VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

November 4, 1981

R. H. LEASBURG
VICE PRESIDENT
NUCLEAR OPERATIONS

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Attn: Mr. D. G. Eisenhut, Director
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Washington, DC 20555

Serial No. 614
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Docket Nos.: 50-280
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License Nos.: DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VEPCO NUCLEAR DESIGN RELIABILITY FACTORS

Enclosed for your review are forty (40) copies of the Vepco Topical Report VEP-FRD-45, "Vepco Nuclear Design Reliability Factors".

This report is one of a series of topical reports supplying general information pertaining to the nuclear reload licensing and core follow support capabilities which have been developed at Vepco. It describes the methods and data base used to derive Nuclear Reliability Factors for application to core physics input parameters used in the reload safety evaluation of Vepco nuclear units. The Vepco topical report, which describes the methods in which these factors will be applied, has been previously submitted (VEP-FRD-42, "Vepco Reload Nuclear Design Methodology", transmitted by letter to you dated June 12, 1981, Serial No. 350).

It has been determined that the methodology and analysis capability described herein do not involve an unreviewed safety question as defined in 10CFR50.59.

We are aware that a fee will be required for the topical report review and will transmit the assessed fee upon completion of the review.

A010
5/40

If you have any questions on the material in this topical report, please contact us.

Very truly yours,

R. H. Leasburg

cc: Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
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Mr. Steven A. Varga, Chief
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VEP-FRD-45
SEPTEMBER 1981

Vepco

NUCLEAR DESIGN RELIABILITY FACTORS

FUEL RESOURCES DEPARTMENT

VIRGINIA ELECTRIC AND POWER COMPANY

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VEPCO NUCLEAR DESIGN RELIABILITY FACTORS

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SEPTEMBER, 1981

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ABSTRACT

This report describes the methods and data base used to derive Nuclear Reliability Factors for application to the reload safety evaluation of Virginia Electric and Power Company (Vepco) operating nuclear units. Where possible the Nuclear Reliability Factors are derived through a comparison of core physics measurements performed at the Vepco nuclear units and the corresponding design predictions of the Vepco physics design calculational models.

ACKNOWLEDGEMENTS

The author would like to express his thanks to Messrs. David A. Daniels, C. B. Franklin and M. L. Smith for their technical assistance in the development and preparation of this report. The author would also like to express his appreciation to a number of people who reviewed and provided comments on this report.

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SECTION 1 - INTRODUCTION

1.1 Purpose and Organization of the Report

This report addresses the derivation of Nuclear Reliability Factors (NRFs) to be applied to safety related design predictions performed with the Vepco physics design models for Vepco reload cycles. When feasible, the value of the NRF for a core physics parameter has been derived from a statistical comparison of core physics measurements with the corresponding predicted values. For those cases where the value of the parameter cannot be measured per se, the NRF is derived from analytical engineering arguments.

The NRFs described in this study will be used in all reload safety evaluation calculations performed with the Vepco physics design models as noted in Reference 1.

Values of the NRF for several of the core physics parameters have been previously reported in the topical reports describing the Vepco physics design models, (References 2, 3, and 4). These reports include a description of the cores of the Vepco nuclear units to which the NRFs are to be applied, as well as a description of the models used to perform the calculations. The present report summarizes the results from these previously published topicals as well as deriving the NRFs for parameters not previously reported. The parameters not previously reported are the Doppler temperature and power coefficients, delayed neutron parameters and the total peaking factor.

1.2 Definitions

The Nuclear Reliability Factor is defined as the allowance to be applied to a safety related physics design calculation to assure conservatism. The application of the NRF to a predicted value can be either multiplicative or additive depending on the physics parameter under consideration. For example, in the case of total peaking factor, the NRF is multiplicative. If the predicted value of the total peaking factor is FQ, the value used in the safety analysis would be:

$$NRF \times FQ \ .$$

An example of a parameter where the NRF is additive is the moderator temperature coefficient (MTC).

The application of the NRF to the predicted value is always in the conservative direction from a core safety consideration. For the case of a multiplicative NRF such as that for the cumulative integral bank worth, the NRF of 1.1 would be used to either increase or decrease the bank worth by 10% so as to yield a conservative value depending on the use of the parameter in the safety analysis. Likewise, for an additive NRF such as that for the MTC, the value used in the safety analysis would be

$$MTC \pm NRF,$$

depending on whether addition or subtraction was in the conservative direction.

The Nuclear Uncertainty Factor (NUF) is defined as the actual physics calculational uncertainty for a parameter derived from a statistical analysis performed on a comparison of measured and predicted results for the parameter. When a sufficiently large sample population is available for the comparison, the NUF is derived so that when it is applied to a predicted value, the result will be conservative compared to the corresponding measurement for 95% of the sample population with a 95% confidence level. Like the corresponding NRF, the NUF will be either multiplicative or additive depending on the parameter it was derived from. For example, if FQ is the predicted value for the total peaking factor and M is the corresponding measured value, then

$$NUF \times FQ > M$$

for 95% of the population with a 95% confidence level.

For those parameters for which a NUF has been derived, the corresponding NRF is chosen such that it is always more conservative then the NUF. For example, for the total peaking factor, the value of the NRF would be chosen such that:

$$NRF > NUF .$$

1.3 Summary of Results

Table 1-1 presents a summary of the Nuclear Reliability Factors derived for the Vepco physics design models. Included in the table is the calculational model used to calculate each parameter and the topical report from which the NRF for the parameter was derived.

For the Doppler temperature coefficient, Doppler power coefficient, effective delayed neutron fraction, and prompt neutron lifetime no direct measurements are available from which to derive the NRFs. Therefore, the NRFs for these parameters were derived from analytical engineering arguments.

The NRFs for the moderator temperature coefficient, critical soluble boron concentration, differential boron worth, individual integral bank worth, cumulative integral bank worth, and differential bank worth were derived from comparison with measuremenets performed at beginning-of-cycle (BOC), hot zero power (HZP) core conditions. It is to be noted that the moderator temperature coefficient results reflect measured and predicted values of the isothermal temperature coefficient since a direct measurement of the moderator temperature coefficient is not possible.

The NRFs for the radial peaking factor (FDH), the core average axial peaking factor (Fz), and the total peaking factor (FQ) are conservative with respect to a NUF which meets the 95%/95% acceptance criteria based on the sample population. The NUFs for FDH, Fz and FQ were derived from a comparison of the predicted power distributions with the measured power distributions calculated by the INCORE code (Reference 5).

The values of the NRFs which are preceded by a multiplication sign, "x", are multiplicative in a conservative direction. Otherwise the NRF is additive in a conservative direction.

TABLE 1-1

SUMMARY OF NUCLEAR RELIABILITY FACTORS

Parameter	Analytical Model	Reference	NRF
Individual Integral Bank Worth	PD207 discrete	FRD-19A	x 1.10
Cumulative Integral Bank Worth	PD207 discrete	FRD-19A	x 1.10
Differential Bank Worth	FLAME	FRD-24A	2 pcm/step
Critical Boron Concentration	PD207 discrete	FRD-19A	50 ppm
Differential Boron Worth	PD207 discrete	FRD-19A	x 1.05
Moderator Temperature Coefficient	PD207 one-zone	FRD-20A	3 pcm/°F
Doppler Temperature Coefficient	PD207 one-zone	FRD-45	x 1.10
Doppler Power Coefficient	PD207 one-zone	FRD-45	x 1.10
Effective Delayed Neutron Fraction	PD207 discrete	FRD-45	x 1.05
Prompt Neutron LIfetime	PD207 discrete	FRD-45	x 1.05

TABLE 1-1 (cont.)

Parameter	Analytical Model	Reference	NRF
FDH	PDQ07 discrete	FRD-19A	x 1.05
Fz	FLAME	FRD-24A	x 1.08
FQ	FLAME	FRD-45	x 1.075

Key:

pcm = percent mille
(1 pcm = change in reactivity of 10^{-5})

ppm = parts per million

SECTION 2 - MEASUREMENT AND CALCULATIONAL TECHNIQUES

2.1 Analytical Models

The major analytical models currently used in the design of a reload cycle are:

1. the Vepco PDQ07 discrete model,
2. the Vepco PDQ07 one-zone model, and
3. the Vepco FLAME model.

The Vepco PDQ07 models perform two-dimensional (x-y) geometry diffusion-depletion calculations for two neutron energy groups. These models utilize the NULIF code (Reference 6) and several auxiliary codes to generate and format the cross section input and to perform fuel assembly shuffles and other data management functions. The two models are differentiated according to their mesh size, (i.e., either a discrete mesh or a one-zone mesh.) The discrete mesh model generally has one mesh line per fuel pin, while the one-zone mesh model has a mesh size of 6x6 per fuel assembly. Either a quarter core symmetric two-dimensional geometry or a full core two-dimensional geometry may be specified. Effects of nonuniform moderator density and fuel temperatures are accounted for by thermal-hydraulic feedback. More complete descriptions of these models and their associated auxiliary codes are presented in References 3 and 4 for the discrete and one-zone models respectively.

The Vepco FLAME model is used to perform three-dimensional (x-y-z) geometry nodal power density and core reactivity calculations using a one energy group, modified diffusion theory. The model utilizes the NULIF code and several auxiliary codes to generate and format cross section input and to perform fuel assembly shuffles and other data management tasks. Each fuel assembly in the core is represented by one radial node and 32 axial nodes. Either a quarter core symmetric three-dimensional geometry or a full core three-dimensional geometry may be specified. As with the PDQ07 models, the effects of nonuniform moderator density and fuel temperature are accounted for by thermal hydraulic feedback. A more complete description of the model and the auxiliary codes used with it will be found in Reference 2.

2.2 Reactivity Computer and Delayed Neutron Data

Reactivity measurements for the Surry and North Anna nuclear power stations are obtained using a Westinghouse reactivity computer. The reactivity computer periodically samples neutron flux level signals from one of four ex-core detectors. Each ex-core detector consists of two five-foot ion chambers stacked one on top of the other. These signals are then converted to overall core reactivity by solving the monoenergetic point reactor kinetics (inhour) equations with six delayed neutron groups. The resulting calculated reactivity and flux level for the core are then displayed on a strip chart recorder.

The delayed neutron data for input to the reactivity computer are calculated by the PDQ07 discrete model. The delayed neutron fraction and decay constant for each of the six delayed neutron groups at a given core condition are calculated by weighting the delayed neutron fraction for each fissionable isotope for each group by the core integrated fission rate of that isotope. Normally, a single set of delayed neutron predictions will be used for all startup physics measurements at hot zero power (HZP) since sensitivity studies performed with the PDQ07 discrete model have indicated that the rodded configuration of the plant has minimal effect on the delayed neturon data, (typically less than 0.2%.)

The delayed neutron parameters of beta-effective (Beff) and prompt neutron lifetime (lp) are required for input to the reload cycle safety analysis. Beff is defined as the product of the core average delayed neturon fraction and the importance factor. The importance factor accounts for the decrease in effectiveness of the delayed neutrons when compared to prompt neutrons in causing fission and is set equal to 0.97. The prompt neutron lifetime is the time from neutron generation to absorption. It is a core average parameter calculated with the cross section generating code.

2.3 Temperature Coefficients

The isothermal temperature coefficient (ITC) is defined as the change in reactivity per degree change in the moderator, clad and fuel temperatures of the core. ITCs are measured at HZP for various core rodded configurations during the startup physics testing of each cycle. For each rodded configuration, reactivity measurements are made during a reactor coolant system (RCS) cooldown of approximately 5°F, a RCS heatup of approximately 10°F, and another RCS cooldown of approximately 5°F. The slopes of the change in core reactivity versus the change in the RCS temperature as plotted by the reactivity computer is then used to derive an average value for the ITC for the core configuration.

Prediction of the isothermal temperature coefficient is performed using the PDQ07 one-zone model. The change in core reactivity is calculated for changes in both the fuel and moderator temperatures of ±5°F about the HZP core average temperature of 547°F. This change in core reactivity divided by the total change in the fuel and moderator temperatures, (i.e., 10°F), yields the value of the ITC in units of pcm/°F. Calculation of the moderator temperature coefficient (MTC) is similar, but with the fuel temperature being frozen at the HZP value for both calculations. Therefore, the moderator temperature coefficient is defined as the change in core reactivity per change in °F of the core moderator temperature only. The Doppler temperature coefficient (DTC) is defined as the change in core reactivity per degree change in fuel temperature and is calculated by taking the difference between the predicted values of the ITC and MTC.

2.4 Power Coefficients

The total power coefficient is defined as the change in reactivity due to the combined effect of the moderator and fuel temperature change due to a change in core power level. The Doppler "only" power coefficient (DPC) relates to the change in power which produces a change in the fuel and clad temperature.

Power coefficient measurements are not routinely performed during the startup physics testing of Vepco nuclear units. The few measurements which have been made are highly unreliable, incorporating a design tolerance of ±30%. Furthermore, direct measurement of the Doppler "only" power coefficient is not possible. For these reasons no comparisons between measured and predicted power coefficients have been performed for the derivation of calculational uncertainties.

Power coefficient predictions are performed with the PDQ07 one-zone model. The DPC is found by subtracting the reactivity change with power due to a change in the moderator temperature only, (i.e., the moderator power coefficient), from the total power coefficient. To calculate the total power coefficient, PDQ07 one-zone model calculations are performed at ±10% power levels about the target power level, all other core conditions being held constant. Thermal hydraulic feedback effects are included in the calculation. The change in reactivity between the two calculations as a function of the change in power level yields the value of the total power coefficient in units of pcm/% power. The Doppler component of the power coefficient is predicted by performing a calculation at the +10% power level, but with the core inlet enthalpy value of the thermal hydraulic feedback part of the calculation adjusted so that the value of the moderator temperature is frozen to the value used in the -10% power level calculation. The resulting change in core reactivity as a function of power level between this calculation and the -10% power level calculation yields the value of the Doppler "only" power coefficient. The Doppler "only" power coefficient is then substracted from the predicted total power coefficient to find the moderator power coefficient.

2.5 Total Power Peaking Factors

The total peaking factor FQ is defined as the ratio of the peak power density in a fuel pellet to the core average power density. The maximum total peaking factor for the core, also referred to as the heat flux hot channel factor, is defined as the peak power density in the core divided by the core average power density. Values of FQ for an axial location z in the core, FQ(z), are calculated using the PDQ07 discrete and FLAME models. If FQ(x,y,z) is the nodewise three-dimensional power distribution for the node located at (x,y,z) calculated by the FLAME model, then the value of FQ at axial location z for radial location (x,y) is given by

$$FQ(z) = FQ(x,y,z) \times FDH(x,y) / RPD(x,y)$$

where FDH(x,y) is the peak radial power for the assembly and RPD(x,y) is the corresponding average assembly power calculated by the two-dimensional PDQ07 discrete model. The ratio FDH(x,y)/RPD(x,y) is referred to as the PDQ07 pin-to-box ratio.

Measured power distributions are calculated by the INCORE code based on detector readings obtained from the movable incore instrumentation system. This system consists of 50 movable detector locations as shown in Figure 2-1. Three-dimensional flux distributions are provided by the axial movement of the detectors in the instrumentation thimbles.

Input to the INCORE program consists of:

1. a description of the reactor conditions when measurements were made (such as power level, control rod positions, etc.)
2. incore detector readings including which flux thimbles were used and neutron cross sections of the sensor, and
3. fast and thermal fluxes, radial assembly average powers and radial pin powers calculated by the PD207 discrete model.

INCORE corrects raw pointwise flux measurements for leakage current, changes in power level between measurements, and relative detector sensitivities to determine the pointwise reaction rate in the flux thimbles. The measured reaction rates are then compared with expected values.

INCORE computes the relative local power produced by each fuel assembly, Pm, and the power in the peak fuel rod for each assembly. For the assemblies with monitored thimble locations, the assemblywise power is given by the equation

$$Pm = Rm \times Pp / Rp$$

where Rm is the measured reaction rate for the thimble, Pp is the power calculated for the thimble by the PD207 discrete model, and Rp is the reaction rate for the thimble calculated by the PD207 model. The values for Pm for all 157 assemblies in the core are normalized so that their sum equals unity.

INCORE calculates the peak FQ(z) for each assembly for 61 axial nodes.

FIGURE 2-1

MOVABLE INCORE DETECTOR LOCATIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1								**							
2										**					
3				**			**	**				**			
4					**			**		**					
5			**		**		**				**	**		**	
6					**			**		**					
7			**				**		**			**		**	
8	**		**		**					**			**	**	
9					**				**	**					**
10			**				**					**		**	
11					**			**		**	**				
12			**				**					**	**		
13								**		**					
14					**				**						
15							**								

** Incore Movable Detector Location

SECTION 3 - STATISTICAL ANALYSIS METHODOLOGY

3.1 Introduction

In order to derive the calculational uncertainty for the total peaking factor FQ, a statistical analysis was performed on the percent difference between the measured and predicted values for each core location; i.e.,

$$Xi = (Mi - Pi) \times 100\% / Mi$$

Here Mi is the measured value for observation i, Pi is the predicted value for observation i and Xi is the percent difference between the measurement and prediction for the ith observation. Xi is assumed to be a normally distributed random variable whose mean X and standard deviation S are defined as:

$$X = \mathrm{SUM}\ (Xi) / n \qquad (5\text{-}1)$$

$$S^2 = \mathrm{SUM}\ (Xi - X)^2 / (n - 1) \qquad (5\text{-}2)$$

where the notation SUM indicates a summation over values of i from 1 to n of the quantity in parentheses which immediately follows.

In general, the standard deviation as calculated above includes the statistical uncertainties due to both measurement and calculation. That is, the variance of Xi is given as:

$$S^2 = Sm^2 + Sp^2$$

where Sm^2 is the variance due only to measurement uncertainty and Sp^2 is the variance due only to calculational uncertainty. Therefore, any standard deviation for calculational uncertainty derived using equation (5-2) is conservative since an additional margin for measurement uncertainty is included.

3.2 Tests for Normality

The distribution of the differences, Xi, for the total peaking factor was tested for normality using the method outlined in Reference 7. This method, the Kolmogorov-Smirnov test (hereafter referred to as the D-test) is valid for distributions containing over 50 observations. All tests were performed for a 95% confidence level with a .05 level of significance being considered as adequate for rejection of the assumption of normality for the data .

The D-test compares the value of a test statistic, D, for the sample distribution with the value of the test statistic for a normal distribution of the same size. Tables 4-3 through 4-6 provide the normality test results for the difference distributions used to derive the reliability factors. The assumption of normality is rejected when the computed values of D are less than the test D value which corresponds to a 95% confidence level with a .05 level of significance for the indicated sample size n. These results are summarized under the columns labeled PROB>D. A value in the column of less than .05 indicates a rejection of the null hypothesis--i.e., the sample is considered to be nonnormal.

3.3 Derivation of Nuclear Uncertainty Factors

For the total peaking factor, nuclear reliability factors were derived using one sided upper tolerance limit methodology, (Reference 8). Assuming that the sample distribution, Xi, is normal, the one sided upper tolerance limit TL is defined as:

$$TL = X + (K \times S) \qquad (5\text{-}7)$$

where K is the one-sided tolerance factor. K is chosen such that 95% of the population is less than the value of TL with a 95% confidence level. The value of K is dependent on the sample size n used to derive TL, (Reference 8). In cases where the value of the mean (or bias) reduces the value of TL in a non-conservative direction, (i.e., a negative bias for total peaking factor), the value of the mean is set equal to zero to insure conservatism.

The value of the Nuclear Uncertainty Factor NUF is derived from the one sided upper tolerance limit as

$$NUF = 1 + (TL/100) \qquad (5\text{-}8)$$

For example if the value of TL is 10%, the NUF is 1.1.

SECTION 4 - RESULTS

4.1 Reactivity and Kinetic Parameters

4.1.1 Doppler Temperature and Power Coefficient

Direct measurement of Doppler reactivity effects in the core is not feasible due to the coupling between changes in the core's fuel temperature and the core's moderator properties. Most of the measurement/prediction uncertainty in the isothermal temperature coefficient can be attributed to the moderator component since the value of the Doppler component is of the order of -2 pcm/°F for Vepco nuclear units and shows little variation over the lifetime of a cycle. Therefore, a Nuclear Reliability Factor of 1.1 (i.e., 10%) will be assumed for the Doppler temperature coefficient.

Measurements of the total power coefficient have been performed during the startup physics testing of Surry 1 Cycle 4, Surry 2 Cycle 4, Surry 1 Cycle 5 and North Anna 1 Cycle 1 for a total of 14 measurements. Since the startup of Surry 1 Cycle 5 the power coefficient measurement has been discontinued from the startup physics testing program for Vepco nuclear units.

The Doppler component of the power coefficient cannot be measured directly. Due to the difficulty of obtaining accurate measurements of the total power coefficient, the design tolerance for the above Vepco measurements was set at

±30%. This large measurement uncertainty along with the small size of the available data base makes a derivation of an uncertainty factor for the Doppler component of the power coefficient based on comparison of measurement and prediction of questionable value. Therefore, a Nuclear Reliability Factor for the Doppler "only" power coefficient is conservatively chosen to be 10%.

4.1.2 Delayed Neutron Parameters

The delayed neutron parameters input to the reload cycle safety analysis are the effective delayed neutron fraction Beff and the prompt neutron lifetime lp. Beff is the more important factor in determining the reliability of core physics design predictions; however, measurements of this parameter are not available for comparing with predictions in order to derive an uncertainty factor.

The major uncertainties associated with the prediction of Beff are the experimental values of the delayed neutron fractions and the percursor decay constants for each delayed neutron group input to the PDQ07 discrete model, the predicted core nuclide concentrations (in particular U^{235}, U^{238} and Pu^{239}), the calculation of the fission sharing of each fissionable isotope for the weighting of the delayed neutron fraction of the isotopes, and the estimate of the importance factor. The experimental uncertainty for the delayed neutron fractions and decay constants are on the order of 4%, (Reference 9). The low uncertainty factor associated with the prediction of the radial peaking factors over cycle lifetime by the PDQ07 discrete model (less than 5%) implies a similar accuracy in the prediction of the core nuclide concentrations of U^{235}, U^{238} and Pu^{239} and the fission sharing for the isotopes. Finally, Beff is relatively insensitive to uncertainty in the importance factor since a typical value for the importance factor, (e.g., 0.97), indicates a reduction in the core average delayed neutron fraction of only 3%.

From these considerations a Nuclear Reliability Factor for Beff and lp of 5% appears to be a reasonably conservative estimate.

4.2 Power Peaking Factors

4.2.1 Data Base Considerations

Uncertainty factors for the total power peaking factors FQ were derived from a comparison of measurements and predictions based on a one-sided 95%/95% upper tolerance limit.

The data base consisted of three Vepco nuclear cycles: North Anna 1 Cycle 1, Surry 2 Cycle 4 and Surry 1 Cycle 5. These cycles were the latest Vepco cycles to have completed operation at the time this report was in preparation. One additional cycle, North Anna 1 Cycle 2, had also completed operation, but due to the radial flux tilt problem experienced during the initial operation of the cycle, it was excluded from the data base. The two Surry cycles are 18-month cycles with large lumped burnable poison loadings. Surry 2 Cycle 4 employed an out/in fuel loading strategy. Surry 1 Cycle 5 employed an in/out fuel loading strategy and is representative of the future fuel loading strategy being planned for Vepco nuclear units. North Anna 1 Cycle 1 was an initial core 18 month cycle with a large loading of lumped burnable poison.

Measured total peaking factors were calculated by the INCORE code. Table 4-1 presents a listing of the INCORE flux maps included in the data base. Each cycle includes flux maps at

HZP, BOC for both a rodded and unrodded core configuration, a map in the mid power range for an essentially unrodded core condition near BOC and a selection of HFP flux maps throughout the remaining cycle lifetime. In addition two mid power range maps near BOC for a pseudo-ejected rod test and a dropped rod test are included for North Anna 1 Cycle 1. Measured peaking factors are compared only for monitored thimble locations in order to avoid the additional uncertainty introduced by the INCORE code in interpolating peaking factors for the non-monitored assembly locations. Thimble readings for a flux map are normally discarded if the readings are incomplete or if the thimble suffered severe misalignment during the measurement. Such thimble locations have been deleted from the data base used to derive the peaking factor calculational uncertainties.

In order to generate total peaking factor predictions, concentration files for FLAME were created at each cycle burnup at which a flux map was taken. Normally the FLAME depletion was performed at an ARO, HFP core condition. However, unlike a two-dimensional calculation, a three-dimensional modeling of the core is sensitive to the actual changes in core conditions which occurred during the burnup depletion. This sensitivity can be monitored by comparing the measured and predicted axial offset (A.O.) for a given flux map core condition. A large difference between the axial offsets is indicative of oversimplified modeling of the core history prior to the time the flux map was taken. The severity of this problem was quantified by comparing the predicted and measured axial offsets for each flux map. If the measured/predicted difference was on the order of 3% or greater a more accurate modeling of the core history was performed by depleting the previous burnup step with the D bank partially inserted. The FLAME calculation for each flux map was then performed at the core condition of the flux map. The total relative power distribution in each three-dimensional node of the FLAME calculation is converted to a total peaking factor by multiplying by the two-dimensional PDQ07 pin-to-box ratio at the appropriate core conditions for the axial region.

Total peaking factor comparisons are performed for 6 axial planes for a North Anna unit and 5 axial plans for a Surry Unit. These axial planes have been selected at locations approximately halfway between neighboring assembly grid straps as shown in Table 4-2. Table 4-2 gives the axial locations of the center of the grids and the locations of the center of the INCORE or FLAME axial nodes used in the analysis in terms of the percent of active core height as measured from the bottom of the active core. INCORE nodes are number from 1 to 61 with node 1 being at the top of the core. The planes selected for the measurement/prediction comparisons correspond to the INCORE nodes listed in Table 4-2. The FLAME model contains 32 axial nodes numbered from the bottom to the top of the core with axial node 1 being at the bottom of the core. In order to derive a predicted FQ value for the percent of core height corresponding to the selected INCORE plane a Lagrange interpolation was performed on the predicted total peaking factors for the 3 axial FLAME nodes which most closely bracketed each selected INCORE plane. These axial FLAME nodes are listed in Table 4-2.

Axial locations approximately halfway between the grids were chosen for the comparisons in order to add conservatism to the derivation of the total peaking factors calculational uncertainty. Since the FLAME model does not model the grids, the predicted axial power distribution is not depressed at the grid locations. This results in a tendency for the maximum difference between measured and predicted FQ to occur about halfway between the grid locations where the measured value usually exceeds the predicted. Hence, using these locations for the data base results in an additional conservatism to be added to the uncertainty factor and removes the necessity of having to apply a special grid correction factor to a predicted value at a between-the-grid location to allow for the unmodeled grid depression effect. Figure 4-1 provides an example of this phenomena in plotting the measured and predicted axial power distribution for a specific monitored thimble location for a North Anna 1 Cycle 1 flux map.

Only radial core locations corresponding to accepted monitored thimble locations were included in the data base.

Since only peaking factors whose relative power distributions (RPDs) are greater than the core average are of interest in the safety analysis of a reload core, only pairs of observations where both the predicted and measured RPDs are ≥ 1.0 have been included in the data base. This approach excludes large percent difference values which often result from comparing the relatively low RPDs that tend to occur near the radial core periphery and at the top and bottom of the core due to the steeper power distribution slopes in these areas.

TABLE 4-1

TOTAL PEAKING FACTOR DATA BASE

Cycle	Flux Map #	% Power Level	Cycle Burnup MWD/MTU	Rodded Condition	Number of Monitored Thimbles
N1C1	1	4	0	D/228	48
N1C1	2	4	0	D/0	46
N1C1	5	30	50	D/195	48
N1C1	6	30	50	Ejected Rod	48
N1C1	10	49	50	Dropped Rod	48
N1C1	15	73	150	D/215	38
N1C1	37	96	3047	D/213	39
N1C1	50	96	7340	D/205	38
N1C1	53	97	9135	D/220	39
N1C1	58	100	11003	D/228	38
N1C1	64	100	12960	D/227	46
N1C1	75	97	15142	D/224	49
S1C5	1	0	0	D/218	40
S1C5	3	4	0	D/0,C/219	43
S1C5	4	50	0	D/200	43
S1C5	12	100	2123	D/218	42
S1C5	17	100	4072	D/223	43
S1C5	19	100	5270	D/224	43
S1C5	23	100	7411	D/224	43
S1C5	26	100	8973	D/226	42
S1C5	30	100	10125	D/226	43
S1C5	32	100	11580	D/216	42

TABLE 4-1 (cont.)

Cycle	Flux Map #	% Power Level	Cycle Burnup MWD/MTU	Rodded Condition	Number of Monitored Thimbles
S2C4	1	4	0	D/218	47
S2C4	2	7	0	D/0	47
S2C4	5	61	8	D/155	47
S2C4	11	100	1800	D/225	45
S2C4	18	100	5266	D/224	45
S2C4	22	100	6968	D/210	43
S2C4	27	100	9250	D/202	42
S2C4	30	100	11006	D/223	49
S2C4	36	100	13200	D/222	49

TABLE 4-2

AXIAL GEOMETRY FOR POWER DISTRIBUTION COMPARISONS

North Anna Units 1 and 2		Surry Units 1 and 2	
% Core Height	* Description	% Core Height	Description
103.5	Grid # 1	104.0	Grid # 1
89.2	Grid # 2	90.8	Grid # 2
85.9	FLAME Node 28	85.9	FLAME Node 28
82.8	FLAME Node 27	83.3	INCORE Node 11
81.7	INCORE Node 12	82.8	FLAME Node 27
79.7	FLAME Node 26	79.7	FLAME Node 26
74.9	Grid # 3	72.6	Grid # 3
70.3	FLAME Node 23	67.2	FLAME Node 22
68.3	INCORE Node 20	64.1	FLAME Node 21
67.2	FLAME Node 22	63.3	INCORE Node 23
64.1	FLAME Node 21	60.9	FLAME Node 20
60.6	Grid # 4	54.4	Grid # 4
57.8	FLAME Node 19	48.4	FLAME Node 16
54.7	FLAME Node 18	45.3	FLAME Node 15
53.3	INCORE Node 29	45.0	INCORE Node 34
51.6	FLAME Node 17	42.2	FLAME Node 14
46.4	Grid # 5	36.2	Grid # 5
42.2	FLAME Node 14	29.7	FLAME Node 10
39.1	FLAME Node 13	26.7	INCORE Node 45
38.3	INCORE Node 38	26.6	FLAME Node 9
35.9	FLAME Node 12	23.4	FLAME Node 8
32.1	Grid # 6	18.0	Grid # 6
29.7	FLAME Node 10	17.2	FLAME Node 6
26.6	FLAME Node 9	14.1	FLAME Node 5
25.0	INCORE Node 46	13.3	INCORE Node 53
23.4	FLAME Node 8	10.9	FLAME Node 4
17.8	Grid # 7	1.3	Grid # 7

TABLE 4-2 (cont.)

North Anna Units 1 and 2

% Core * Height	Description
17.2	FLAME Node 6
15.0	INCORE Node 52
14.1	FLAME Node 5
10.9	FLAME Node 4
0.8	Grid # 8

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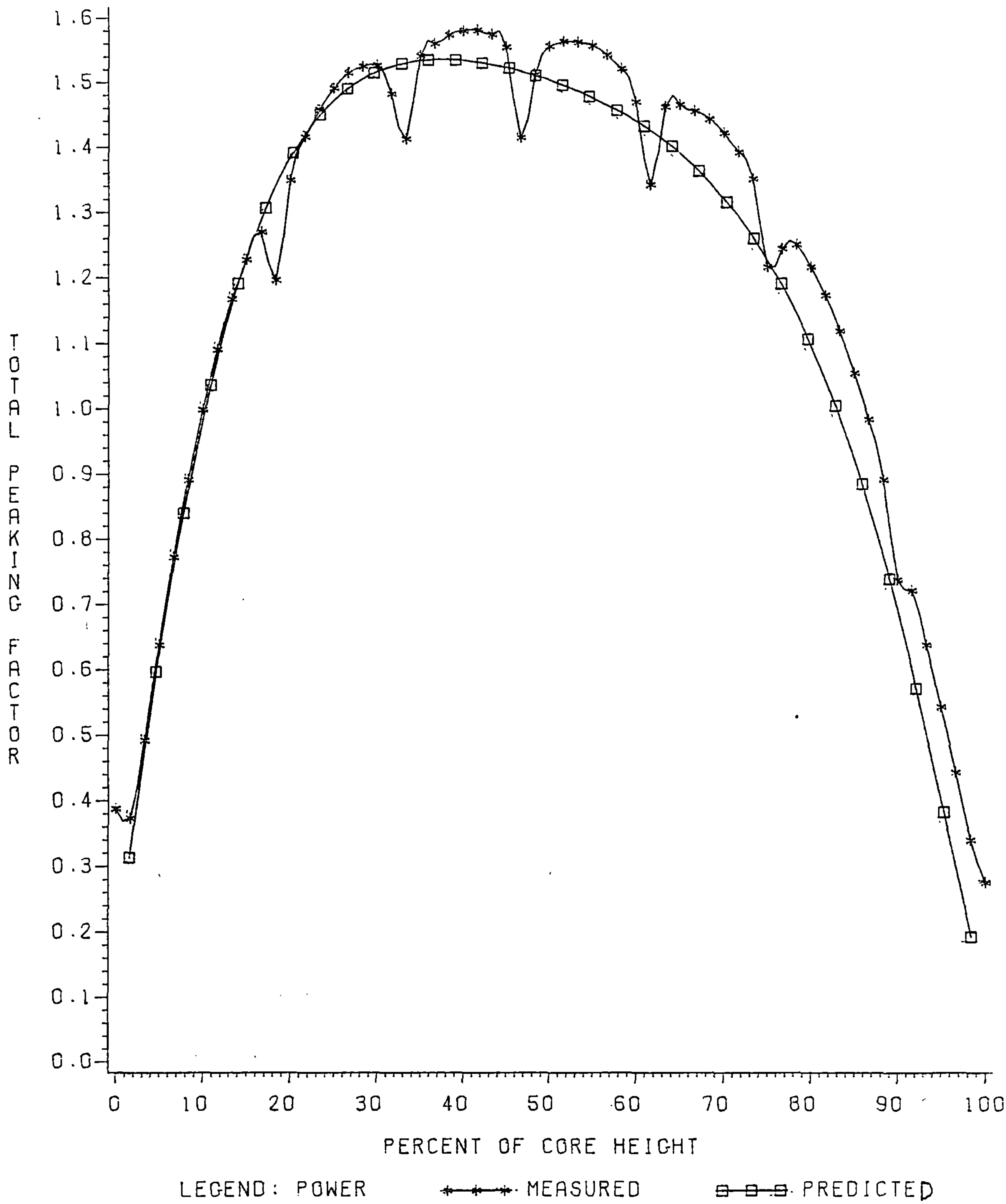
* % Core Height is measured from the bottom of the core.

FIGURE 4-1

TYPICAL MEASURED/PREDICTED AXIAL POWER DISTRIBUTION COMPARISON

NORTH ANNA 1 CYCLE 1 FLUX MAP 37 -- THIMBLE LOCATION H13



4.2.2 Results

Tables 4-3 through 4-5 present a summary of the total peaking factor comparisons for each cycle on a flux map by flux map basis. Included in the tables is a listing of the measured and predicted axial offsets (A.O) and the arithmetic difference between the two for each map.

Figures 4-2 through 4-4 present histograms of the comparison results for the total peaking factors for each cycle. The histograms may be used as a visual check on the normality of each percent difference distribution.

Table 4-6 presents a summary of the peaking factor data base statistics. No problem in the normality testing of any of the cycles for the total peaking factor was found although results for individual maps for a particular cycle often failed the normality test.

Based on the 95%/95% uncertainty factors listed in Table 4-6 it is concluded that an acceptable Reliability Factor for the total peaking factor is 1.075.

TABLE 4-3

TOTAL PEAKING FACTOR RESULTS -- NORTH ANNA 1 CYCLE 1

For Measured and Predicted $F_Q \geq 1.0$

Map #	n	X Mean (%)	S Std. Dev. (%)	PROB>D	Min. % Diff.	Max. % Diff.	Meas. A.O.	Pred. A.O.	A.O. Diff.
1	232	1.19	3.62	>0.15	-6.75	8.89	0.6	-0.2	0.8
2	231	0.70	5.06	>0.15	-10.53	12.18	-0.1	-0.5	0.4
5	241	1.29	3.97	>0.15	-7.24	11.43	8.3	7.3	1.0
6	241	0.94	3.62	0.037	-7.20	8.98	6.1	7.7	-1.6
10	252	0.97	3.86	0.072	-8.70	10.59	-4.4	-3.8	-0.6
15	213	-0.03	4.43	>0.15	-10.24	11.76	-3.3	-5.3	2.0
37	215	0.67	3.67	<0.01	-8.06	9.21	-5.6	-7.4	1.8
50	218	0.68	2.45	0.031	-8.00	5.91	-7.4	-6.9	-0.5
53	224	0.37	2.09	0.093	-5.06	5.34	-2.7	-3.2	0.5
58	216	-0.35	4.38	<0.01	-9.30	9.25	0.4	-3.3	3.7
64	261	-0.14	2.89	<0.01	-6.44	7.78	-2.4	-3.5	1.1
75	278	-0.35	4.21	>0.15	-10.06	11.02	0.5	-2.3	2.8

Summary statistics for North Anna 1 Cycle 1 F_Q data base:

% Diff. = (Measured - Predicted) x 100% / Measured

n = 2822
Mean = 0.49%
Standard Deviation = 3.81%
PROB>D = >0.15

TABLE 4-4

TOTAL PEAKING FACTOR RESULTS -- SURRY 1 CYCLE 5

For Measured and Predicted $F_Q \geq 1.0$

Map #	n	X Mean (%)	S Std. Dev. (%)	PROB>D	Min. % Diff.	Max. % Diff.	Meas. A.O.	Pred. A.O.	A.O. Diff.
1	129	0.76	4.47	>0.15	-12.11	15.30	27.4	24.8	2.6
3	138	1.22	2.99	>0.15	-4.77	9.51	22.7	22.4	0.3
4	166	1.18	3.66	>0.15	-7.99	9.98	6.3	8.2	-1.9
12	170	0.88	3.91	>0.15	-8.60	11.25	-1.6	-3.8	2.2
17	173	0.93	4.44	>0.15	-8.91	10.26	-1.6	-4.7	3.1
19	171	1.28	3.98	<0.01	-5.58	10.98	-2.5	-5.2	2.7
23	175	0.99	2.96	0.018	-5.95	7.99	-3.2	-3.7	0.5
26	175	0.89	2.27	0.047	-4.41	7.25	-2.9	-3.3	0.4
30	175	1.54	3.73	0.105	-7.80	8.97	-3.7	-1.6	-2.1
32	175	1.17	3.10	>0.15	-6.24	8.61	-3.5	-2.5	-1.0

Summary statistics for Surry 1 Cycle 5 F_Q data base:

% Diff. = (Measured - Predicted) x 100% / Measured

n = 1647
Mean = 1.09%
Standard Deviation = 3.59%
PROB>D = >0.15

TABLE 4-5

TOTAL PEAKING FACTOR RESULTS -- SURRY 2 CYCLE 4

For Measured and Predicted FQ ≥ 1.0

Map #	n	X Mean (%)	S Std. Dev. (%)	PROB>D	Min. % Diff.	Max. % Diff.	Meas. A.O.	Pred. A.O.	A.O. Diff.
1	157	1.35	4.00	>0.15	-9.45	10.57	21.9	23.5	-1.6
2	163	1.06	4.53	>0.15	-11.47	12.93	17.6	20.3	-2.7
5	178	-0.33	3.54	>0.15	-7.47	7.81	-10.4	-8.1	-2.3
11	203	0.30	4.19	0.092	-11.02	11.14	-2.9	-5.2	2.3
18	205	1.11	3.28	0.130	-7.39	9.55	-2.6	-4.2	1.6
22	203	1.51	3.33	>0.15	-6.65	9.76	-4.0	-5.4	1.4
27	185	2.09	3.24	0.078	-5.55	9.92	-5.8	-5.3	-0.5
30	217	1.52	2.58	0.145	-4.81	8.56	-1.7	-2.8	1.1
36	213	1.30	2.98	<0.01	-5.91	7.92	-1.4	-3.3	1.9

Summary statistics for Surry 2 Cycle 4 FQ data base:

% Diff. = (Measured - Predicted) x 100% / Measured

n = 1724
Mean = 1.11%
Standard Deviation = 3.58%
PROB>D = >0.15

TABLE 4-6

SUMMARY OF TOTAL PEAKING FACTOR STATISTICS

Cycle	n	X Mean (%)	S Std. Dev. (%)	Normality Test PROB>D	95%/95% Uncertainty Factor
N1C1	2822	0.49	3.81	>0.15	1.069
S1C5	1647	1.09	3.59	>0.15	1.072
S2C4	1724	1.11	3.58	>0.15	1.072

FIGURE 4-2

HISTOGRAM OF TOTAL PEAKING FACTOR RESULTS -- NORTH ANNA 1 CYCLE 1
PERCENT DIFFERENCE DISTRIBUTION FOR MEASURED/PREDICTED FQ > 1.0

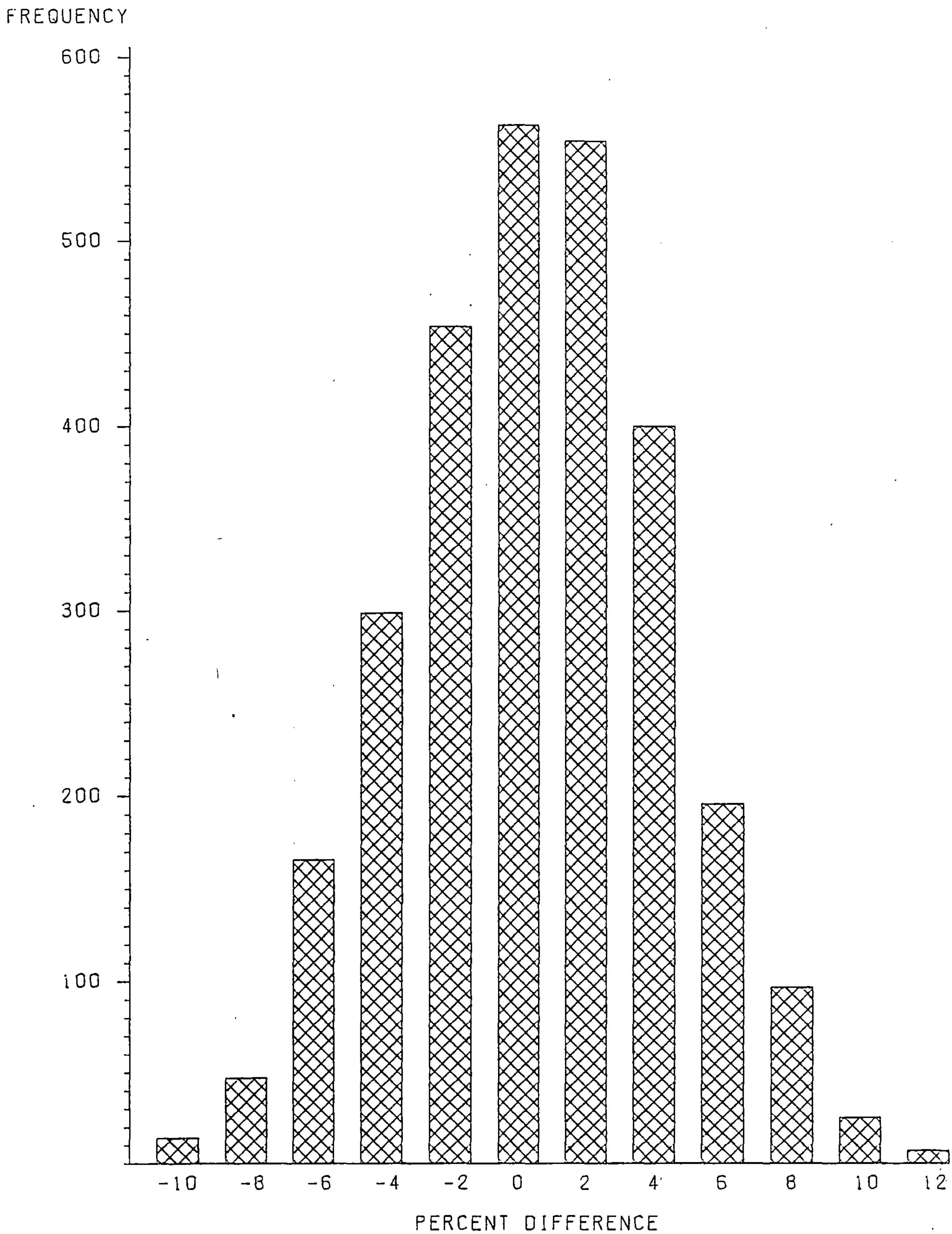


FIGURE 4-3

HISTOGRAM OF TOTAL PEAKING FACTOR RESULTS -- SURRY 1 CYCLE 5
PERCENT DIFFERENCE DISTRIBUTION FOR MEASURED/PREDICTED FQ > 1.0

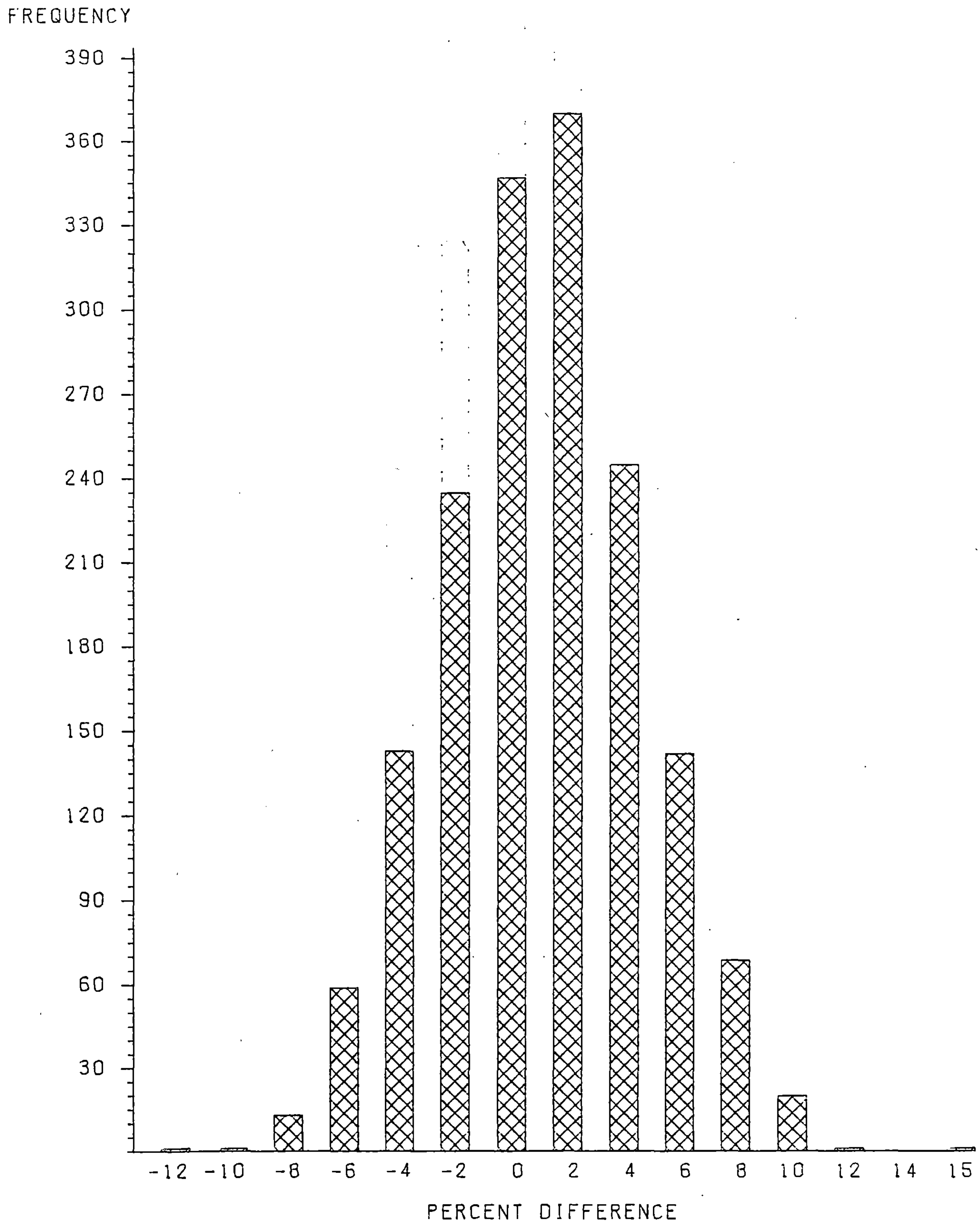
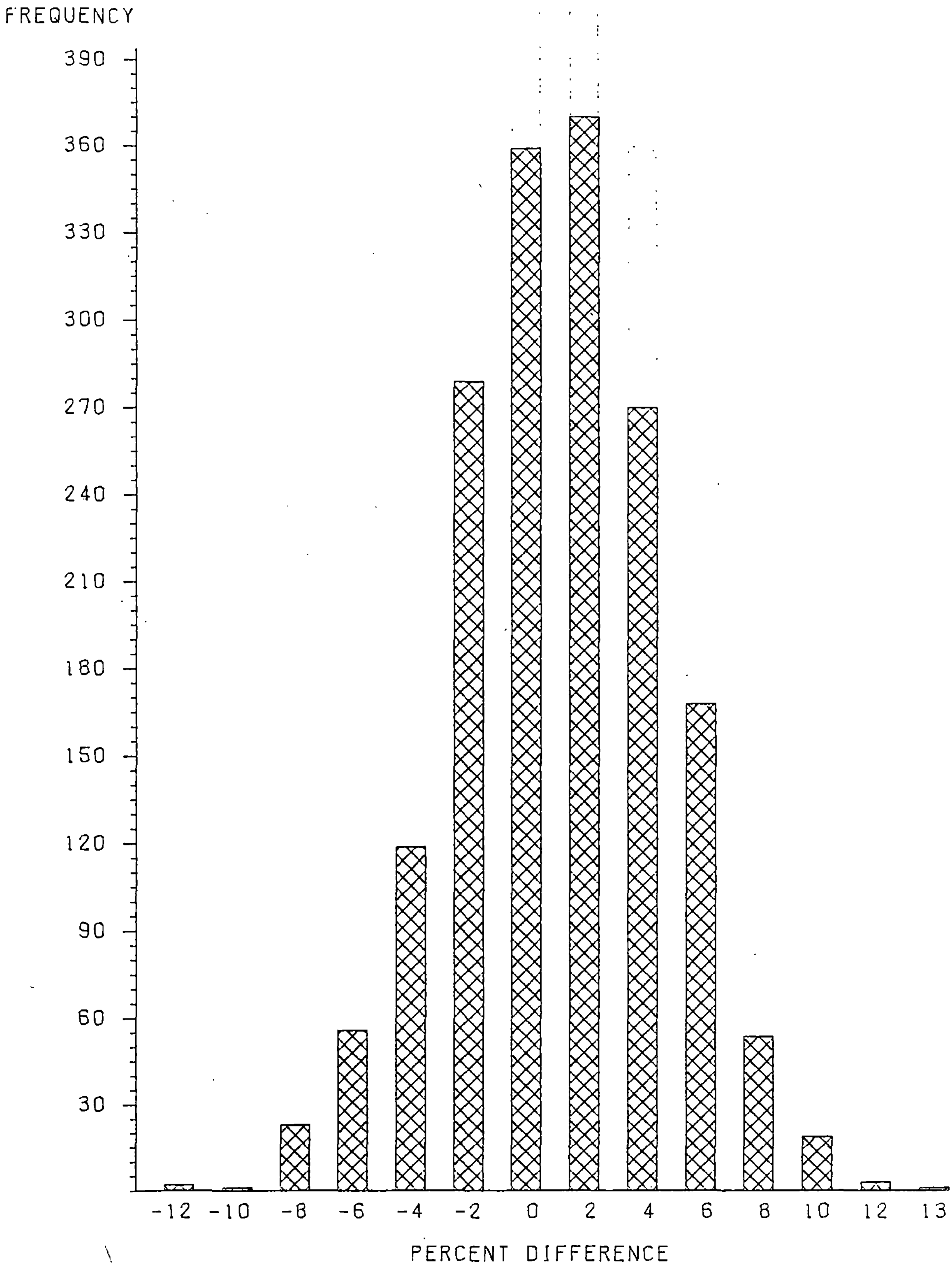


FIGURE 4-4

HISTOGRAM OF TOTAL PEAKING FACTOR RESULTS -- SURRY 2 CYCLE 4
PERCENT DIFFERENCE DISTRIBUTION FOR MEASURED/PREDICTED FQ > 1.0



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