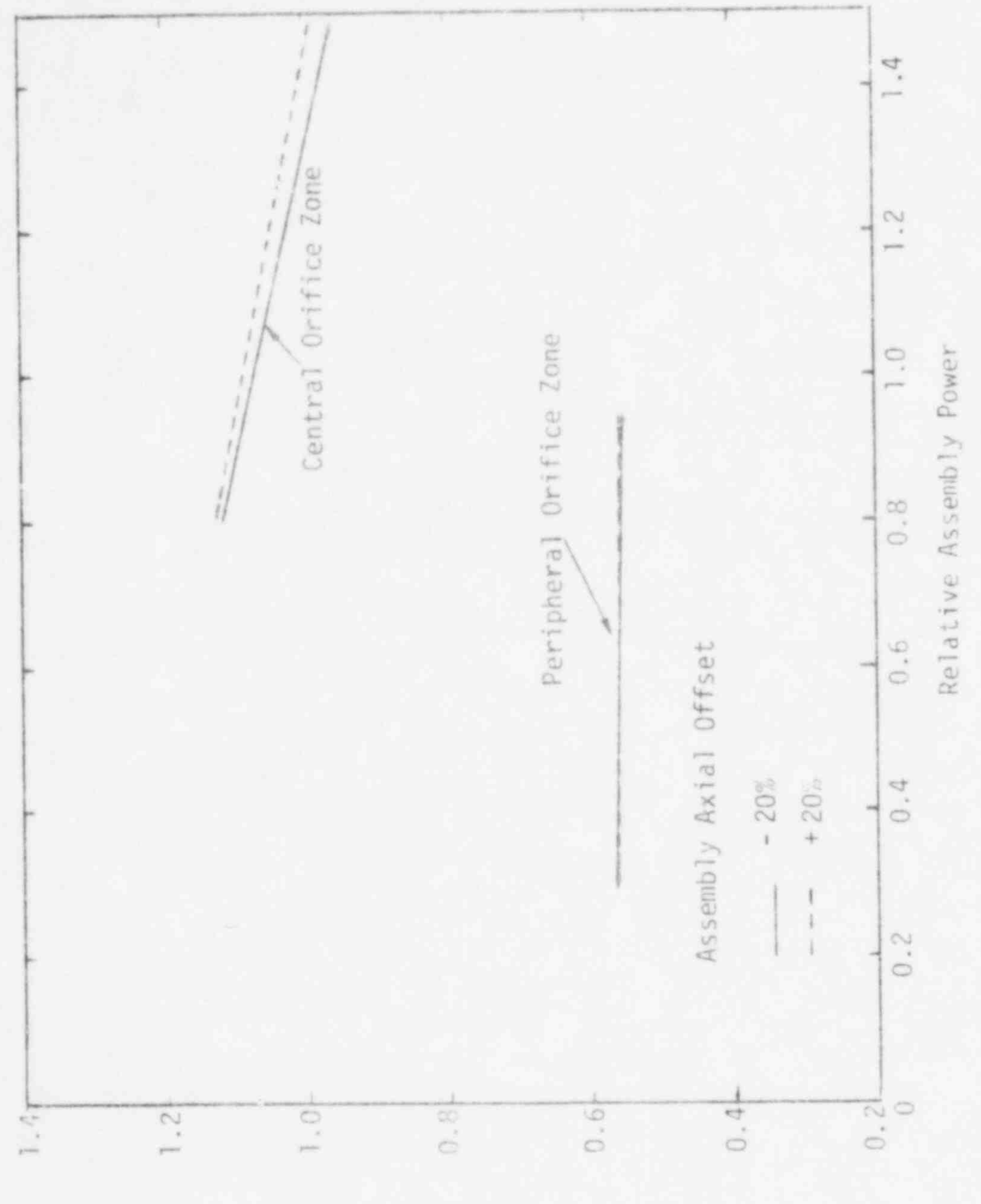


Figure 3.3 Core flow distribution.

of the accuracy of the two-phase methodology and thereby provides a basis for estimating the subcomponent of the total flow uncertainty due to two-phase flow modeling.

A total of 419 data points were predicted for five separate test assemblies employing two different spacer designs, three different axial power profiles, and operating in a wide range of mass velocity, pressure, inlet enthalpy, quality and assembly power. The basis of comparison of predicted and measured pressure drops was the relative error defined as the ratio of the predicted minus the measured pressure drop to the measured pressure drop. The overall standard deviation was determined to be 0.033. No significant biases were observed in the data prediction as indicated by the data comparison shown in Figure 3.4. The data comparison may be conservatively represented as a normal distribution. Further details of the data comparison are provided in Reference 11.

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. Because diabatic test data was used as a basis in the pressure drop data comparison, the void fraction model has been implicitly

Figure 3.4 Comparison of measured and predicted two-phase pressure drop

verified by the absence of any trends in the data comparison as well as by the agreement of the test predictions with the test data. The void fraction model is described fully in Reference 11.

3.2.2 Two-Phase Friction Multipliers

Correlations are used to determine the two-phase friction multipliers applicable in the bare rod region, and through components such as spacers and the upper tie plate. The two-phase friction multipliers are dependent upon the assembly pressure, mass velocity, local quality, and rod surface heat flux. The two-phase friction multiplier correlations, which are described fully in Reference 11, are not dependent upon fuel design and are applicable to both ENC fuel and the fuel supplied by other vendors.

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 ENC fuel and existing fuel. This eliminates the potential for experimental uncertainty due to the use of different test facilities and testing procedures. For example, the PHTF was used to determine the single-phase loss coefficients of both the ENC fuel design and the GE 8x8R fuel design.

The uncertainty in the assembly loss coefficients determined by measurement in the PHTF is 1.8 percent. The uncertainty in the total pressure drop of a measured fuel assembly is therefore 0.9 percent since approximately one-half of the total assembly pressure drop is due to the orifice which is common to both fuel designs. The 0.9 percent uncertainty in the total assembly pressure drop is equivalent to a 0.45 percent uncertainty in the flow rate. Because the flow split in a BWR is determined by the difference in assembly pressure drop, there is a 0.6 percent uncertainty in the assembly flow rate to either fuel type if both are simultaneously loaded in the core and both have been hydraulically characterized in the PHTF.

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 XN-3 critical power correlation.[10] The XN-3 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 percent increase in power before it is expected that boiling transition will occur on the most limiting rod within that assembly.

4.1 XN-3 CRITICAL POWER CORRELATION

The XN-3 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 range of assembly geometry in the XN-3 data base allows application of the XN-3 correlation to both ENC and other vendor fuel designs.

The XN-3 correlation is comprised of a base correlation with correctors for pressure, local rod power peaking, grid spacer design, and non-uniform axial power distribution. The XN-3 data base is comprised of 1,501 data points taken with 26 different test assemblies. The test assemblies include both partial and full-length rods, uniform and non-uniform axial heat flux profiles, different grid spacer designs, and a variety of rod diameters, assembly hydraulic diameters, rod-to-wall spacings, and rod-to-rod spacings. The data base is consistent with data taken at three different test laboratories: Battelle-Northwest, Columbia University, and CISE in Milan, Italy.

The test data base and correlation address the effects upon boiling transition due to operating pressure level, mass velocity, enthalpy, axial power peaking and distribution, local power peaking and distribution, rod diameter, assembly hydraulic diameter and heated length. Comparison of the XN-3 correlation to boiling transition test data indicates that the critical power ratio is predicted as a normal distribution with a mean of 1.00, and a standard deviation of 0.041 for grid spacer designs prototypic of BWR fuel assemblies. A comparison of measured and predicted critical powers is shown in Figure 4.1, while a histogram of predicted CPR's for the experimental data points is provided in Figure 4.2.

The XN-3 correlation has also been compared to transient boiling transition data.[12] Assembly power and flow were varied in a manner typical of anticipated transients until boiling transition occurred. It was determined that the XN-3 correlation consistently underpredicted the time to boiling transition, indicating that use of the XN-3 correlation to predict critical power during transient operating conditions is conservative.

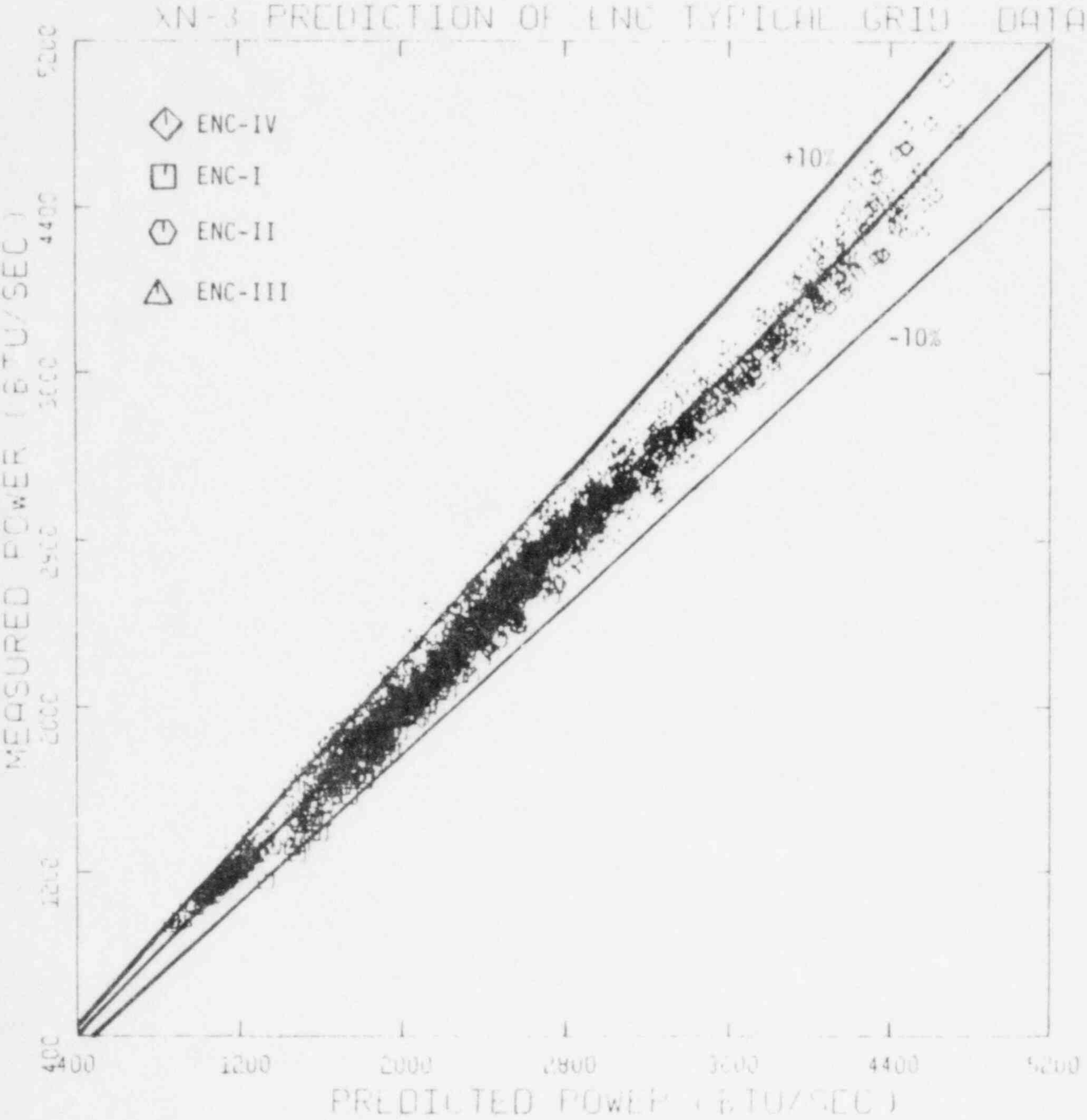


Figure 4.1 XN-3 Prediction of Typical Grid Data

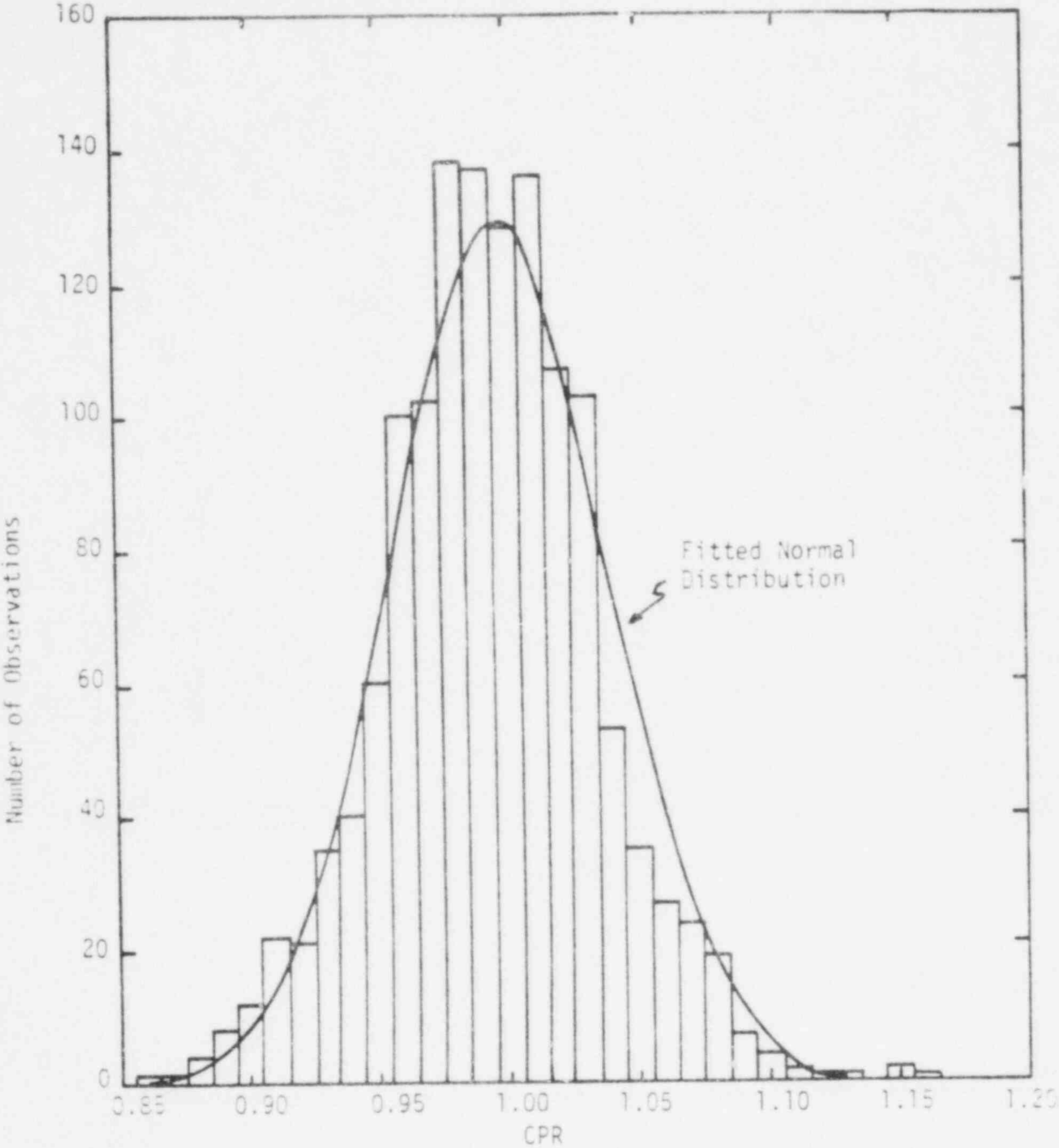


Figure 4.2 Histogram of XN-3 Predictions for Typical Grid Data

The XN-3 correlation has also been used to predict the number of rods experiencing boiling transition for the test data most prototypic of ENC BWR fuel assemblies. The probability of boiling transition for all the rods in an assembly, as predicted by the XN-3 correlation, were summed to yield a prediction of the total number of rods in boiling transition for a particular data point, and thereby predict the occurrence of multiple rod indications of boiling transition. Use of the XN-3 correlation in this manner was determined to overpredict the probability of boiling transition, indicating that use of the XN-3 correlation to calculate the number of rods in boiling transition for a particular set of reactor operating conditions is conservative.

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. With the conditions of pressure, flow, inlet enthalpy, and local enthalpy known, the critical power is determined based upon the XN-3 critical power correlation.[10] The procedure is iterative, in that for fixed conditions of pressure, flow, and inlet enthalpy bundle power is adjusted until boiling transition is just predicted to occur. The ratio of this adjusted bundle power to the actual bundle power is defined as the critical power ratio. A complete description of the step by step procedure for determining critical power is presented in Reference 10.

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 percent of the rods in the core are expected to avoid boiling transition. The MCPR safety limit is determined by a statistical convolution of all the uncertainties associated with the calculation of thermal margin. The set of uncertainties which form the basis for the statistical convolution are established by the relative sensitivity of all the parameters which are incorporated into the MCPR calculation. These parameters include both fuel-related uncertainties, which may vary with reactc. loading cycle, and non-fuel-related uncertainties, which are characteristics of the reactor system.[7]

The non-fuel-related uncertainties are those uncertainties which do not depend upon the particular type of fuel present in the reactor core. Examples of non-fuel-related uncertainties are the measurement uncertainties associated with reactor pressure, feedwater flow rate and temperature, total core flow rate, and core inlet subcooling. Examples of fuel-related uncertainties are those introduced by the XN-3 critical power correlation, the calculation of core-wide power peaking factors, and the calculation of the core-wide flow distribution, which inlcudes uncertainties associated with the core hydraulic model. The contribution of the various subcomponents to the overall MCPR uncertainty is determined from the calculational procedure used to evaluate MCPR.

The uncertainties used in determining the MCPR safety limit are statistically convolved via a Monte Carlo procedure. The Monte Carlo procedure simulates a variety of reactor states around a base state, where the reactor states are determined by randomly varying the reactor conditions according to the magnitude of their uncertainties. For a particular base reactor state and core-wide power distribution, the core parameter values

are randomly varied according to the probability distribution of the respective uncertainty.

The CPR for each of the rods in the model is determined by using the XN-3 correlation in a calculation

and the associated boiling transition probability for each rod is calculated using the XN-3 correlation uncertainty. The rod boiling transition probabilities are then summed over the entire core to determine the number of rods expected to be in boiling transition for the reactor state chosen for a particular Monte Carlo trial. This procedure is repeated until a sufficient number of trials have been performed to adequately determine the expected number of rods in boiling transition.

If the expected number of rods in boiling transition for the base reactor state, as determined by this procedure, is equal to 0.1% of the rods in the core, the MCPR for the base state is defined as the MCPR safety limit. If the number of rods in boiling transition is greater than 0.1% of the rods in the core, a new base reactor state which is less severe is chosen, and the Monte Carlo procedure is again performed to determine the expected number of rods in boiling transition for that reactor state. Conversely, if the expected number of rods in boiling transition for the chosen base reactor state is less than 0.1% of the rods in the core, a new base reactor state which is more severe is chosen, and the Monte Carlo procedure is repeated. This procedure is then iteratively performed until the minimum acceptable MCPR which results in an expected number of rods in boiling transition of less than or equal to 0.1% of the rods in the core is determined. The MCPR of that base reactor state is then defined as the MCPR safety limit.

Table 5.1 summarizes the fuel and non-fuel-related uncertainties which are used to generate the MCPR safety limit. The reactor system uncertainties[7] shown in Table 5.1 are typical of operating BWR's and are generic in nature. Those uncertainties are convolved to determine the MCPR safety limit using a design basis reactor core power distribution. The design basis power distribution is comprised of design basis radial, local, and axial power distributions, all of which conservatively envelope 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.

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 Uncertainties Used To Generate MCPR Safety Limit

Reactor System Uncertainties[7]

Parameter	Standard Deviation of Uncertainty (Percent of Nominal
Feedwater Flow Rate	1.76
Feedwater Temperature	0.76
Core Pressure	0.5
Total Core Flow Rate	2.5
Core Inlet Temperature	0.2

Fuel Related Uncertainties

Parameter	Standard Deviation of Uncertainty (Percent of Nominal
XN-3 Correlation[10]	4.11
Assembly Flow Rate	2.7 for ENC cores 2.8 for mixed cores
Radial Bundle Power[13]	5.28
Local Power[13]	2.46
Axial Power[13]	2.99

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

ΔZ = calculational increment

α = void fraction

v_m = specific volume for momentum transfer

$\bar{\rho}$ = average density

ρ_g = density of saturated vapor

ρ_f = density of saturated fluid

ρ_ℓ = density of liquid phase

σ = area ratio

ϕ^2_{BR} = bare rod two-phase friction multiplier

ϕ^2_C = Component two-phase friction multiplier

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