CEN-89(F)-NP

Increased Water Hole Peaking in Operating Reactors (St. Lucie-1)

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Introduction

This document contains the information presented to the NRC on December 16, 1977 and January 18, 1978. Section 1 presents the results of investigations of pin power peaking in assemblies with CEA waterholes using a newly developed multigroup transport theory calculation method. These results indicate that the current standard design model is underpredicting the power in fuel pins adjacent to CEA waterholes by approximately 4.5%. Section 2 describes conservatisms in the analysis package used to accommodate the increased water hole peaking for first cycle of St. Lucie Unit 1. Explicit use of the methods described will be employed in reload designs starting with Cycle 2. Sections 3 and 4 further describe improvements in thermal margin/low pressure trip synthesis and a partial credit for TORC/CE-1 for DNB LCO respectively that are employed in the analysis package discussed in Section 3.

Table of Contents

New Peaking Model

Slide 1

The purpose of this presentation is to present the results of recent investigations of pin power peaking in assemblies with CEA waterholes using a newly developed multigroup transport theory calculation for the fuel assembly. These results, which include a new evaluation of the available experimental data base, indicate that the standard design model is underpredicting the power in pir adjacent to CEA waterholes by about 4.5%.

We will also describe how the increased pin peaking will be incorporated in the design power distribution calculations.

We will also present here an evaluation of the effects of the increased peaking on current operating plants. We do not plan to change the methods used on these plants now, but rather will incorporate the new methods in future reload submittals. We will show sufficient margin presently exists to cover these effects and that there are no safety concerns with continued operation of these plants.

Slide 2

The new peaking model will be incorporated in the overall setpoint analysis along with the use of a reduced power distribution measurement uncertainty using the values justified in C-E Topical Reports on this subject (CENPD-153 and CENPD-145).

An evaluation of the current cycles of all C-E operating plants (except Maine Yankee and Palisades) shows that the increased pin peaking can be accounted for with available margins in the overall setpoint analysis. There are no safety considerations or power capability restrictions and no significant technical specification changes are required. As we will describe later, a small (0.5%) additional conservatism will be added to the monitoring limits for peak KW/FT. An evaluation of the increased peaking will be shown for each cycle as the action requirement depends on the power distribution uncertainty presently in effect on the current or upcoming cycle.

Slide 3

For purposes of the safety analysis and power distribution measurement uncertainty assessment, the total 3-D nuclear peaking factor is constructed from 3 components. The fuel assembly axial peaking factor (F_Z) is obtained from the normalized axial shape in each fuel assembly and its uncertainty is evaluated from signal

reproducibility and the precision of the axial fitting technique used to construct a continuous distribution from the 4 rhodium incore detector segments. The radial power distribution (F_R), which represents the average fuel assembly axially integrated power, has an uncertainty component that is evaluated from comparisons of calculated and measured incore instrument signal radial distributions. This is the uncertainty component that makes use of reactor operating data.

The third factor is the maximum pin power to fuel assembly average power and is the subject of this presentation. Since the incore instrument measures instrument reaction rates rather than the desired maximum pin power, it is necessary to rely on calculated factors to infer the pin power in an operating reactor. The uncertainty component associated with these calculated factors is evaluated from comparison of calculated and measured fuel pin relative powers in separate critical experiments. We have used such critical experiments to determine the uncertainty associated with use of the new pin power peaking model.

<u>Slide 4</u>

The incore instrument system in C-E reactors was initially installed to provide general information on the details of the core power distribution and was not used to demonstrate compliance with any Technical Specification operating limits. An uncertainty of 10% was originally assigned by NRC for the initial use of the system. With the introduction of very low KW/FT limits associated with the new LOCA requirements, it became desirable to make direct use of the incore instruments to demonstrate plant operation within these limits. C-E submitted a report to NRC in late 1973 justifying an uncertainty of 6.5% when using the incore instruments for this purpose. NRC assigned an uncertainty of 8% which was the highest for all PWR vendors. Since this time, C-E has submitted detailed topical reports on both the basic uncertainty of the fixed instrument system (CENPD-153) and the uncertainty associated with use of the C-E incore instrument analysis code CENPD-145. These reports are still being reviewed by NRC.

<u>Slide 5</u>

A continuing effort has been underway at C-E to improve all aspects of the physics design models including pin power peaking, rod worths, temperature coefficients, and reactivity (boron) rundown. In 1968, C-E performed a set of critical experiments with measurements of pin power peaking and rod worths and assemblies with CEA waterholes. The evaluation of pin peaking and rod worth showed good agreement with design models in use at that time (approximately
3 [ ]on pin peaks near CEA waterholes). Subsequent effort then focused on improved agreement on reactivity rundown and temperature coefficients. The detailed treatment of the [illegible] CEA waterholes has a large impact on these quantities as well as on the power peaking in adjacent fuel pins and changes made

here can impact the predictions of pin peaking.

In 1974, additional critical experiments were performed for UO_2 and mixed oxide lattices. These experiments included the first hot conditions and design model comparisons shifted to this newer set of experiments. These analyses, using the most recent improvements in the design model, did suggest a small
3 (approximately [ ] underprediction of power in pins near CEA waterholes. These comparisons are reported in CENPD-153; this small underprediction was included as part of the overall uncertainty. One of the NRC questions on CENPD-153 questioned whether or not the underprediction at waterholes should be treated separately as a bias rather than as part of the overall uncertainty, since the peak power pins were expected to be adjacent to CEA waterholes.

A new version of the DIT code, which contains an 85-group transport theory calculations of the entire fuel assembly including the detailed interaction between waterholes, power pins, and poison shims, has now been applied to the prediction of waterhole peaking. These first results indicated that the current design model underpredicted pin peaking near CEA waterholes by amounts considerably larger than indicated by the comparisons with the 1974 experiments. In order to resolve this, the 1968 experiments have been reanalyzed with the standard design model and with DIT. An evaluation with the full data base, shows that DIT over-
3 predicts pin peaking by [ ] and the standard design model underpredicts pin peaking by 4 to 5%. These results led to the current action (described on the following slides) to account for this effect.

<u>Slide 6.</u>

This slide outlines the procedure that will be employed to account for higher pin peaks adjacent to CEA waterholes in the standard design models. The standard design diffusion theory model (PDQ) is used for quarter core pin-by-pin calculations of the power distribution, PDQ can also be applied to calculations of specific fuel assemblies using appropriate operating conditions. The transport theory DIT calculation can only be run for the assembly geometry. While it is possible to adjust diffusion theory constants so that pin power peaking will match transport theory results, we have not yet developed a prescription that will provide the simultaneous prediction of power peaking, temperature coefficients and reactivity rundown in diffusion theory to the desired accuracies. For this reason, our present plan then is to use the transport theory DIT code to define a pin peaking bias that will be applied to the results of the standard design model calculations.

This will be accomplished in two steps; first, DIT is used for an analysis of the critical lattice experiments to define a calculational uncertainty and bias for the best method. In the second step, the different fuel assemblies (varying enrichment, poison pin loadings, etc) are analyzed with both DIT and the standard design PDQ models and the difference determined for each type of fuel assembly. These results are then combined with the results of the first step to produce a bias and uncertainty for the design method.

A consistency check of this procedure has been made by direct evaluation of the critical experiments with the standard design model. As expected, the uncertainty and bias established from this approach is comparable to that obtained from the described procedure. The bias is applied to the design model calculations for subsequent use in the safety analysis and generation of coefficient libraries for the incore instrument power distribution monitoring conditions.

Slide 7

The results of application of the above procedure for the high enrichment in a typical C-E reload batch is shown here. These results are obtained by depleting an assembly calculation using soluble boron representative of the core variation with cycle depletion. As can be seen on the slide, DIT gives
3 a pin peak of about[ ]above the standard PDQ model. As stated earlier,
3 DIT is expected to overpredict the pin peak by about[ ]. This bias is subtracted from the upper curve to obtain the (approximately 4.5%) bias to be applied to the design PDQ. In this case, the difference between PDQ and DIT remains fairly constant throughout the cycle as a result of compensation between burndown of the pin peaks and reduction of soluble boron in the waterholes.

Slide 8

Results of similar calculations are shown for C-fuel. In this case, the similar bias has a slight downward trend with burnup in the first cycle and when the soluble boron level is increased to a value typical of beginning of second cycle, the bias shows a step decrease of about 1%. This is a result of a smaller difference between the standard PDQ model and the DIT transport theory calculation when the waterhole contains a large amount of poison.

Similar curves will be generated for each specific fuel assembly in C-E reactors.

Slide 9

This slide is another presentation of the procedure described on Slide 6.

Slide 10

The DIT code includes an 85-group neutron spectrum calculation with a spatial geometry that accounts for interaction effects between waterholes, fuel shims, and poison pins. The assembly spatial calculation is based on integral transport theory and includes an explicit representation of the fuel pin and its surrounding moderator. The fuel pins are not homogenized and a detailed mesh structure is used inside the pins to fully account for the details of the flux distribution within the fuel pin and its surrounding moderator.

Several approximations are employed in DIT to minimize computer storage requirements and to achieve reasonable running times. These approximations were evaluated and justified through comparisons with explicit Monte Carlo calculations. However, they will introduce small inaccuracies or biases and it is reasonable to expect DIT will overpredict pin peaking near waterholes. This is what is found in the evaluation of critical experiments.

Slide 11

This slide provides the characteristics of each of the critical experiments and the number of pins adjacent to waterholes that were measured in each experiment. It is noted that a range of enrichments, volume fractions, temperature and boron PPM levels, have been covered. When account is taken of the difference in moderator temperature, C-E experiments #53 and #56 are closest to the hydrogen to uranium ratio found in the C-E operating lattices. The BNWL lattices have volume ratios considerably greater than encountered in the C-E lattices and so have been excluded from the data base for evaluation of DIT biases and uncertainties.

Slide 12

Comparisons between the MORSE Monte Carlo code, the DIT integral transport code, and the DOT discrete ordinance transport code, are shown for a typical C-E fuel assembly with poison shims. It is noted that, on the average, DIT gives a small
3 [ ] overestimate of the flux in pins adjacent to CEA waterholes.

Slide 13

A comparison of measured and calculated pin peaking for the Kritz lattice experiment is shown here. The measurement and calculations are normalized to one over the central 14x14 fuel pin array in the experiment. DIT shows an average
3 overprediction of [ ] for pins adjacent to CEA waterholes.

Slide 14

This new information will be used to update the pin peaking uncertainty evaluations in Chapter 4 of CENPD-153 to arrive at a new overall uncertainty for use of the incore instrument system in measuring core power distributions. These new results are based on an enlarged data base, including the C-E criticals from 1968 as well as the 1974 Kritz experiments. The basis for the evaluation of pin peaking uncertainty will also be modified to include only those pins adjacent to CEA waterholes. This directly responds to an NRC question on CENPD-153 and provides an improved basis because the peak pin in the core is expected to occur adjacent to a CEA waterhole.

Slide 15

The DIT evaluations of the critical experiments are shown here. The evaluation
3 of all of the experimental points provides a bias (overprediction) of [ ]
3 and a 95-95 confidence level uncertainty of [ ]; the uncertainty is less than that previously reported in CENPD-153. The variation of the bias for the

six sets of critical experiments can be correlated with changes in fuel to water volume ratios. The larger biases shown for C-E #53, C-E #56 and Kritz are closest to the volume ratios encountered in the C-E reactor lattice. However, an average of all the experiments has been conservatively chosen for application to the standard design model.

RRL:kf

PURPOSE IS TO DESCRIBE

- RESULTS OF RECENT INVESTIGATIONS OF POWER PEAKING IN ASSEMBLIES WITH CEA WATERHOLES.

- IMPLEMENTATION OF NEW MODEL.

- EVALUATION OF OPERATING PLANTS.

SLIDE 1

IMPLEMENTATION OF NEW MODEL

INCORPORATE NEW PEAKING MODEL IN OVERALL SETPOINT ANALYSIS CONCURRENT WITH REDUCED POWER DISTRIBUTION MEASUREMENT UNCERTAINTY (5.2% on F_q; 4.6% on F_r).

EVALUATION OF OPERATING PLANTS (CURRENT CYCLES)

- ACCOUNT FOR EFFECT ON OVERALL SETPOINT ANALYSIS WITH AVAILABLE MARGINS.

- NO SAFETY CONCERNS OR POWER CAPABILITY RESTRICTIONS - NO SIGNIFICANT TECHNICAL SPECIFICATION CHANGES REQUIRED.

- ACTION DEPENDS ON POWER DISTRIBUTION UNCERTAINTIES IN EFFECT ON CURRENT OR UPCOMING CYCLE.

FIXED INCORE SYSTEM

UNCERTAINTY

F_q CONSISTS OF THREE COMPONENTS

$$F_q = F_Z \cdot F_R \cdot F_P$$

F_Z - FUEL ASSEMBLY AXIAL PEAKING FACTOR

- UNCERTAINTY COMPONENT EVALUATED FROM SIGNAL REPRODUCIBILITY AND PRECISION OF AXIAL FITTING TECHNIQUE

F_R - AVERAGE FUEL ASSEMBLY INTEGRATED POWER - RADIAL

- UNCERTAINTY COMPONENT EVALUATED FROM COMPARISON OF CALCULATED AND MEASURED INSTRUMENT SIGNAL RADIAL DISTRIBUTIONS (AXIALLY INTEGRATED)

F_P - MAXIMUM PIN POWER TO FUEL ASSEMBLY AVERAGE POWER

- UNCERTAINTY COMPONENT EVALUATED FROM COMPARISON OF CALCULATED AND MEASURED FUEL PIN RELATIVE POWERS IN SEPARATE CRITICAL EXPERIMENTS

SLIDE 3

HISTORY OF POWER DISTRIBUTION UNCERTAINTY

1972 — EARLIEST PLANTS (PALISADES AND MAINE YANKEE) STARTED WITH 10% ON TOTAL PEAKING FACTOR.

LATE 1973 — INTERIM REPORT TO NRC JUSTIFYING 6.5% - NRC ASSIGNED 8% - HIGHEST FOR ALL PWR VENDORS.

AUG. 1974 — C-E STATED INTENT TO SUBMIT TOPICAL REPORT (CENPD-153P) ON BASIC UNCERTAINTY OF FIXED INSTRUMENT SYSTEM FOR MEASURING POWER PEAKING - JUSTIFIED 5.2% 95-95 UNCERTAINTY ON F_q.

OCT. 1974 — NRC RESPONDED BY REQUESTING TOPICAL REPORT ON INCORE INSTRUMENT ANALYSIS CODES OF ALL PWR VENDORS.

NOV. 1974 — C-E SUBMITTED CENPD-153P

APRIL 1975 — C-E SUBMITTED TOPICAL REPORT (CENPD-145) ON INCA - USING INCA F_q UNCERTAINTY OF 5.7% JUSTIFIED.

DEC. 1975 — CENPD-153
MARCH 1976 — CENPD-145
} NRC MADE EXTENSIVE REQUEST FOR ADDITIONAL INFORMATION

UP TO THIS TIME, OTHER PWR VENDORS EITHER HAVE NOT SUBMITTED REPORTS OR NRC DID NOT PLAN TO REVIEW.

JULY 1977 — C-E SUBMITTED RESPONSES TO NRC QUESTIONS ON CENPD-145.
C-E SUBMITTED UPDATE TO CENPD-153, SECTION 3.2.4 - DETECTOR DEPLETION.

DESIGN MODEL EVOLUTION

1968 -MEASUREMENTS OF PIN POWER PEAKING AND ROD WORTH PERFORMED FOR ASSEMBLIES WITH CEA WATERHOLES.

-PEAKING AND ROD WORTH AGREEMENT WITH DESIGN MODELS WAS GOOD.

-SUBSEQUENT EFFORT FOCUSED ON IMPROVING AGREEMENT ON REACTIVITY RUNDOWN AND TEMPERATURE COEFFICIENTS.(WATERHOLE TREATMENT HAS LARGE IMPACT).

1974 -ADDITIONAL CRITICAL/EXPERIMENTS PERFORMED FOR UO_2 AND MIXED OXIDE LATTICES .

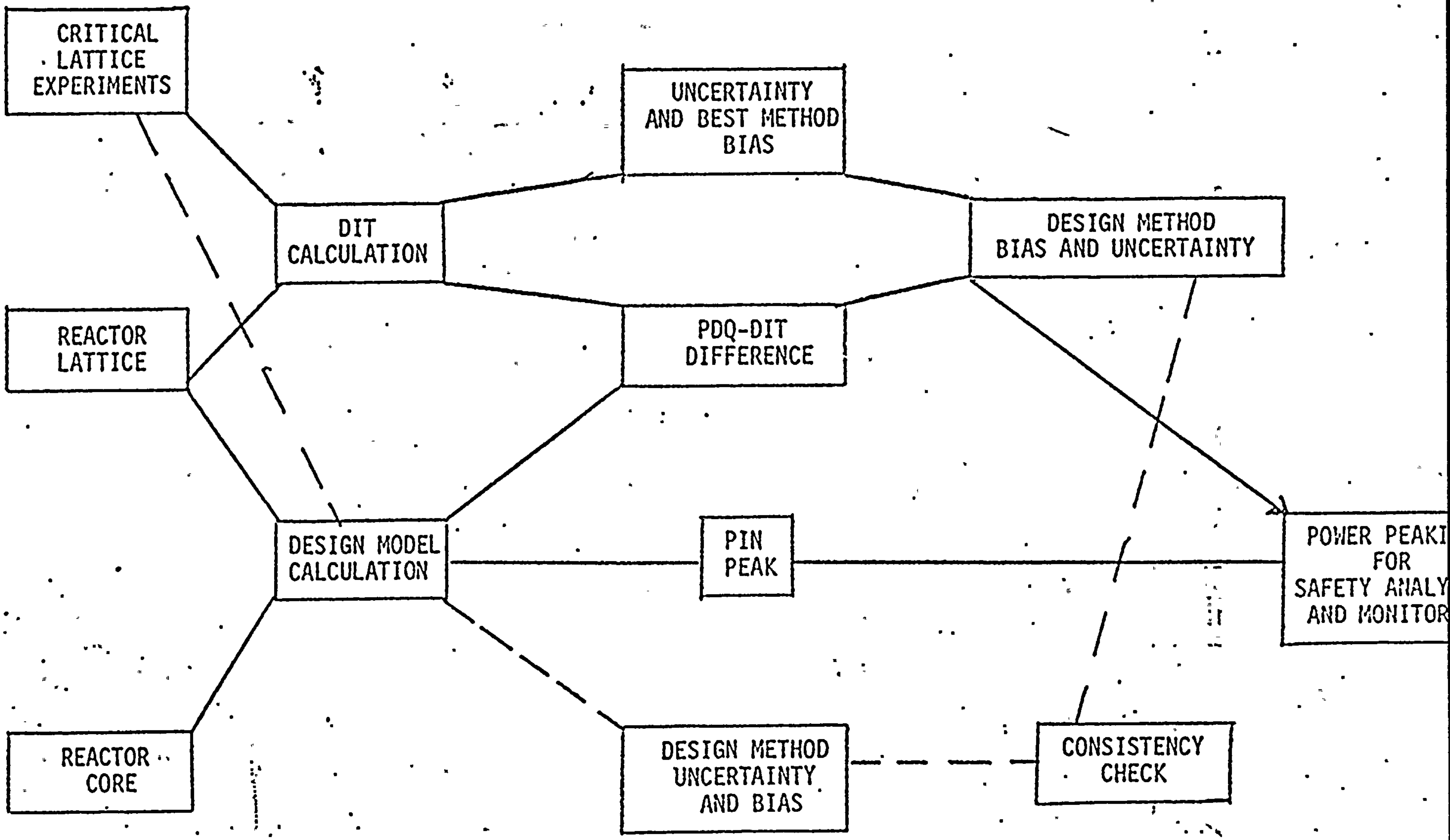
-ANALYSIS SUGGESTED SMALL (FIRST HOT EXPERIMENT) UNDERPREDICTION OF POWER IN PINS NEAR WATERHOLES - ONLY PART RECOVERED BY MODEL CHANGES (SEE CENPD-153) BUT WAS COVERED BY UNCERTAINTY.

1977 -FIRST DIT (MULTI-GROUP TRANSPORT THEORY) CALCULATIONS COMPLETED - PREDICTION OF WATERHOLE PEAKING QUANTIFIED.

-LEAD TO CURRENT ACTION TO ACCOUNT FOR EFFECT.

SLIDE 5

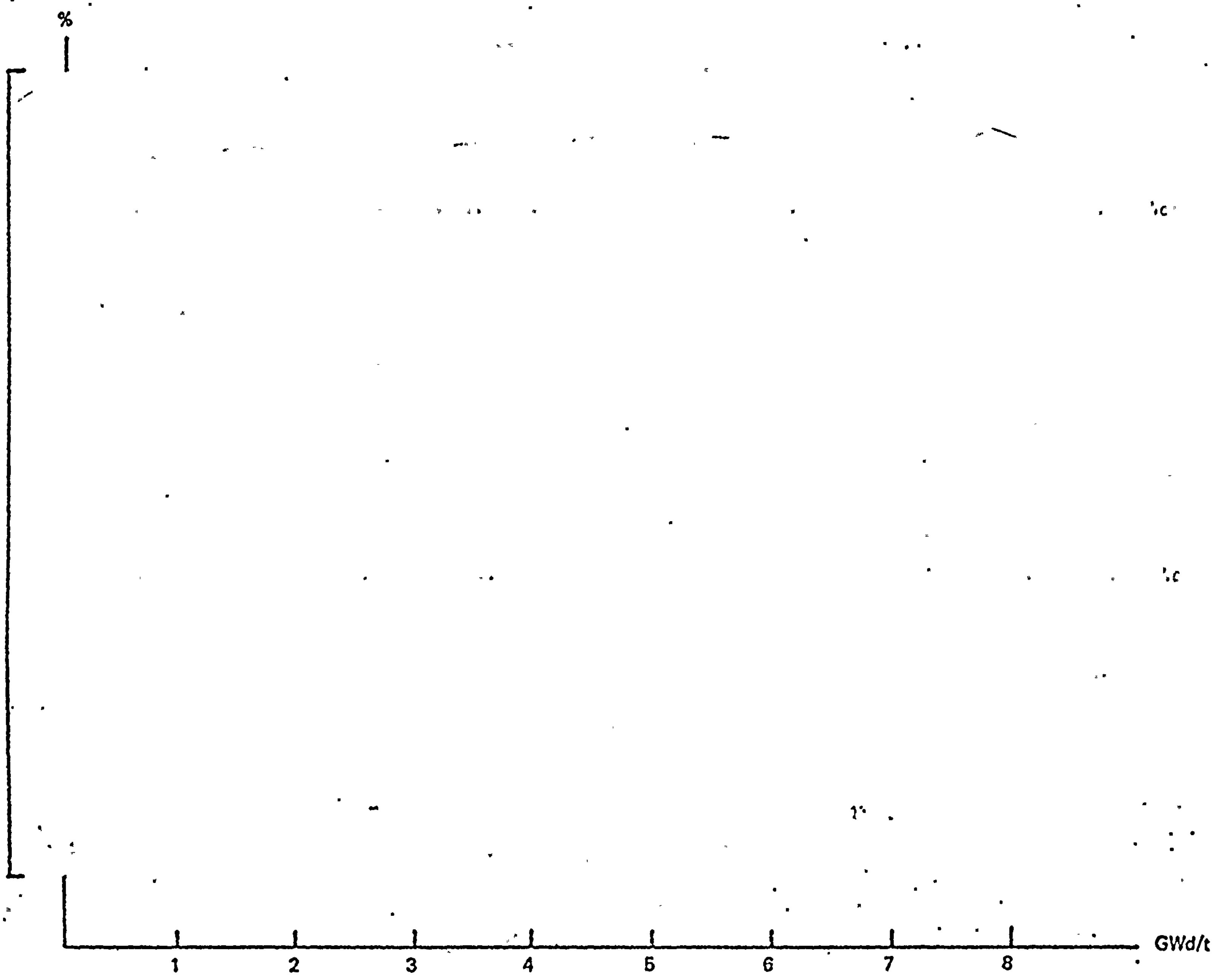
CALCULATION OF DESIGN POWER PEAKS



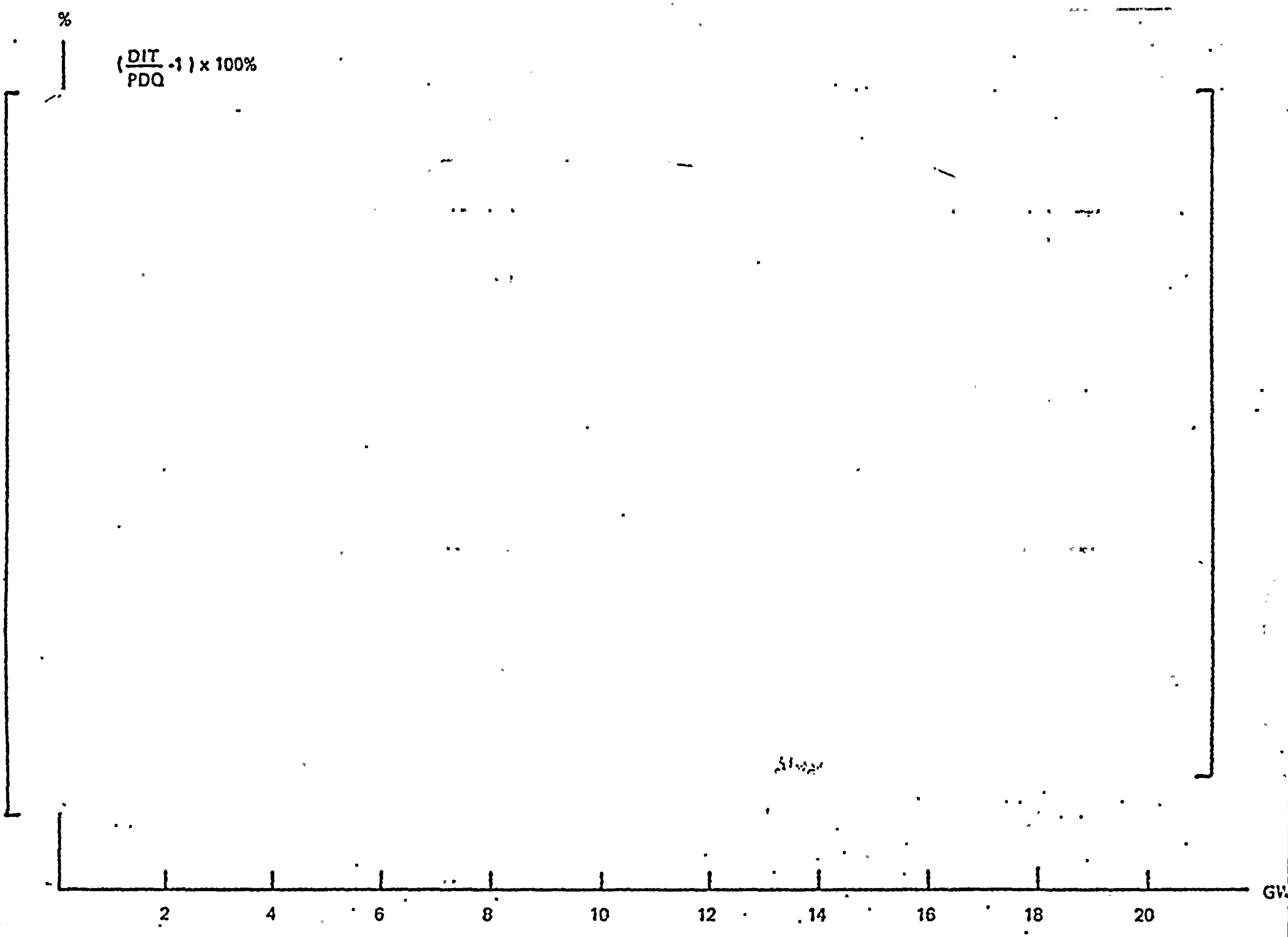
SLIDE 6

SLIDE 7

14 x 14 D-FUEL, F_p-BIAS



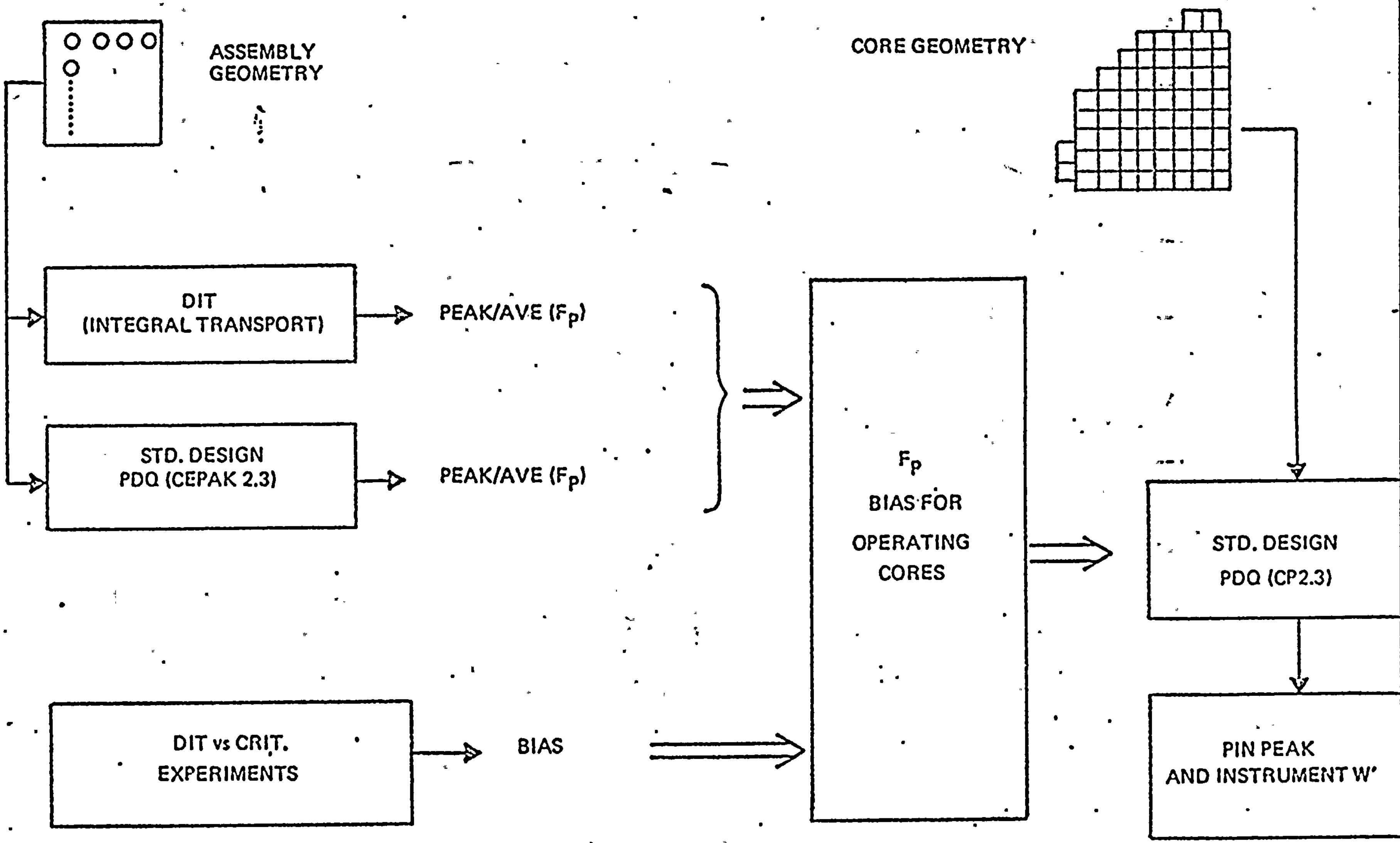
14 x 14 C_O FUEL, F_P-BIAS



SLIDE 8_3

SLIDE 9

PROCEDURE FOR THE CALCULATION OF PIN PEAKS



APPROXIMATIONS IN DIT

* PIN-CELLS COUPLED BY INTERFACE CURRENTS CONSTANT OVER EACH FACE
* DOUBLE P_0 AT EACH INTERFACE (SEPARATE VERSION WITH HIGHER ORDER TERMS IN EXISTENCE)
* COLLISION PROBABILITIES WITHIN EACH CELL CALCULATED FOR CYLINDRICIZED GEOMETRY.
  (OPTIONAL: SQUARE BOUNDARIES IN CALCULATION OF CP'S

SLIDE 10

CRITICAL EXPERIMENTS WITH CEA WATERHOLES

EXPERIMENT	ENR. %	V_m/V_f	T_m	PPM	SIZE OF CORE	No. OF PINS MEAS. AT W.H.	No. OF WATERHOLES
14 x 14	3 w/o	1.63	–	–	–	–	–
16 x 16	3 w/o	1.71	–	–	–	–	–

3

SLIDE 11

SLIDE 12

3

CALVERT CLIFFS I

INFINITE LATTICE OF C12 LO ASSEMBLY

THERMAL FLUX x 10^{-2}

DEVIATION (%) FROM MORSE

[ ]

1 MORSE, HET.
2 NUTEST - DIT, K=1, HET.
3 DOT, S_8, 28 x 28, HOM.

[ ]

KRITZ - UWH2

[ ]

(D/M -1) x 100%

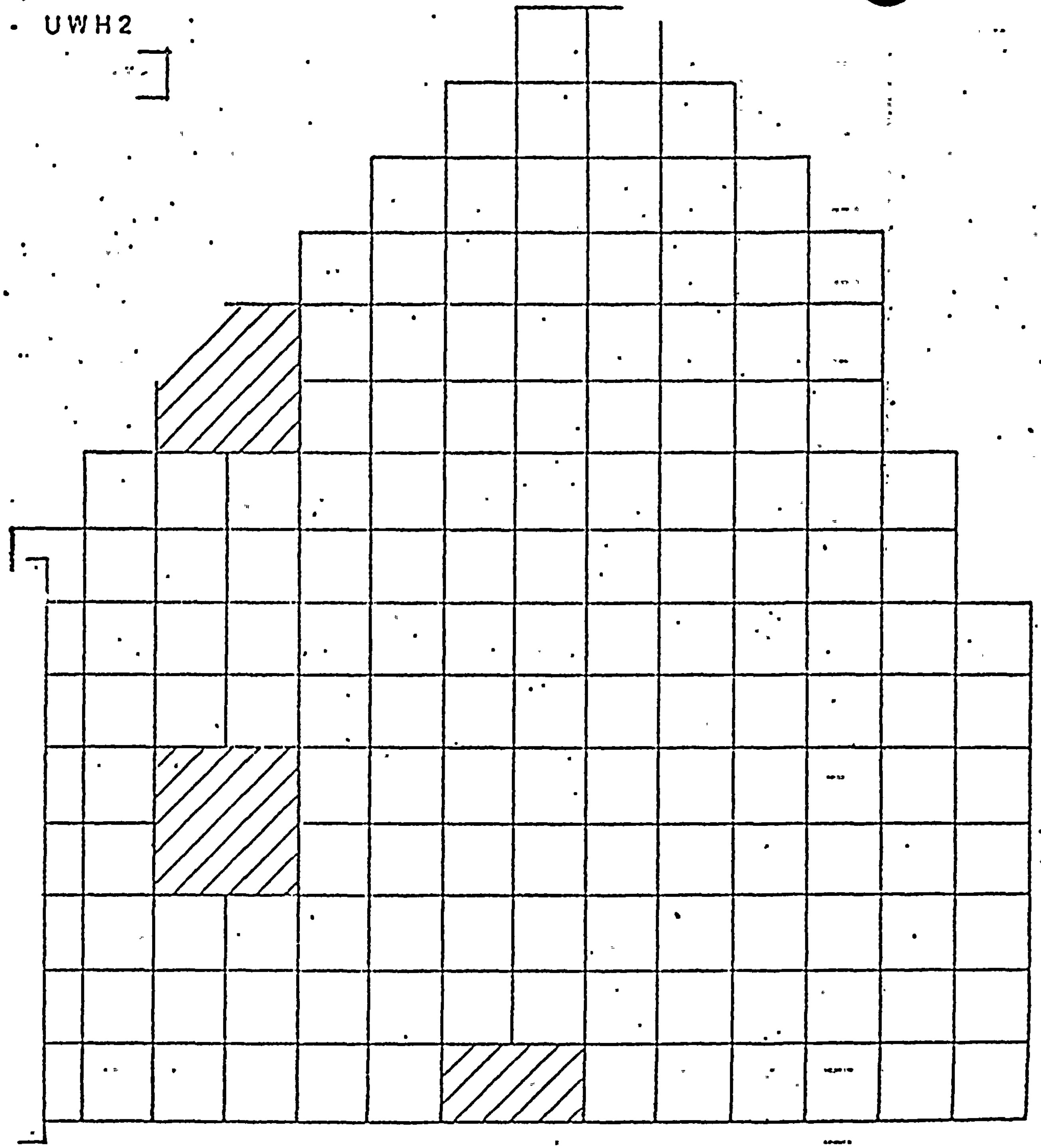
D = DIT

M = MEAS/GLOBAL

$\overline{X}_{WU}$ = [ ]

SLIDE 13

3



DIFFERENCES FROM CHAPTER 4

OF CENPD-153.

* ENLARGED DATA BASE INCLUDING C-E CRITICALS

* PEAK PINS SPECIFICALLY ADDRESSED.

  CENPD-153 GAVE:

3 $\bar{X}$ =[ ](CALC. BIAS)

3 $k\bar{\sigma}$ =[ ](CALC. UNCERTAINTY)

SLIDE 14

DIT vs. Exp.

OBSERVED BIAS AND STANDARD DEVIATION FOR PINS FACING CEA WATERHOLES

LATTICE	NO. OF POINTS	TOTAL MEAS.	BIAS %	STD. DEV. %
3 [				]

3 $\bar{X} \pm k\sigma = [\quad]$

SLIDE 15

SECTION 2

Actions to Accommodate Increased Waterhole Peaking for St. Lucie Unit 1

The purpose of the information presented in this section is to demonstrate that the analysis package used for the first cycle of the subject plant to date has employed methodology with sufficient conservatisms to accommodate the increased water hole peaking discussed elsewhere in Section 1. The only Technical Specification changes required are a tightening by 0.5% of the surveillance requirements on in-core or ex-core surveillance of peak linear heat rate (kw/ft) relative to the LCO. Since the subject plant has been operated with at least 0.5% margin to the operating limits, there have been no violations of these limits (Slides 1 and 2).

This summary will take the form of a commentary on the remainder of the slides presented to the NRC Staff on December 16, 1977. Slide 3 displays the sources of margin which offset the increased water hole peaking for Calvert Cliffs Unit 1 Cycle 2, as an example. The top half of the slide deals with DNBR-related limits - the LSSS and the LCO. The lower half of the slide deals with kw/ft-related limits - the LSSS and the LCO.

Let us consider first the DNBR LSSS. The uncertainty allowance employed in the analysis whose resultant setpoints are contained in the submittal for Cycle 2 of the subject unit was 5%. However, the total allowance which must be made is 9.8%, of which 4.6% is the increase in water hole peaking, and 5.2% is the INCA measurement

uncertainty on Fr. It will be remembered that INCA is used to surveille Fr relative to its limit. The technical specification limit on Fr is an assumption in the setpoint analysis. INCA uncertainty must also be accounted for in the setpoint analysis, since it does not appear explicitly in the surveillance requirement. The next column of Slide 3 shows that an additional penalty of 4.8% exists (9.8% minus 5%).

The margin credits which offset the additional penalty are displayed in the column headed "REMEDIES". There is a 3% credit for improved thermal margin/low pressure trip synthesis. This was summarized on Slides 4-7 of the December 16, 1977 presentation, but will not be discussed further here since a detailed presentation of this topic appears in Section 3.

A 2% credit arises from the statistical treatment of selected uncertainties. This topic is treated in detail under the heading for Slides 7 and 8. Since the two listed credits offset the 4.8% penalty there is no requirement for changes to the technical specifications or the setpoints in connection with the DNBR LSSS.

The next line of Slide 3 deals with the DNBR LCO. The same additional penalty of 4.8% margin must be accommodated as for the DNBR LSSS. 3% credit is taken for margin gains computed by TORC in conjunction with the CE-1 correlation relative to the design basis COSMO/INTHERMIC analyses employed for these operating reactors. The basis for the 3% credit was summarized on Slides 9 through 11 in the December 16, 1977 meeting, and will not be dealt with

further here since Section 4 contains a detailed discussion of this material. The remainder of the 4.8% additional penalty on the DNBR LCO is offset by the credit for the statistical uncertainty treatment covered under the heading for Slides 7 and 8.

We now invite attention to the line of Slide 3 headed "KW/FT LSSS". The setpoint analysis for the kw/ft LSSS contains a 10% uncertainty allowance at present. This must be revised to 10.4% to take account of the 4.6% increase in water hole peaking and the 5.8% INCA allowance on Fq measurement. There is, thus, a 0.4% additional penalty to be accommodated for the kw/ft LSSS (10.4% minus 10%). In the "REMEDIES" column we see that this penalty is offset by a credit from the statistical uncertainty treatment (Slide 8). Therefore, no change is required to the technical specifications or to the setpoints..

The kw/ft operating limit monitoring process presently includes an 8% allowance for uncertainties. However, a 10.4% allowance must be accommodated, leaving an additional penalty of 2.4%. Of this, 1.9% is offset by the statistical uncertainty treatment (Slide 8), and the balance by a tightening of the relevant portions of the technical specifications on kw/ft surveillance by 0.5%.

For reference, some data is included at the bottom of the slide on variations of Fr and Fxy measured at the time in life when these quantities were their highest during steady state operation and the technical specification limits on these quantities.

Slide 7 displays the source of the 2% credit on the Fr uncertainties arising from the use of statistics to combine selected individual uncertainties. In the rod bow topical (CENPD-225-P) was presented a computation of Fr uncertainty by statistical means in comparison with the multiplicative combination of the individual uncertainties. The terms which enter into the uncertainties on Fr include the engineering factor on Fr, the nuclear uncertainty on Fr, and the fuel rod bowing factor. (I) of Slide 7 reproduces the computation precisely as presented in the topical report for reference, with the conclusion that the employment of a statistical combination of the uncertainty components discussed in that report yielded an overall uncertainty 2.35% lower than just the multiplicative combination of the engineering and nuclear factors.

The setpoint analyses, which form the basescof the submittals for the initial cycles and reloads for the plants in question, included a multiplicative combination of the nuclear and engineering components of the overall uncertainty. We may take credit for the statistical combination of the relevant uncertainties only after using the appropriate numerical values for the components, and showing that these components meet the normal criteria of randomness and independence. The arguments for randomness and independence for these components are presented in detail in CENPD-225-P and will not be repeated here.

II of Slide 7 shows the computation of the uncertainty credit using the appropriate value of 5.2% for the Fr uncertainty with a 2% result, including the contribution of the fuel rod bow factor.

Slide 8 is the analog of Slide 7, but for the Fq uncertainty. By a similar process we compute a 1.9% credit for the use of a statistical error combination including the fuel and poison rod bowing factors, relative to the multiplicative combination of just the engineering and nuclear factors. It should be noted that the random component of the INCA uncertainty on Fq is 5.7%, and that $\overline{X}$ is 0.1% for a total INCA uncertainty on Fq of 5.8% as seen on Slide 3.

Slide 12 summarizes the situation for Cycle 1 of St. Lucie-1, which is expected to be completed about April 1, 1978. Uncertainty allowances of 10% and 8% were included in the setpoint analyses for the DNBR LSSS and LCO, respectively, with the result that the only additional penalty required was 1.8% on the DNBR LCO. This is entirely offset by the credit from the statistical uncertainty treatment. The kw/ft situation is precisely the same as for Calvert Cliffs-1 Cycle 2 discussed above.

Slide 13 summarizes the Technical Specification changes required for implementing the 0.5% tightening of the surveillance on peak linear heat rate.

In conclusion (Slide 14), existing margins and conservatisms in the setpoint analyses for submittals now in the hands of the NRC Staff compensate for the increased water hole peaking, with the exception of a 0.5% tightening of Technical Specification surveillance requirements for peak linear heat rate.

STATUS OF OPERATING C-E PLANTS* AND PROPOSED ACTION

— TOPICS —

- PLANT-AND-CYCLE-SPECIFIC EVALUATION
- MARGIN CREDITS
  - — IMPROVED TM/LP TRIP SYNTHESIS
  - — TORC/CE-1 CREDIT FOR DNB/LCO
  - — STATISTICAL UNCERTAINTY TREATMENT
- SUMMARY

SUMMARY

1. NO CHANGE IS REQUIRED TO F_R, F_{XY} TECH. SPEC LIMITS, BECAUSE:

2. THE INCREASED PEAKING IS ACCOMODATED BY EXISTING MARGINS AND CONSERVATISMS.

3. TECH. SPEC. SURVEILLANCE REQUIREMENTS MUST BE TIGHTENED BY 0.5% FOR IN-CORE OR EX-CORE KW/FT LCO.

4. SINCE THERE HAS BEEN >>0.5% MARGIN BETWEEN ACTUAL OPERATION AND THE OPERATING LIMITS ON KW/FT, THERE HAVE BEEN NO VIOLATIONS OF THESE LIMITS.

CALVERT CLIFFS - UNIT 1, CYCLE 2
(NOW OPERATING; SHUTS DOWN 1/78)

EFFECTS ON LSSS & LCO:

REQUIRED ACTION:

LIMIT	PRESENT F_R UNCERT. ALLOW.	REVISED F_R UNCERT. ALLOW	ADDITIONAL PENALTY	REMEDIES — TYPE	REMEDIES — AMT.	CHANGES TO TECH. SPECS.
BR LSSS	5%	4.6 + 5.2 = 9.8	4.8%	1. IMPROVED TM/LP TRIP SYNTHESIS 2. CREDIT FROM STATISTICAL UNCERTAINTY TREATMENT	3% 2%	— NONE —
BR LCO	5%	4.6 + 5.2 = 9.8	4.8%	1. CREDIT FROM TORC/CE-1 2. CREDIT FROM STATISTICAL UNCERTAINTY TREATMENT	3% 2%	— NONE —

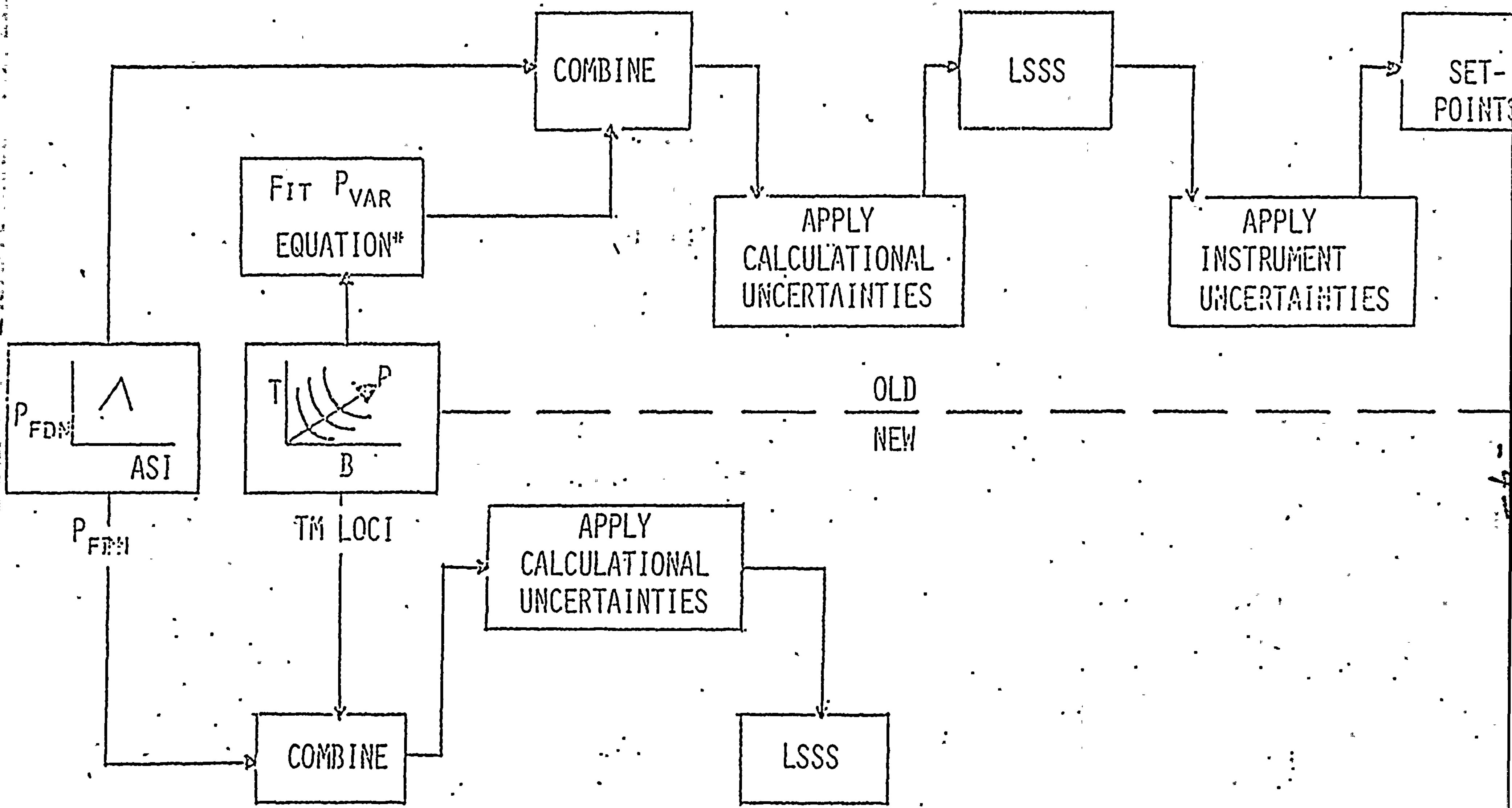
LIMIT	PRESENT F_R/F_{XY} UNCERT. ALLOW.	REVISED F_R/F_{XY} UNCERT. ALLOW.	ADDITIONAL PENALTY	REMEDIES — TYPE	REMEDIES — AMT.	CHANGES TO KW/FT TECH. SPECS
/FT LSSS	10%	4.6 + 5.8 = 10.4	0.4%	1. CREDIT FROM STATISTICAL UNCERTAINTY TREATMENT	0.4%	-NONE-
/FT LCO	8%	4.6 + 5.8 = 10.4	2.4%	1. IN-CORE MONITORING — REDUCE ALARM LIMITS 2. EX-CORE MONITORING — CHANGE SCALING EQUATION 3. CREDIT FROM STATISTICAL UNCERTAINTY	0.5% 1.9%	— YES —

CURRENT MARGINS ON:

QUANTITY	MEASURED (3,970 MWD/T)	TECH. SPEC. LIMIT	MARGIN
TEGRATED RADIAL (F_R)	1.35	1.43	5.9%
NAR RADIAL	1.41	1.50	6.4%

IMPROVED TM/LP TRIP SYNTHESIS

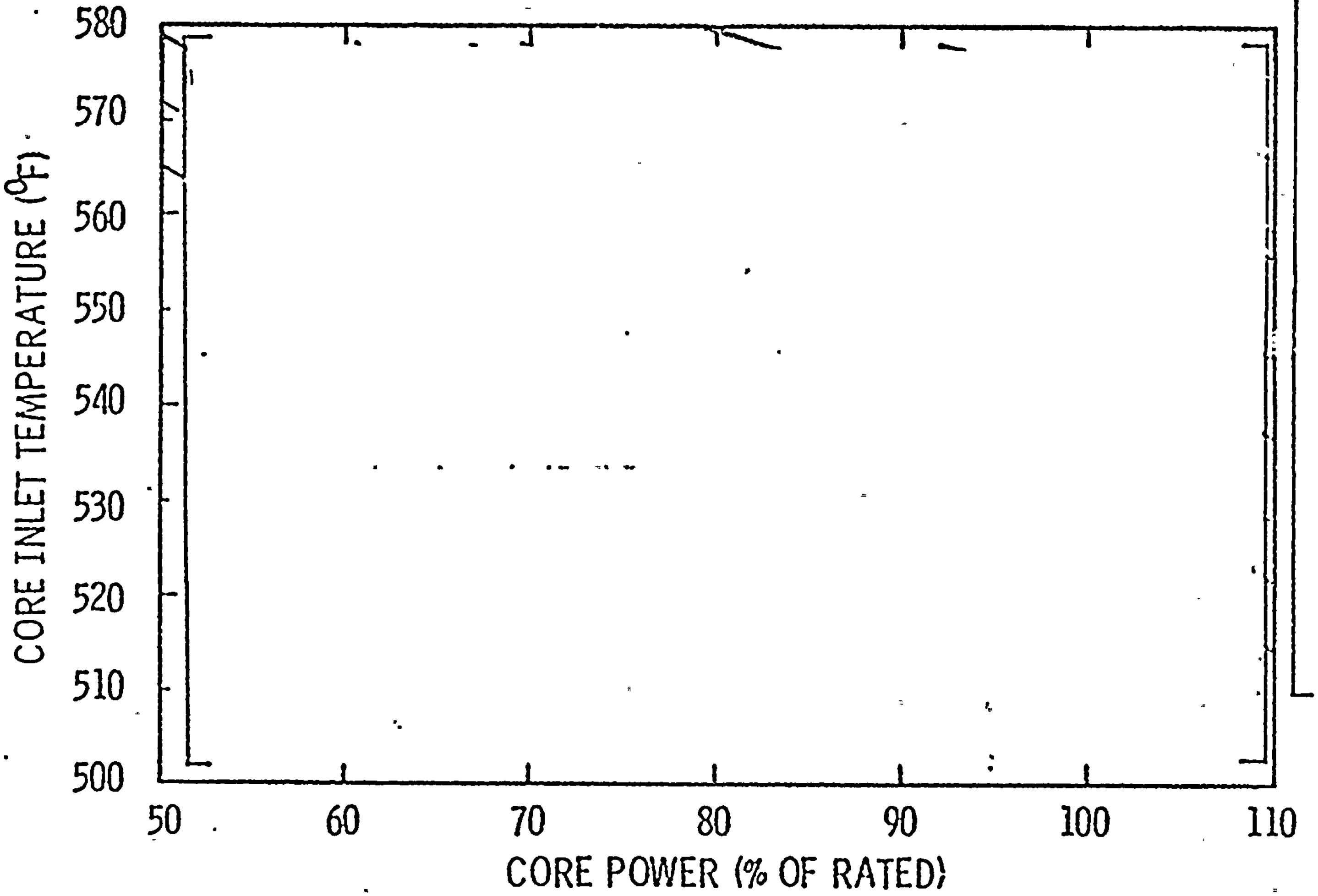
(SEE ALSO CENPD-199-P)



$$*P_{VAR} = \alpha\,(\text{POWER}) \times \text{POWER} + \beta \times T_{INLET} + \gamma$$

3

3



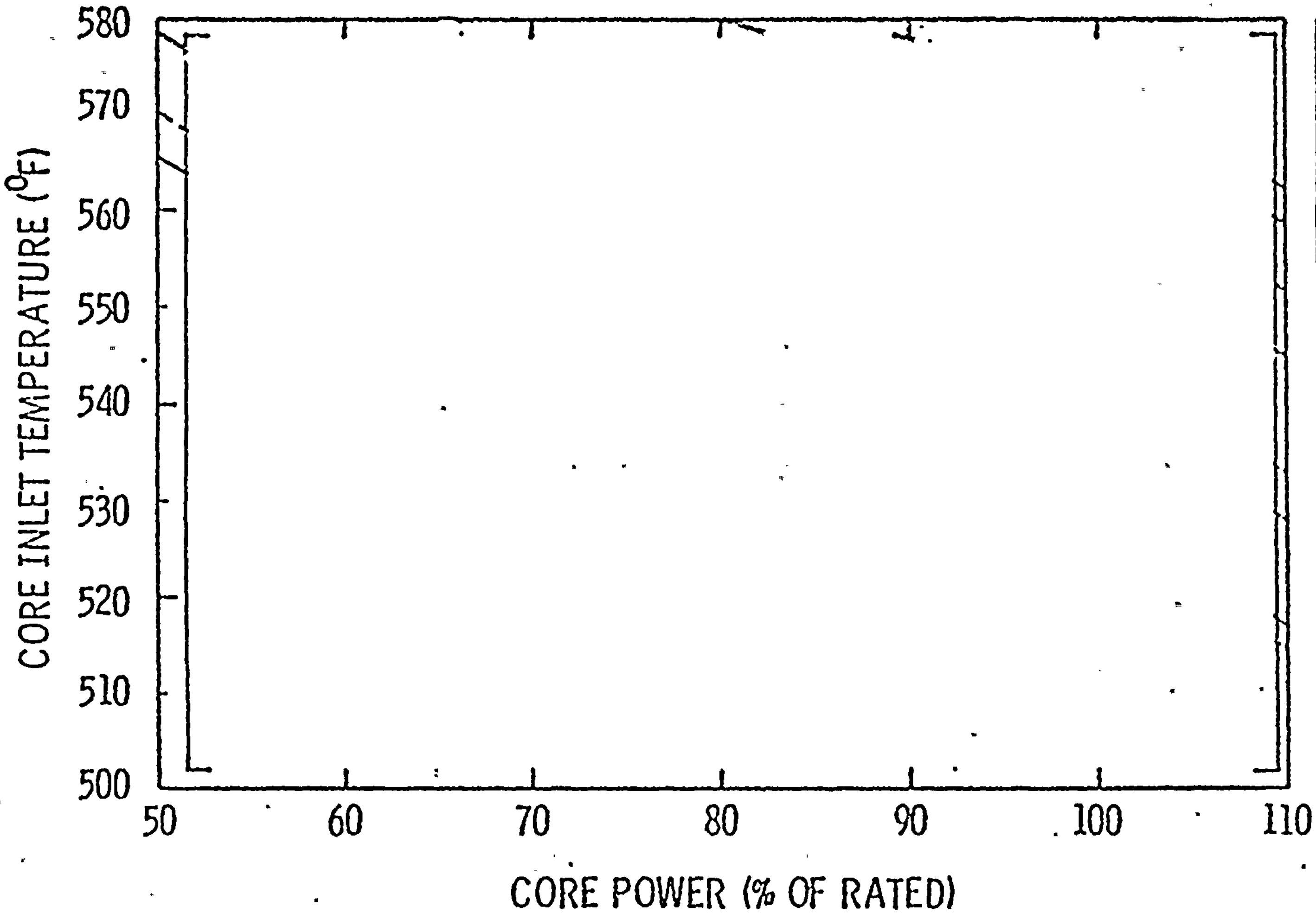
CORE INLET TEMPERATURE (°F)
580
570
560
550
540
530
520
510
500
50
60
70
80
90
100
110
CORE POWER (% OF RATED)

3

3

3

3



580
570
560
550
540
530
520
510
500
CORE INLET TEMPERATURE (°F)
50
60
70
80
90
100
110
CORE POWER (% OF RATED)

3

F_R UNCERTAINTY

I. COMPUTATION OF F_R^U IN ROD BOW TOPICAL REPORT (CENPD 225P)

A. RSS COMBINATION OF F_R^E, F_R^N, F_R^F

$$F_R^U = 1 + \sqrt{(F_R^E - 1)^2 + (F_R^N - 1)^2 + (F_R^F - 1)^2}$$

$$= \underline{1.0695}$$

FOR: F_R^E = ENGINEERING FACTOR = 1.03

F_R^N = NUCLEAR FACTOR = 1.06

F_R^F = FUEL ROD BOWING FACTOR = 1.0180

B. MULTIPLICATIVE COMBINATION OF F_R^E AND F_R^N

$$F_R^U = (F_R^E)\ (F_R^N) = \underline{1.0913}$$

II. COMPUTATION OF F_R^U FOR $F_R^N = 1.052$

A. RSS COMBINATION OF F_R^E, F_R^N, F_R^F :

$$F_R^U = \underline{1.0627}$$

B. MULTIPLICATIVE COMBINATION OF F_R^E AND F_R^N

$$F_R^U = \underline{1.0836}$$

C. CREDIT FOR STATISTICAL UNCERTAINTY TREATMENT:

$$1.0836 - 1.0627 = 0.0209 \cong \boxed{2\%}$$

F_Q UNCERTAINTY

I. COMPUTATION OF F_Q^U IN ROD BOW TOPICAL REPORT (CENPD-225P)

A. RSS COMBINATION OF F_Q^E, F_Q^N, F_Q^F, F_Q^P

$$F_Q^U = 1 + \sqrt{(F_Q^E-1)^2 + (F_Q^N-1)^2 + (F_Q^F-1)^2 + (F_Q^P-1)^2}$$

$$= \underline{1.0898}$$

FOR: F_Q^E = ENGINEERING FACTOR = = 1.03

F_Q^N = NUCLEAR FACTOR = 1.08

F_Q^F = FUEL ROD BOWING FACTOR = 1.018

F_Q^P = POSION ROD BOWING FACTOR = 1.021

B. MULTIPLICATIVE COMBINATION OF F_Q^E, F_Q^N

$$F_Q^U = (F_Q^E)\ (F_Q^N) = \underline{1.1124}$$

II. COMPUTATION OF F_Q^U FOR F_Q^N = 1.057

A. RSS COMBINATION OF F_Q^E, F_Q^N, F_Q^F, F_Q^P

$$F_Q^U = \underline{1.0701}$$

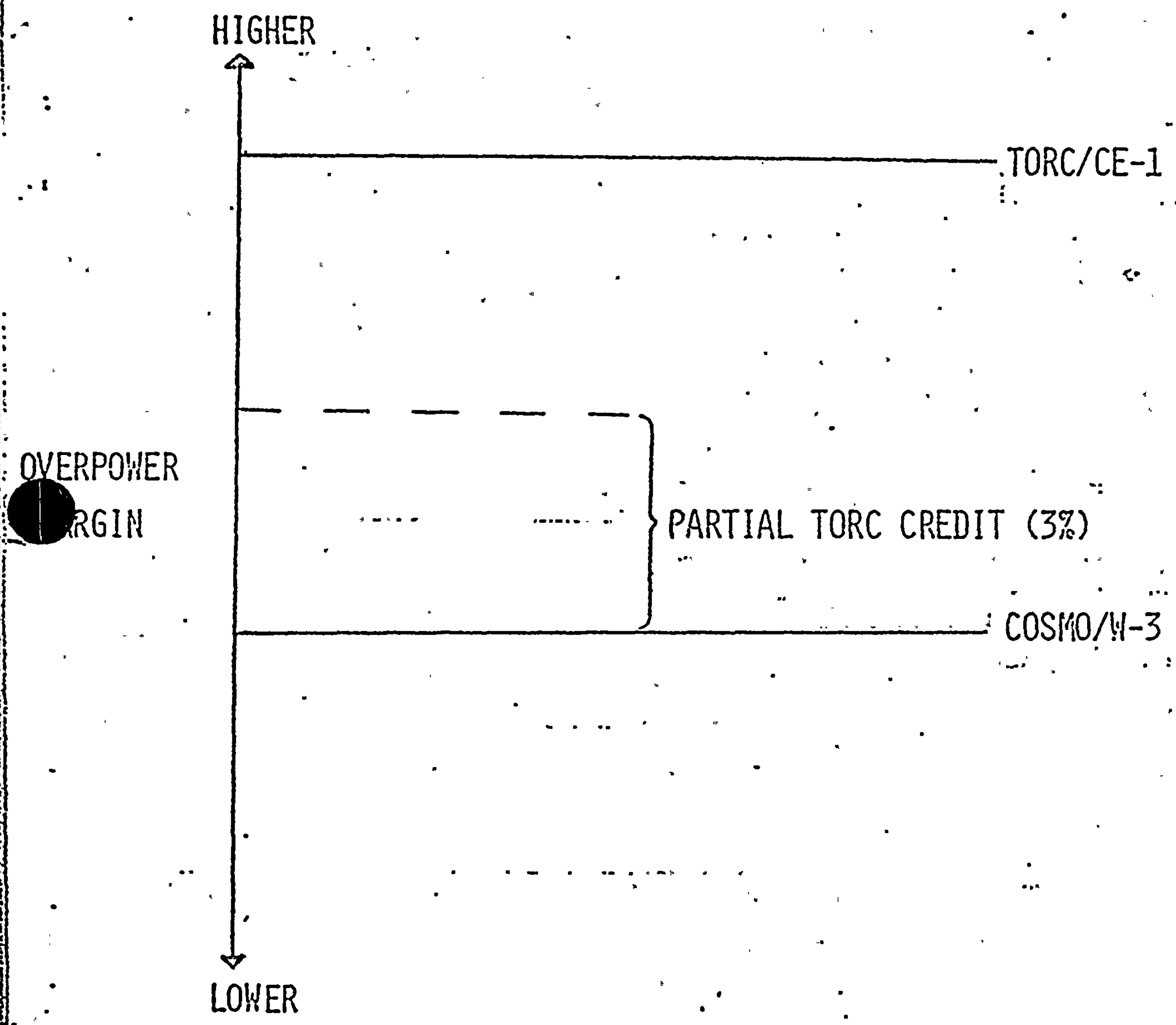
B. MULTIPLICATIVE COMBINATION OF F_Q^E AND F_Q^N

$$F_Q^U = \underline{1.0887}$$

C. CREDIT FOR STATISTICAL UNCERTAINTY TREATMENT

1.0887 - 1.0701 = 0.0186 ≅ **1.9%**

OVERPOWER MARGIN GAIN FOR DNBR LCO



TORC/CE-1 GAIN

Number of Points in Parameter Space	691
Average Gain, % Overpower	7
Operating Point Gain, % Overpower	12

USE OF TORC/CE-1 CREDIT

CONCLUSION

- PARTIAL CREDIT FOR TORC/CE-1 WILL BE TAKEN IN LCO (3%)
- LOFA IS LIMITING AOO FOR BOTH COSMO/W-3 AND TORC/CE-1
- REQUIRED MARGIN DOES NOT CHANGE
- PROTECTION SYSTEM CONSTANTS DO NOT CHANGE
- INITIAL CONDITIONS ASSUMED IN SAFETY ANALYSIS ARE NOT EXCEEDED IN PLANT OPERATION.

ST. LUCIE - UNIT 1, CYCLE 1

(NOW OPERATING, SHUT DOWN 3/78)

REQUIRED ACTI

EFFECTS ON LSSS & LCO:

MIT	PRESENT F_R UNCERT. ALLOW.	REVISED F_R UNCERT. ALLOW	ADDITIONAL PENALTY	— REMEDIES — TYPE	AMT.	CHANGES TO TECH. SPECS.
R LSSS	10%	4.6 + 5.2 = 9.8	-0-	NOT REQUIRED	-	– NONE –
R LCO	8%	4.6 + 5.2 = 9.8	1.8	1. CREDIT FROM STATISTICAL UNCERTAINTY TREATMENT	2%	– NONE –

IMIT	PRESENT F_R/F_{XY} UNCERT. ALLOW.	REVISED F_R/F_{XY} UNCERT. ALLOW.	ADDITIONAL PENALTY	— REMEDIES — TYPE	AMT.	CHANGES TO KW/FT TECH. SPECS
FT LSSS	10%	4.6 + 5.8 = 10.4	0.4%	1. CREDIT FROM STATISTICAL UNCERTAINTY TREATMENT	0.4%	-NONE-
FT LCO	8%	4.6 + 5.8 = 10.4	2.4%	1. IN-CORE MONITORING – REDUCE ALARM LIMITS 2. EX-CORE MONITORING – CHANGE SCALING EQUATION 3. CREDIT FROM STATISTICAL UNCERTAINTY	0.5% 1.9%	– YES –

CURRENT MARGINS ON:

QUANTITY	MEASURED (7388 MWD/T)	TECH. SPEC. LIMIT	MARGIN
EGRATED RADIAL (F_R)	1.32	1.36	3.0%
[illegible]	1.34	1.36	1.5%

TECH SPEC CHANGES

IN-CORE MONITORING

ADD PENALTY FACTOR OF 1.005 TO EXISTING PENALTIES USED TO SET IN-CORE ALARM LIMITS

EX-CORE MONITORING

MODIFY EX-CORE SCALING EQUATION TO CONTAIN A FACTOR OF $\frac{1}{1.005}$

TECH SPEC SECTION AFFECTED

SURVEILLANCE REQUIREMENTS PORTION OF LINEAR HEAT RATE TECH SPEC:
ST. LUCIE-1 - SECTION 3/4.2.1

SUMMARY

1. NO CHARGE IS REQUIRED TO F_R, F_{XY} TECH. SPEC LIMITS, BECAUSE:

2. THE INCREASED PEAKING IS ACCOMODATED BY EXISTING MARGINS AND CONSERVATISMS.

3. TECH. SPEC. SURVEILLANCE REQUIREMENTS MUST BE TIGHTENED BY 0.5% FOR IN-CORE OR EX-CORE KW/FT LCO.

4. SINCE THERE HAS BEEN >>0.5% MARGIN BETWEEN ACTUAL OPERATION AND THE OPERATING LIMITS ON KW/FT, THERE HAVE BEEN NO VIOLATIONS OF THESE LIMITS.

SECTION 3

IMPROVED THERMAL MARGIN/LOW PRESSURE TRIP SYNTHESIS

The purpose of this section is to describe the improved Thermal Margin/Low Pressure (TM/LP) trip synthesis methodology for which a 3% credit is claimed as discussed in Section 2. The generation of data, as described in CENPD-199-P "Topical Report on Setpoint Methodology," remains unchanged. The change, as discussed in this section, is to the curve fitting procedure for the thermal margin limit lines. The generation of the thermal margin limit lines and the P_{fdn} vs. I_p curves remains as described in CENPD-199-P.

Slide 1, which is Figure 5-18 from CENPD-199-P, is a summary of the previously used TM/LP LSSS synthesis methodology. A brief discussion of this slide is included here so that a base is provided from which the changes, resulting the improved synthesis methodology, can be understood.

3, 5

3, 5

The combined results of these two procedures is an equation of the form shown in Block F which is the TM/LP LSSS defined in terms of pressure, axial shape index, power and inlet temperature.

Slide 1 is then a summary of how the thermal margin limit lines and P_{fdn} vs. I_p curves were previously combined to synthesize the TM/LP LSSS.

Slide 2 is Slide 1 modified to show a summary of the improved synthesis methodology.

3, 5

3,

3, 5

Slide 13 illustrates the magnitude of the improvement resulting from use of the improved TM/LP trip synthesis methodology. This slide shows the results of TM/LP LSSS calculations for a 2700 Mwt NSSS using both the original and the improved synthesis methodology. The TM/LP LSSS from this slide is repeated on Slide 14 for comparison with TM/LP LSSS generated using the improved synthesis methodology but assuming a radial peak 3% higher than that used for Slide 13. Slide 14 shows that, even with a 3% higher radial peak, the improved methodology results in an improvement over the original synthesis methodology.

Figure 5-18 (CENTD-???)

TM/LP DNB LSSS SYNTHESIS

SLIDE 1

3,5

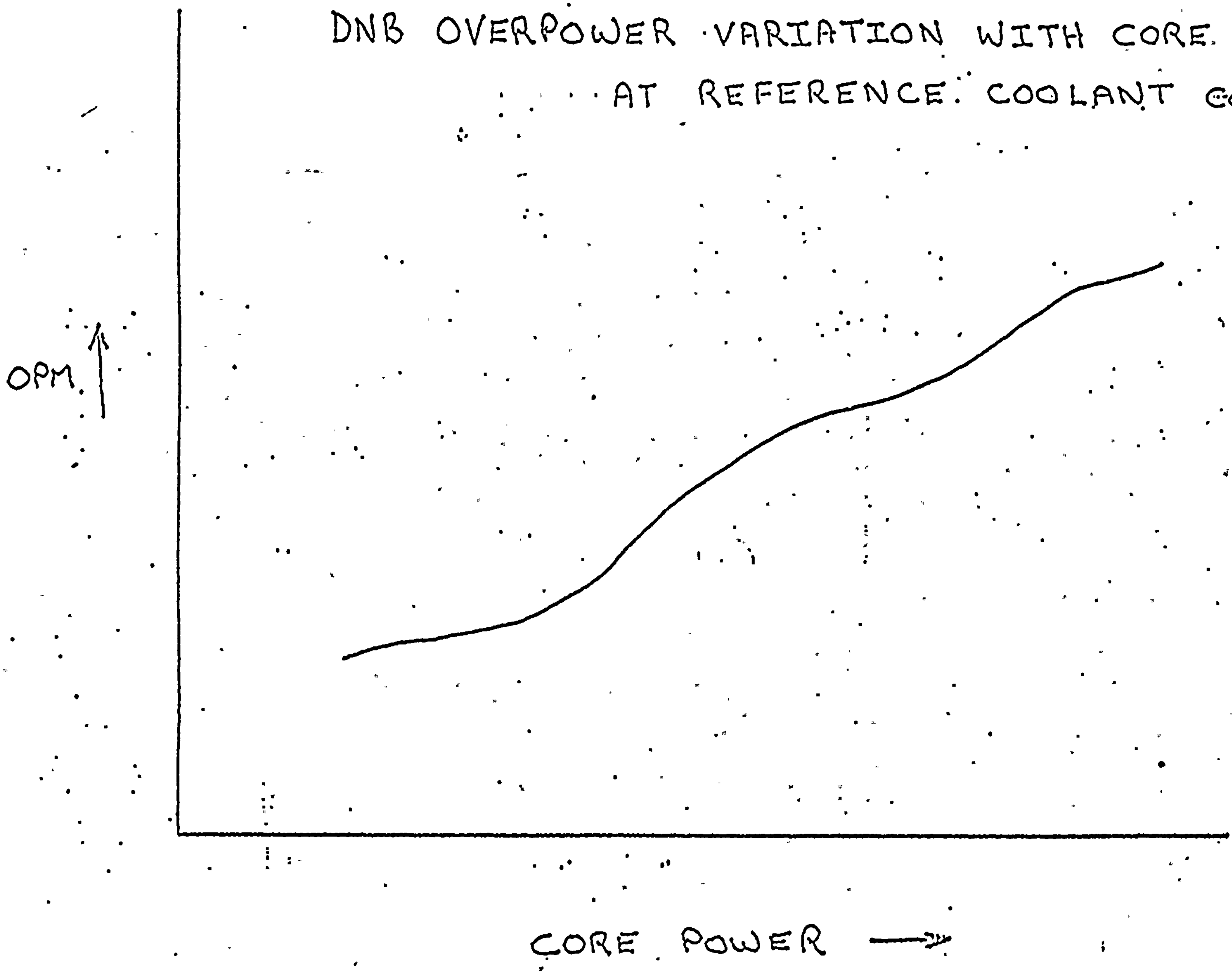
TM/LP DNB LSSS SYNTHESIS

SLIDE 2

3.5

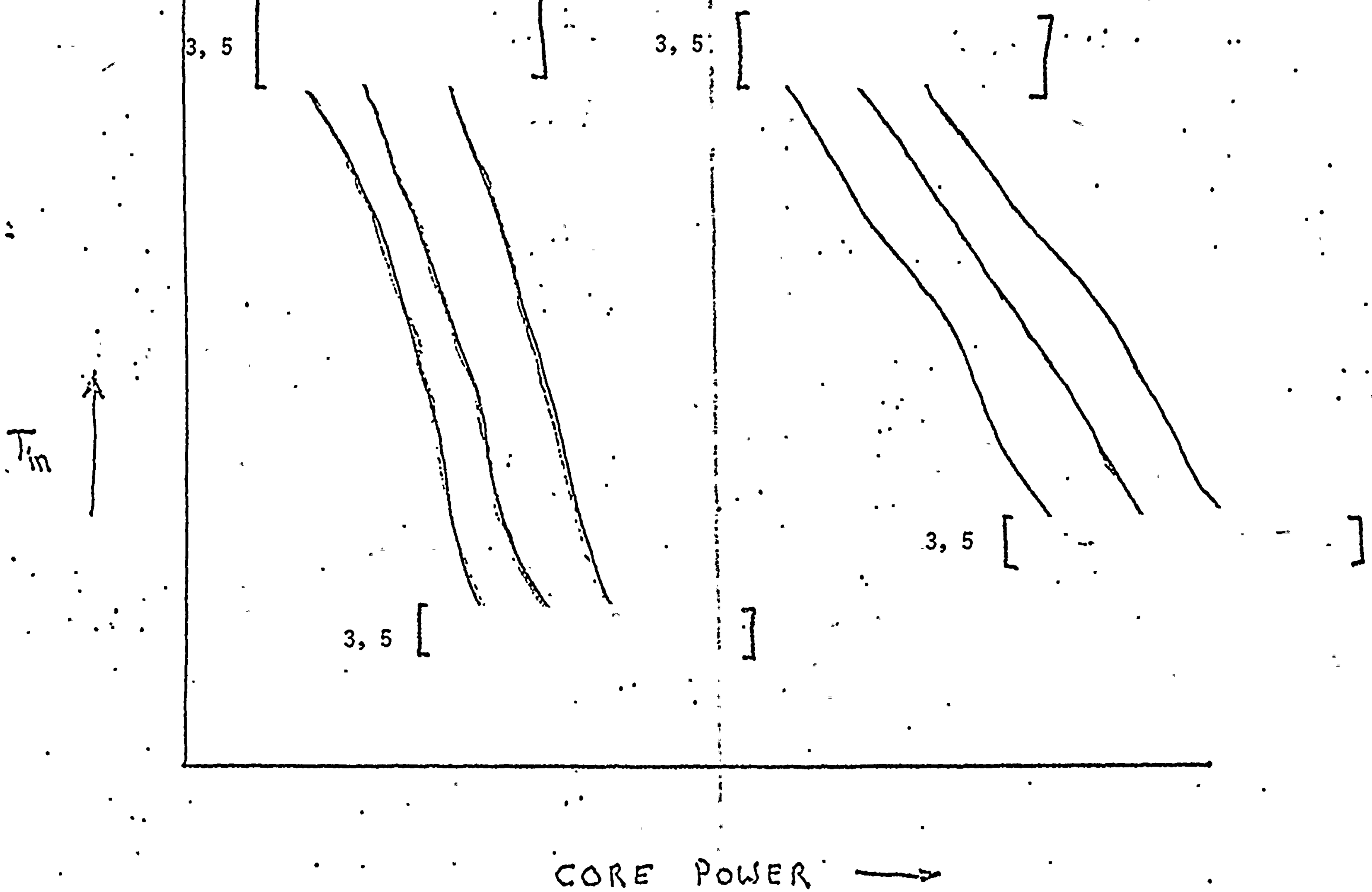
SLIDE 3

DNB OVERPOWER VARIATION WITH CORE POWER AT REFERENCE COOLANT CONDITIONS



SLIDE 4

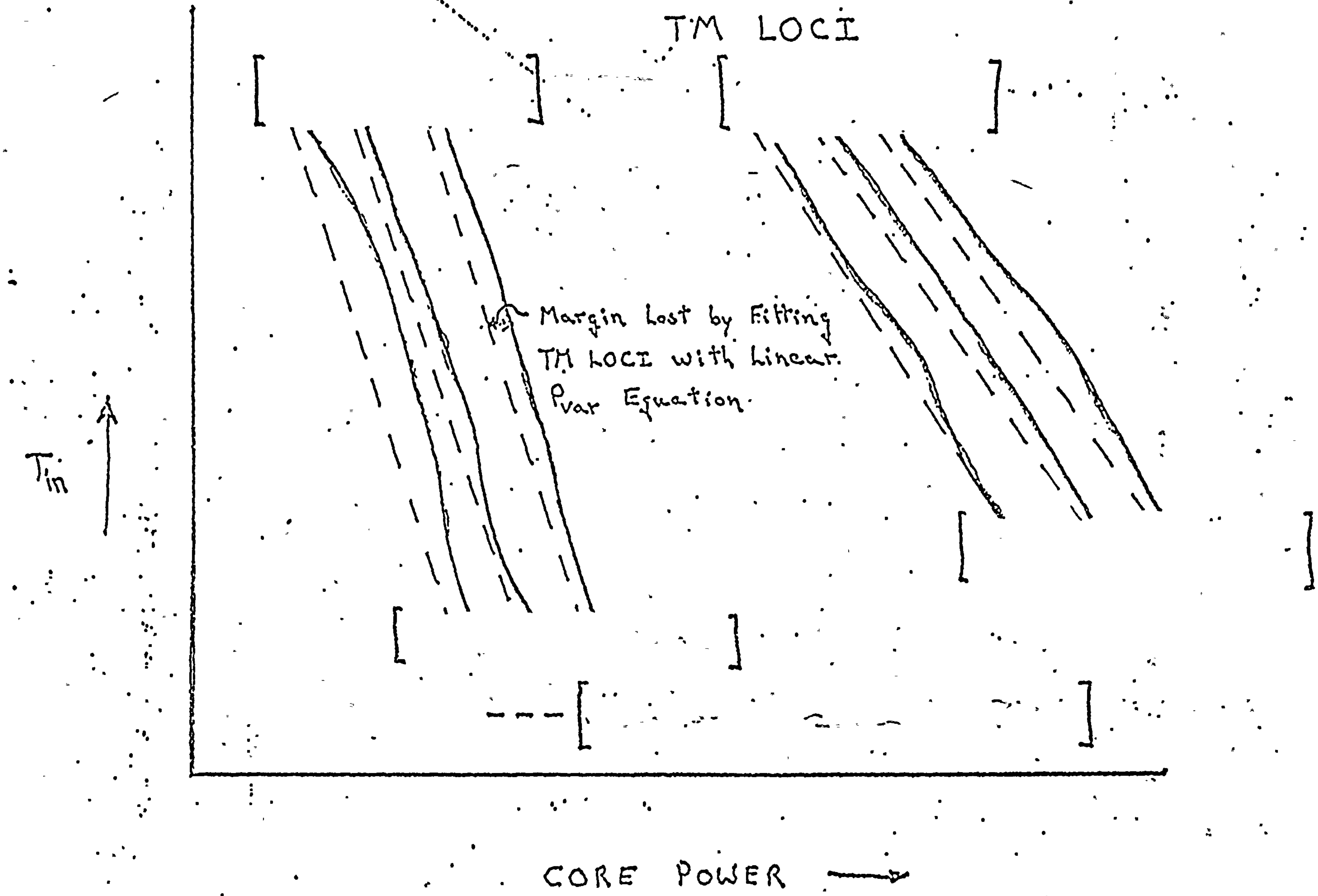
TM LOCI



SLIDE 5

FIT TO TM LOCI

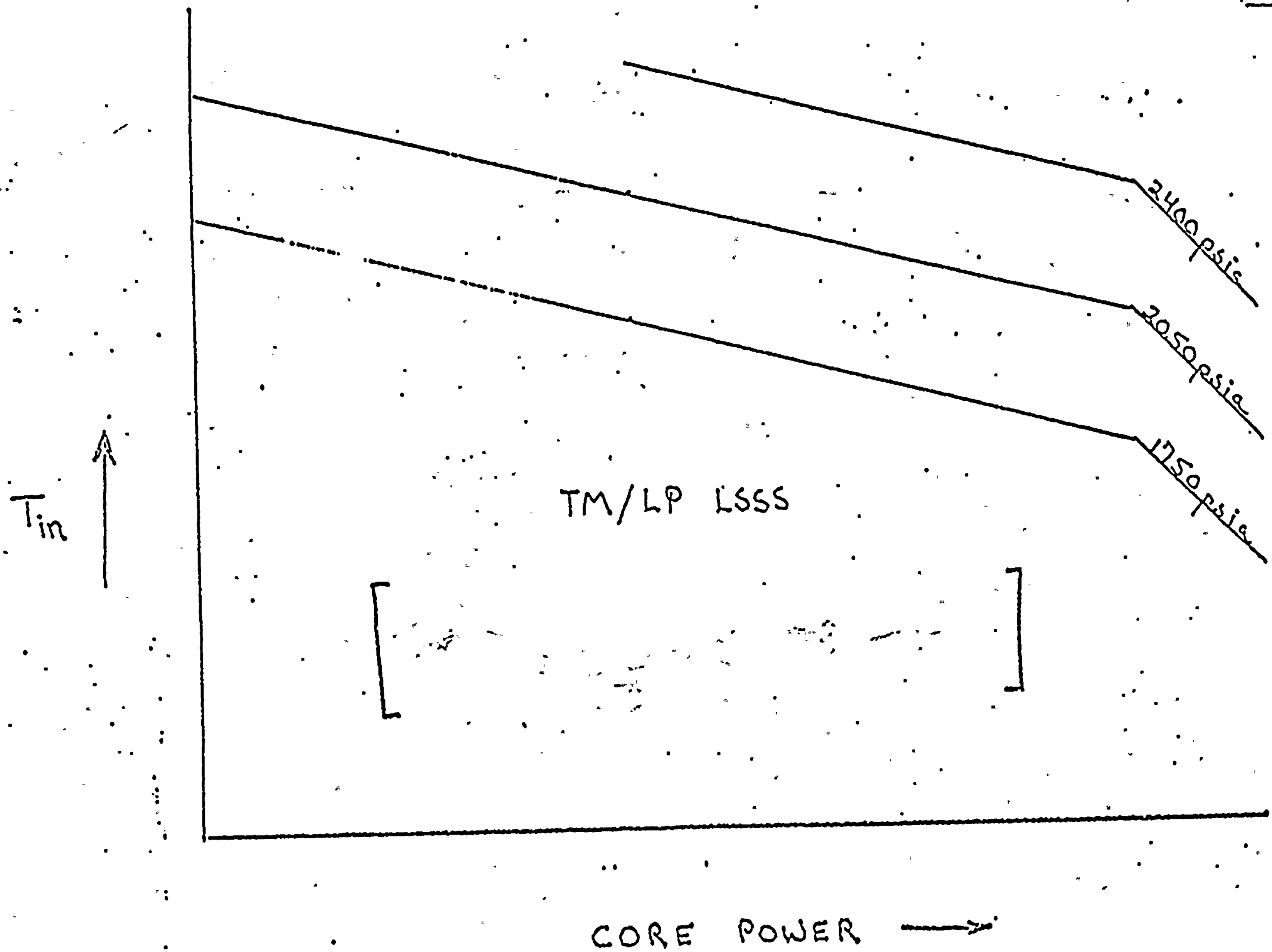
3, 5



3, 5

3, 5

SLIDE 6



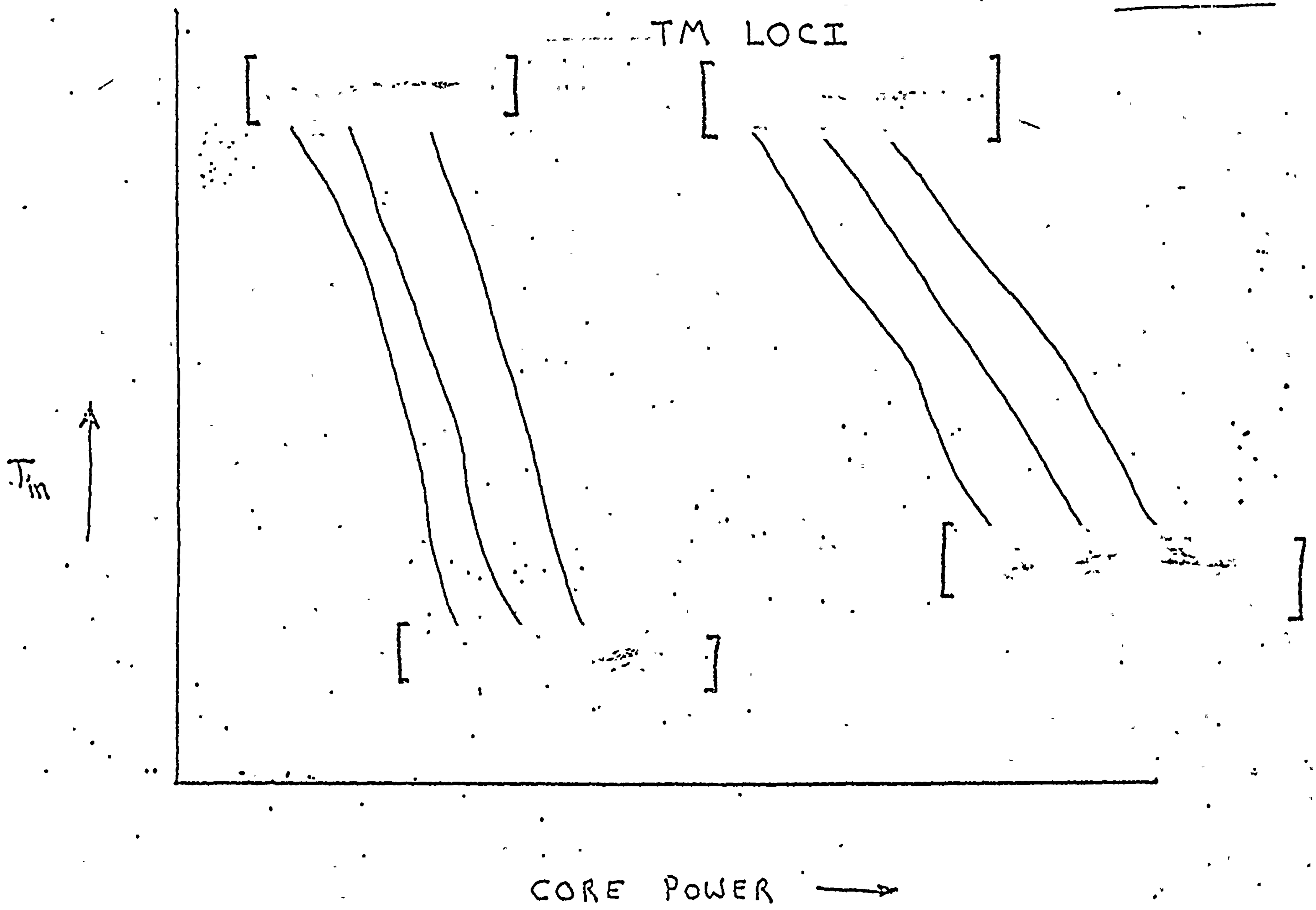
3, 5

[illegible]

SLIDE 7

3, 5

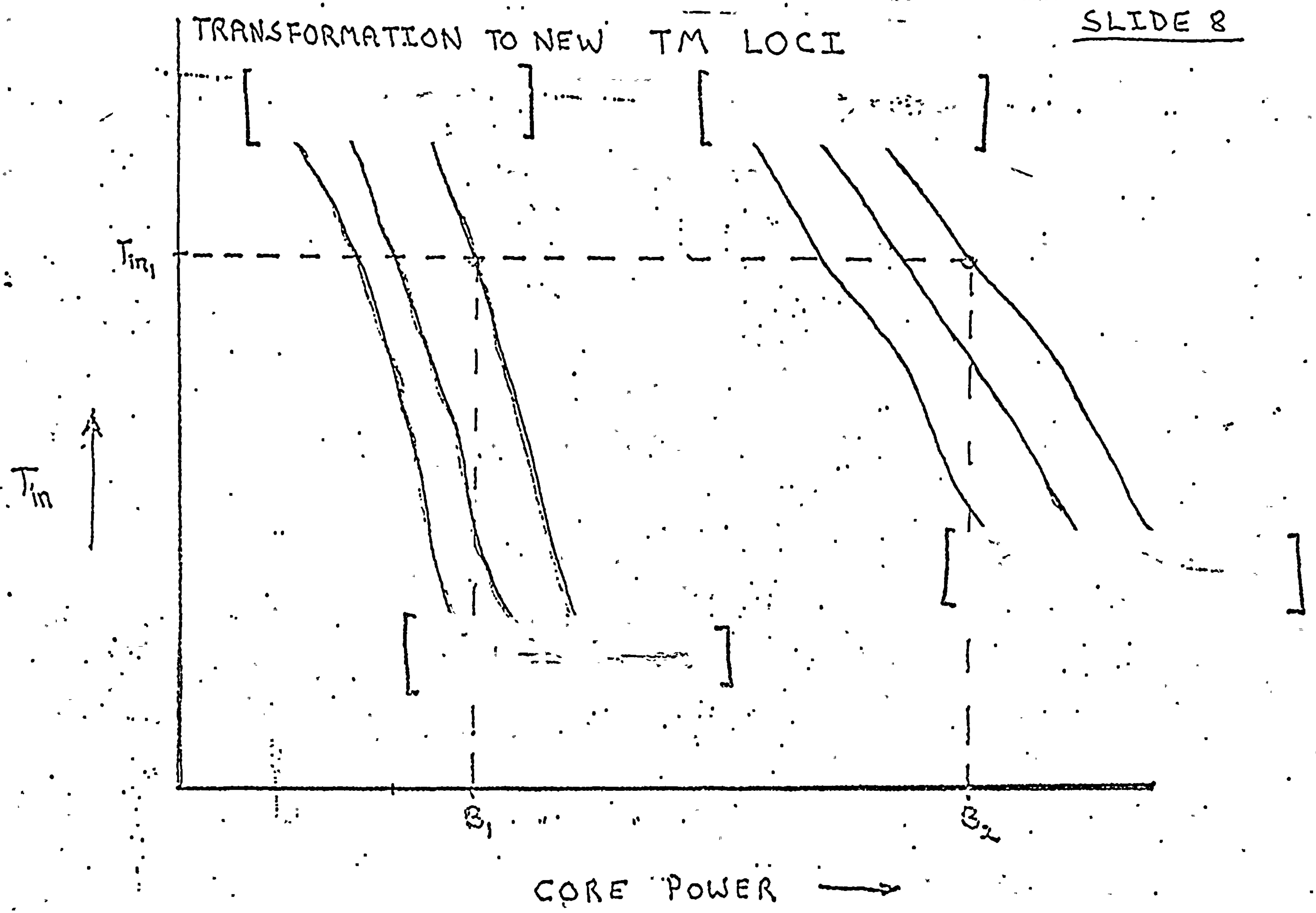
3, 5



SLIDE 8

TRANSFORMATION TO NEW TM LOCI

3, 5.

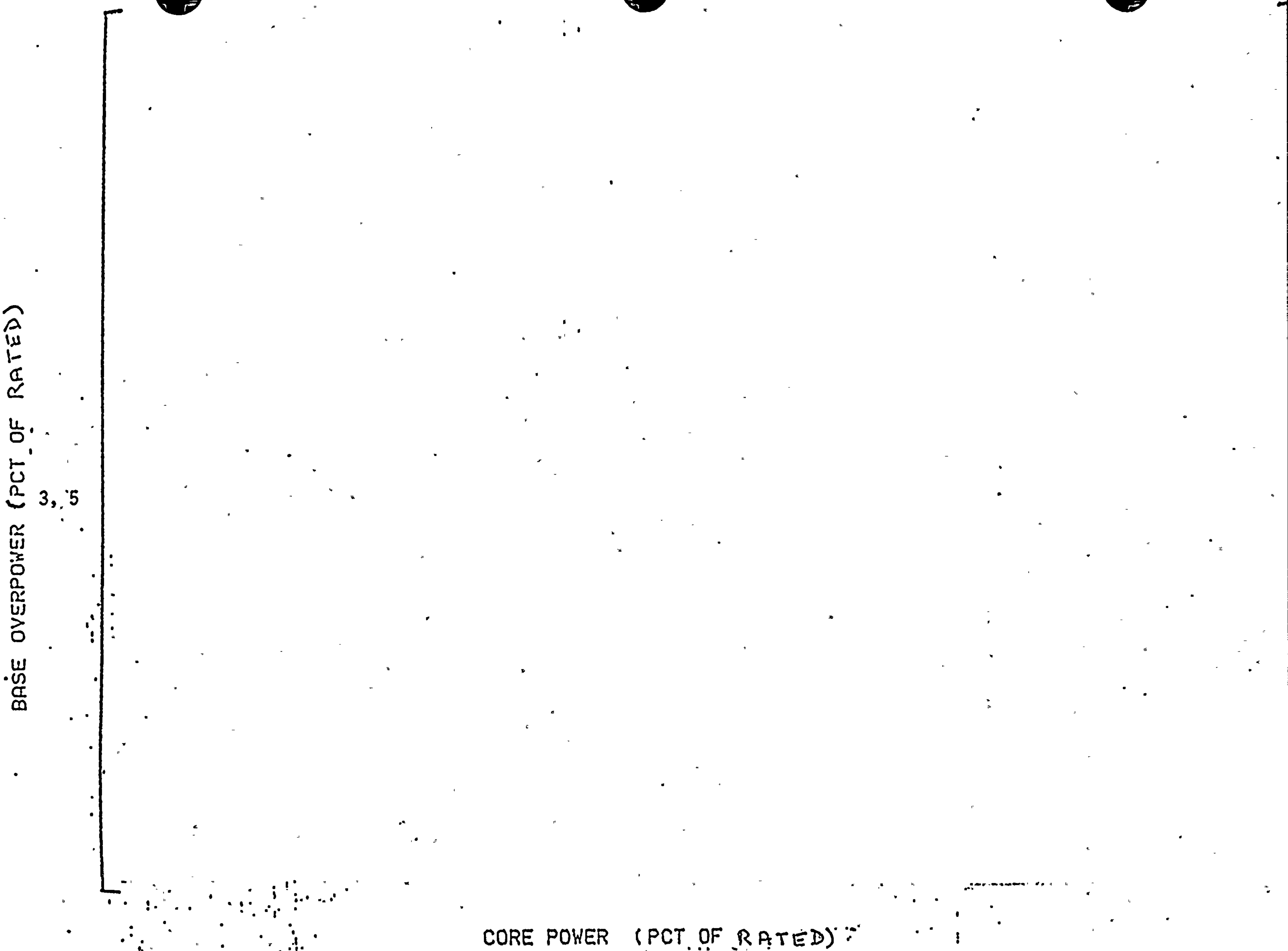


3, 5

SLIDE 9

3,5

CORE POWER ⟶


BASE OVERPOWER (PCT OF RATED)
3.5
CORE POWER (PCT OF RATED)

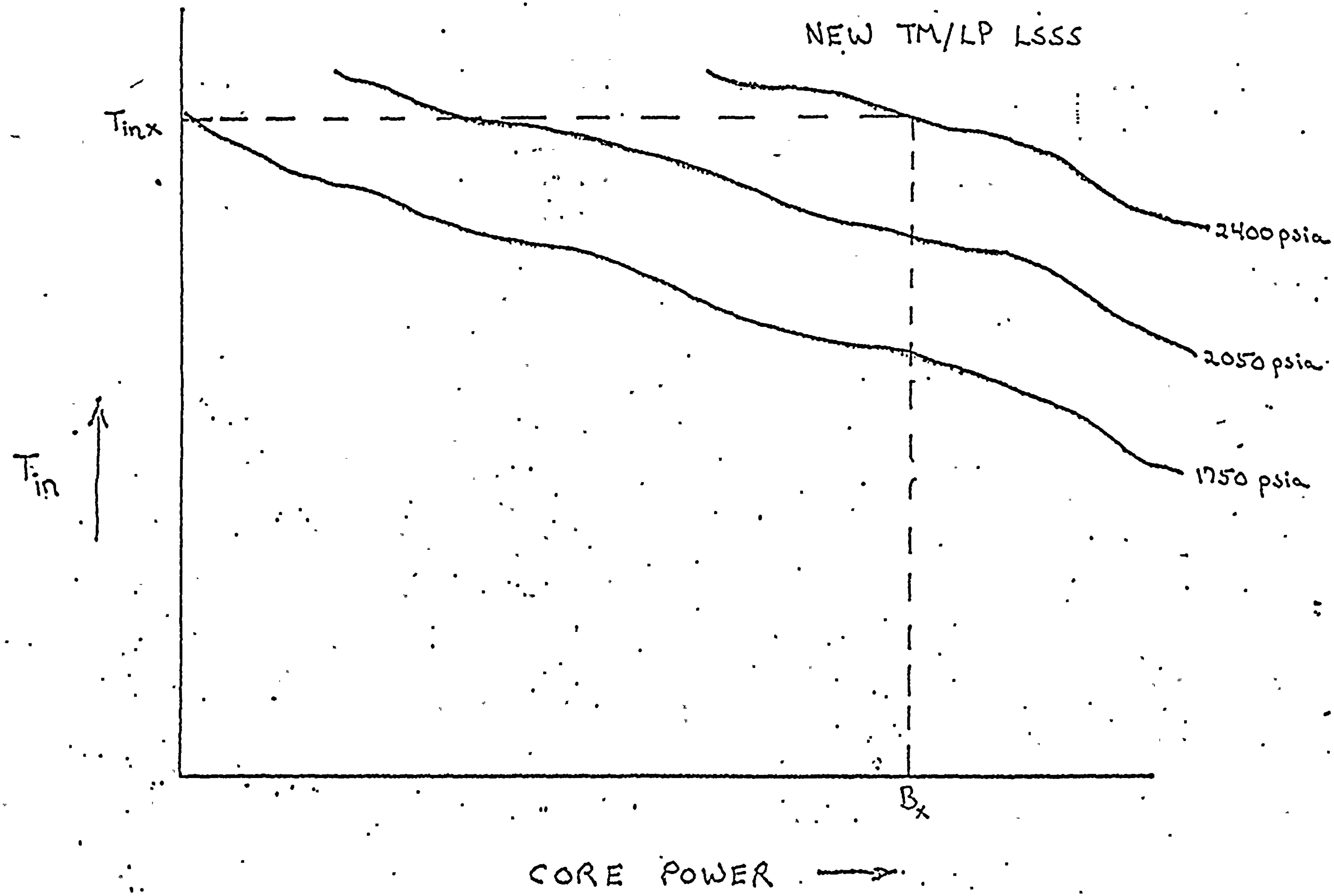
BASE OVERPOWER (PCT OF RATED)

3.5

SLIDE 10

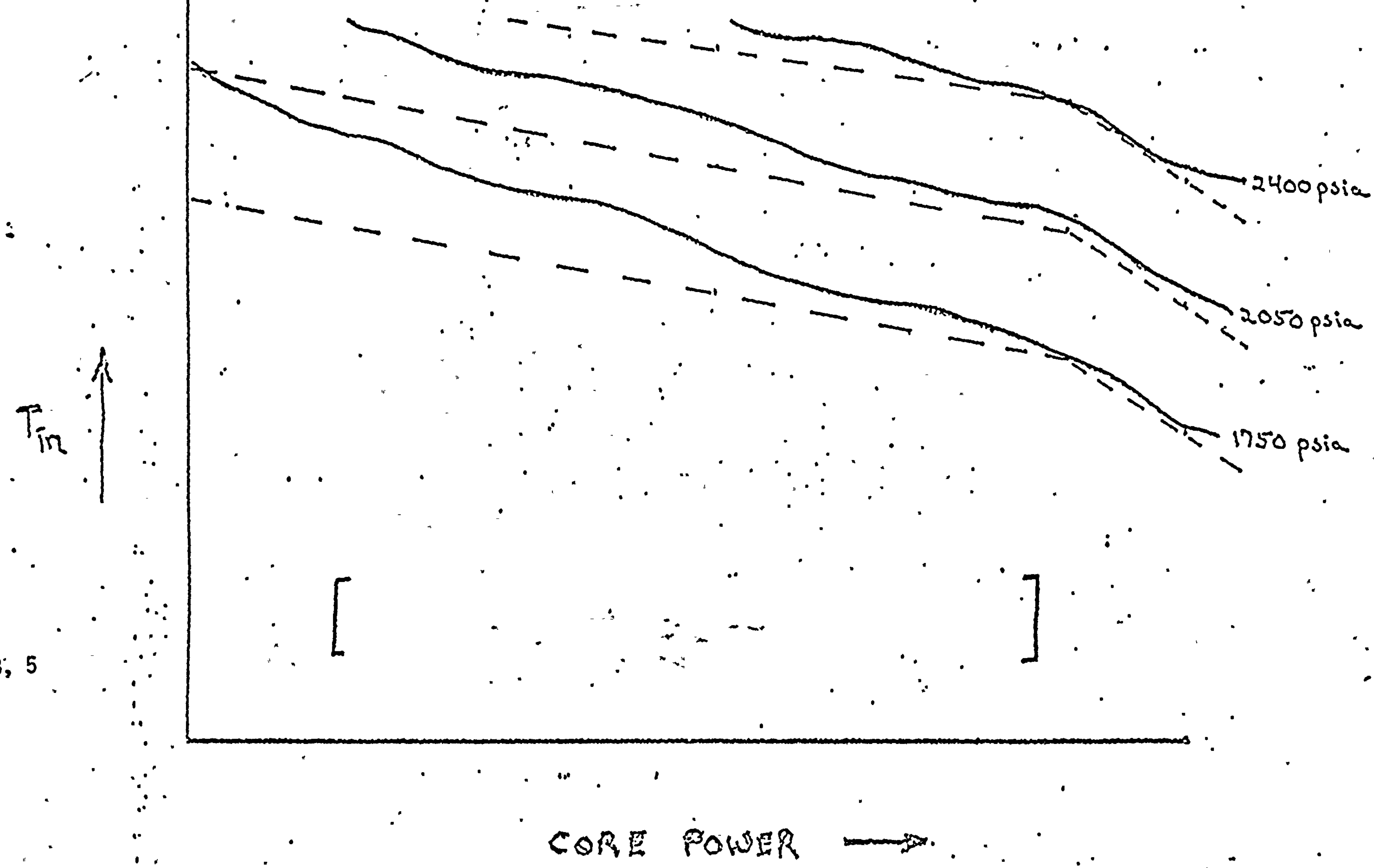
3, 5

CORE POWER →


NEW TM/LP LSSS
T_{inx}
T_{in}
2400 psia
2050 psia
1750 psia
B_x
CORE POWER

SLIDE 12

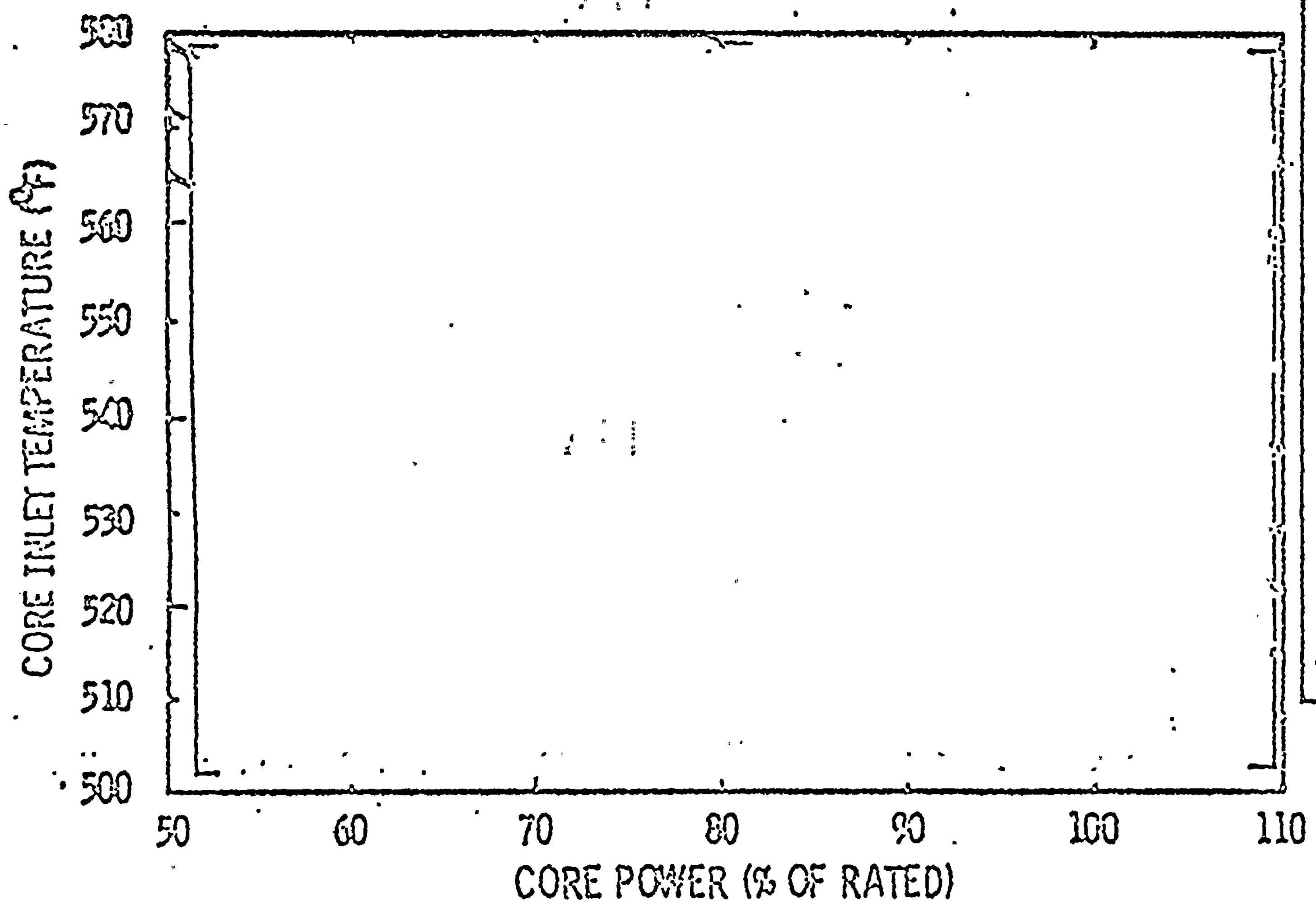
FIT TO NEW TM/LP LSSS



3, 5

SLIDE 13

3, 5



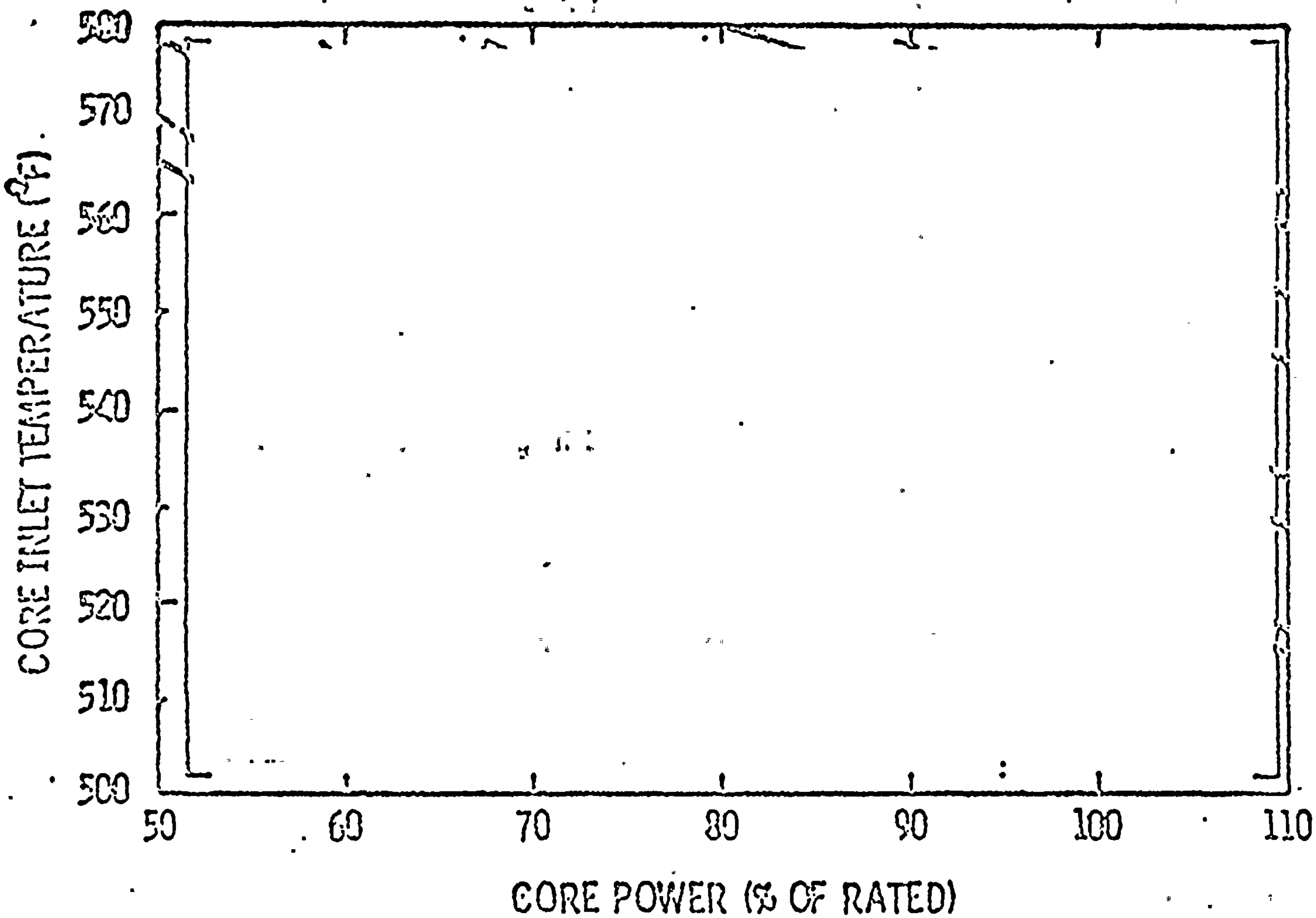
CORE INLET TEMPERATURE (°F)
580
570
560
550
540
530
520
510
500
50
60
70
80
90
100
110
CORE POWER (% OF RATED)

[illegible]

[illegible]

SLIDE 14

3, 5



CORE INLET TEMPERATURE (°F)
580
570
560
550
540
530
520
510
500
50
60
70
80
90
100
110
CORE POWER (% OF RATED)

SECTION 4

PARTIAL CREDIT for TORC/CE-1 in ASI MONITORING LIMITS

SLIDE 1

The purpose of this portion of the slide description is to explain the technical details of the plan to take partial credit for TORC/CE-1 thermal margin gains in the monitoring system, i.e., the ASI Monitoring Limits. No credit is taken for TORC in the trip setting.

Basically, the objective is to demonstrate the existence of inherent conservatism in the "open hot channel" design code COSMO/W-3 relative to the "open core" design code TORC/CE-1. Some background on these methodologies will provide the thermal-hydraulic bases for the conservatisms.

There are three fundamental models used in the thermal-hydraulic analysis of pressurized water reactors. The closed hot channel, the open hot channel, and the open core methods. Closed hot channel postulates coolant in a sub-channel bounded by four (4) fuel rods, is heated through its axial flow upward through the core without any mass or energy exchange with its neighboring sub-channels. It can be viewed as a sub-channel with an inpenetrable lateral boundary. The axial pressure drop across the core computed using the core average flow rate, provides the boundary conditions for determining the closed hot channel flow rate. That is, the flow rate in the closed hot channel is

raised or lowered as required until the axial pressure drop in the closed hot channel equals the core average pressure drop.

The COSMO code is a open hot channel thermal hydraulic model. That is, it permits an exchange of energy from the hot channel with its neighboring sub-channel. The code does not solve the energy and momentum equations explicity. Instead, separate analysis is performed to determine mixing factors. Mixing factors are defined as the enthalpy rise in a axial length of the hot channel where turbulent interchange of coolant is allowed divided by the enthalpy rise in the same length of hot channel where no turbulent interchange is allowed (e.g., the closed channel enthalpy rise). Mixing factors are numerical values less than unity. They are applied as multipliers on the hot channel enthalpy rise. Another feature of the COSMO model is in the evaluation of the hot channel flow rate. The boundary conditions start with the average pressure drop, but proceed to the hot assembly to determine the local nodal axial pressure gradient imposed on the hot sub-channel. Important features of the COSMO open hot channel analysis are that it computes the impact of turbulent interchange of coolant in the hot sub-channel with its neighbor sub-channels and it places the hot sub-channels in the hot assembly for the determination of axial boundary conditions for the computation of sub-channel flow.

The TORC/CE-1 code was developed from the COBRA-III code. described

in the literature. The TORC/CE-1 code differs from both of the models described above in that explicit solution of the momentum and energy equations is performed on a core-wide basis. Both turbulent interchange (mixing of coolant with the effect of reduced enthalpy with no net mass exchange) and divergent cross flow (the net exchange of mass from one region of the core to another) is permitted if calculated to occur.

SLIDE 2

It is important to note that the thermal-hydraulic design bases of the plant does not change. COSMO/W-3 remains the model on which the safety analysis is performed. No credit is taken in the trip settings for the inherent conservatisms in the COSMO/W-3 model relative to TORC/CE-1. COSMO/W-3 is used to identify the limiting anticipated operational occurrence and in the assessment of the required margin. The acceptability of the consequences of the analyzed events is based on the DNBR computed using COSMO/W-3. Therefore, the transient safety analysis does not change and need not be repeated. The required margin maintained in the monitoring system as quantified using COSMO/W-3 does not change and no credit is taken in the thermal margin low pressure limiting safety systems settings.

SLIDE 3

This slide shows the type of margin gain which we have identified by taking analysis of hundreds of cases where process variables

are permuted and running these cases with COSMO/W-3 and TORC/CE-1. The Y axis shows overpower margin. The overpower margin is defined as the percent or fraction of rated core power at which a DNBR limit is achieved in the analysis. The margin gain shown as the difference in margin as indicative of the inherent conservatism in COSMO/W-3 relative to TORC/CE-1.

SLIDE 4

This slide shows explicit analyses of the comparison of COSMO and TORC for several hundred cases. This analysis simulates the
3, 5 P_{fdn} analysis. Here using [ ] typical shapes, there is at minimum
[ ] thermal margin gain by using TORC relative to COSMO. The maximum gain in thermal margin as evaluated from this data equals
3, 5 about [ ]

SLIDE 5

For an analysis over all operating space, 691 data points were examined. This data was representative of Combustion Engineering's 16x16 fuel assembly.

Operating space in temperature, pressure, flow, integrated radial peak and axial shape were permutted. Results of this indicate an average margin gain of 7% overpower margin. As was stated on the previous slide, the operating point margin gain equals about 12% overpower. From this analysis, Combustion Engineering will claim a 3% net credit for TORC in quantifying the inherent conservatism in the open hot channel COSMO/W-3 thermal margin code. Although the data was generated for the 16x16 reactor fuel,

there are no fundamental thermal-hydraulic concerns with regard to the outcome of an analysis of Combustion Engineering 14x14 fuel using a similar data base. That is, we anticipate the thermal margin gain, if explicit analyses on 14x14 fuel were to be performed, would be of similar magnitude. This 3% is applied in the ASI Monitoring System.

SLIDE 6

The conclusions from this analysis then, are that a 3% TORC/CE-1 limiting condition operation credit is a conservative estimate of the increased available margin relative to COSMO/W-3. A typical
3, 5 16x16 P_{fdn} analysis has shown [ ] minimum and [ ] maximum gain. Over all operating space a 7% average gain has been demonstrated. This latter assessment was performed over a wider parameter space than allowed by current 14x14 plant technical specifications.

SLIDE 1

PURPOSE

THE PURPOSE OF THIS PORTION OF THE MEETING IS TO EXPLAIN THE TECHNICAL DETAILS OF THE PLAN TO TAKE PARTIAL CREDIT FOR TORC/CE-1 THERMAL MARGIN GAINS IN THE MONITORING SYSTEM. (ASI MONITORING LIMITS).

SLIDE 2

MAJOR ELEMENTS

1. DESIGN BASIS DOES NOT CHANGE

COSMO/W-3 IS USED AS THE DESIGN BASIS UPON WHICH THE SAFETY ANALYSIS IS PERFORMED.

A. LIMITING AOO
B. QUANTIFIES THE REQUIRED MARGIN
C. ACCEPTABLE CONSEQUENCES

THEREFORE:

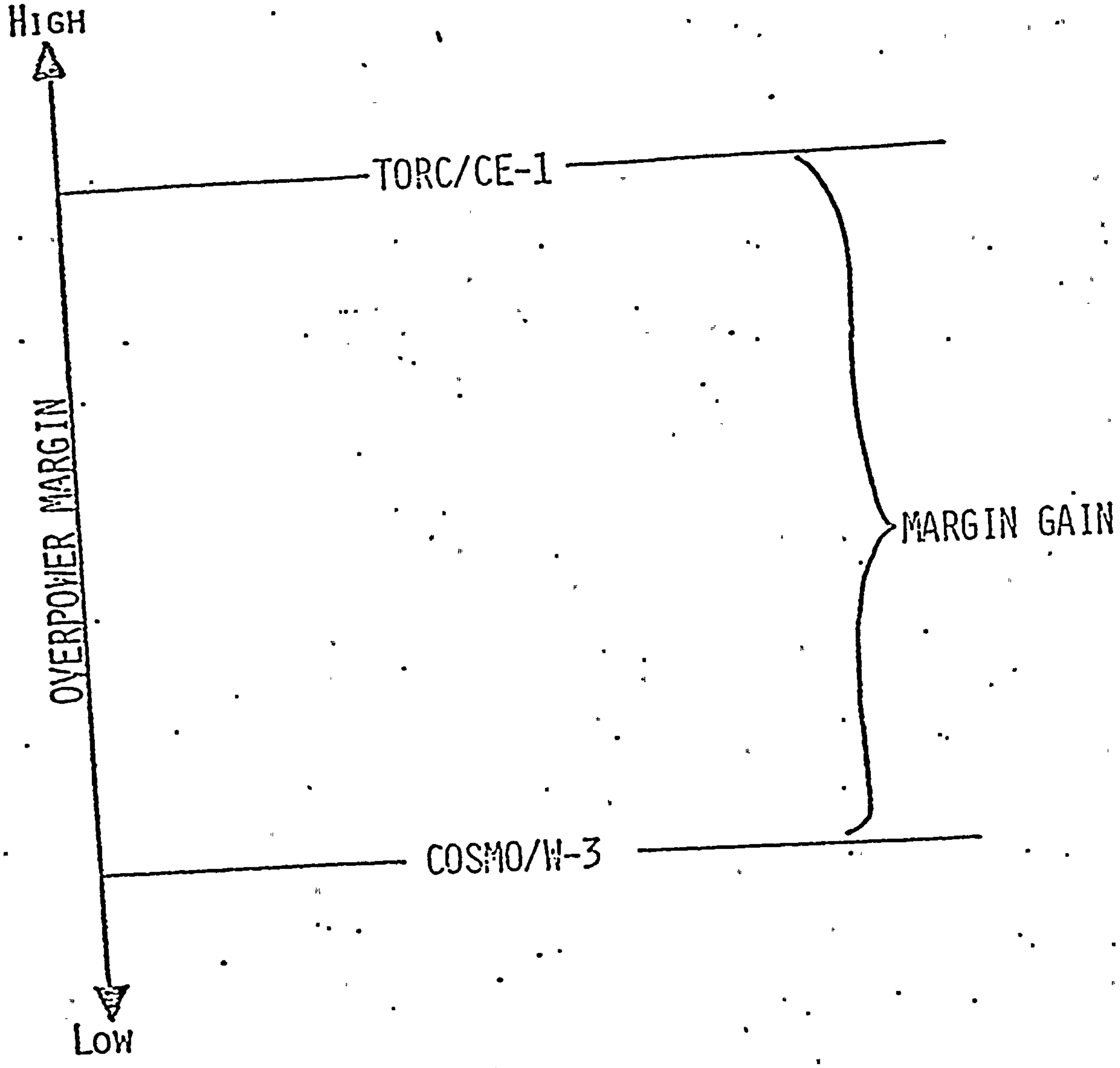
TRANSIENT ANALYSIS DOES NOT CHANGE.

REQUIRED MARGIN MAINTAINED IN THE MONITORING SYSTEM DOES NOT CHANGE.

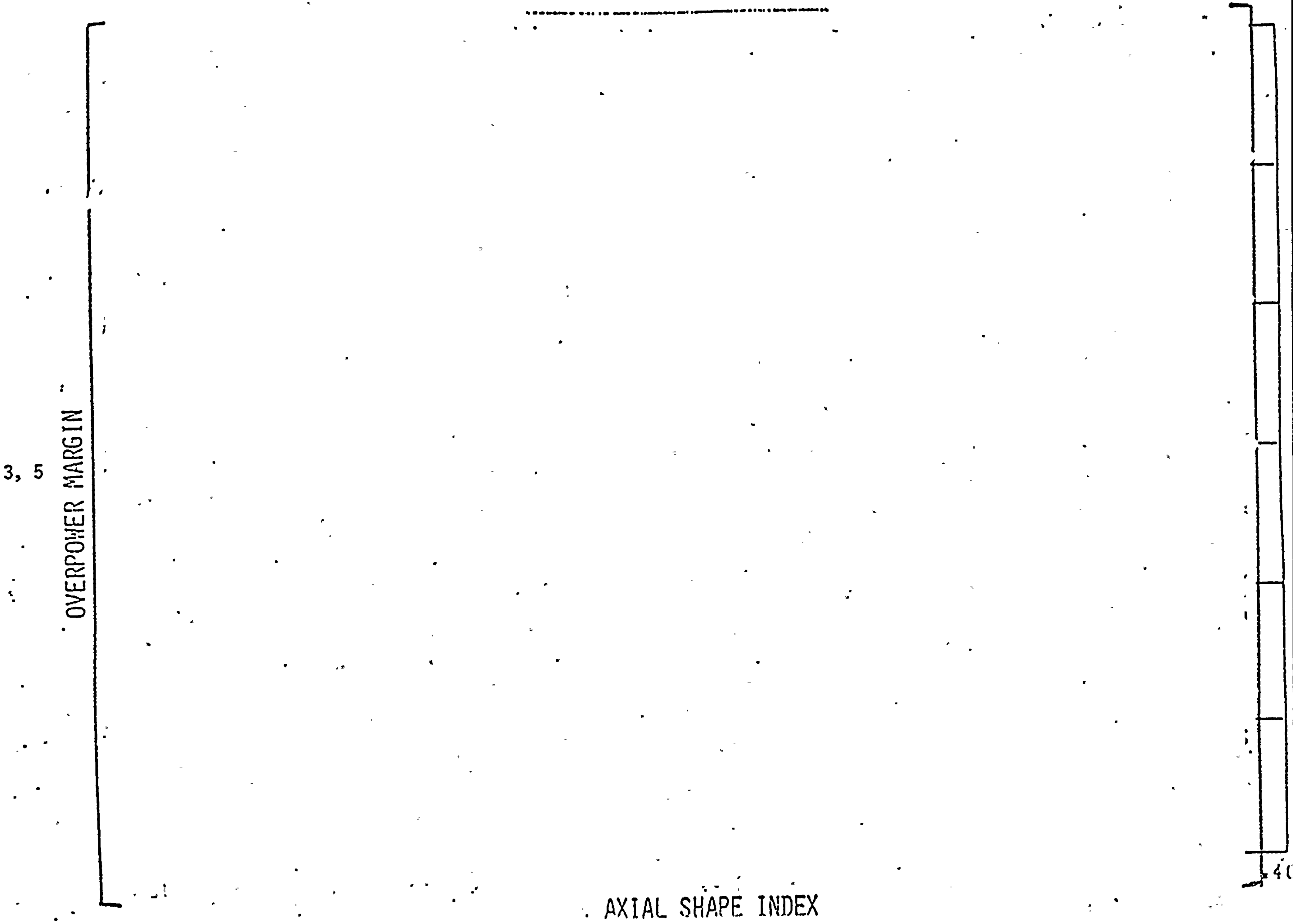
NO CREDIT TAKEN IN TM/LP LSSS.

SLIDE 3

OVERPOWER MARGIN GAIN



OVERPOWER MARGIN Vs. ASI



SLIDE 5

TORC/CE-1 GAIN

Number of Points in Parameter Space	691
Average Gain, % Overpower	7
Operating Point Gain, % Overpower	12

SLIDE 6

CONCLUSIONS

- 3% TORC/CE-1 LCO CREDIT IS A CONSERVATIVE ESTIMATE OF THE INCREASED AVAILABLE MARGIN RELATIVE TO COSMO/W-3.

- A TYPICAL 16 X 16 PLANT ANALYSIS HAS SHOWN [ ] MINIMUM AND [ ] MAXIMUM GAINS. 3,5

- THIS ASSESSMENT WAS PERFORMED OVER WIDER PARAMETER SPACE THEN ALLOWED BY CURRENT 14 X 14 PLANT TECH. SPECS.