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ENC SETPOINT METHODOLOGY FOR CE REACTORS

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FOR CE REACTORS

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1.0 INTRODUCTION

Pressurized water reactors constructed by the Combustion Engineering Company use an analog protection and monitoring system for the core protection system. This system assures the safe operation of the reactor core during steady-state or transient conditions. The protection system equipment is designed and programmed to act automatically to prevent or suppress conditions that could result in exceeding the design specifications of the fuel or the reactor core. This protection system, called the Reactor Protection System (RPS), continuously monitors core parameters related to the Limiting Safety System Settings (LSSS).

The fuel design limits are protected by the RPS and prevent any potential fuel damage due to the departure from nucleate boiling (DNB) and fuel centerline melt. Correlations between the fuel design limits and the RPS monitored variables are determined to assure that these limits are not exceeded. These calculated limiting safety system settings (LSSS) values are programmed into the reactor protection system to assure that appropriate action is taken if the monitored values equal the limiting safety system settings.

The Limiting Conditions of Operation (LCO's) are calculated parameters that are correlated to the RPS monitored parameters. The LCO's are more restrictive than the LSSS and the core is administratively operated within the bounds of the LCO limits. The LCO's are determined such that fuel design limits will not be exceeded during a transient which may occur while the plant is being operated within the envelope of the LCO's.

Normal operation of the plant is usually well within the bounds of the LCO's and is maintained administratively by the reactor operator.

Exxon Nuclear Company has developed a methodology to calculate the Limiting Safety System Settings (LSSS) and Limiting Conditions of Operation (LCO) which are associated with the Specified Acceptable Fuel Design Limits (SAFDL). This methodology is compatible and similar to that currently used for Combustion Engineering reactors. This report and the references describe the ENC methodology used to calculate setpoints for CE reactors and provides the basis required for the justification of the methodology.

2.0 SUMMARY

An extensive review and evaluation of the methodology required to determine setpoints for CE PWR's have been conducted. It is concluded, based on verification against current setpoints in both Palisades and Fort Calhoun, that the general methods and computer codes currently used at ENC are acceptable for setpoint calculations.

The reactor protection system (RPS) continuously monitors those core parameters that assure the safe operation of the reactor core when the values of the parameters are within the limits specified by the Limiting Safety System Settings (LSSS) and Limiting Conditions of Operaton (LCO).

The determination of the LSSS's and LCO's addressed in this report are limited to those safety values which are calculated on a cycle by cycle basis. The LSSS's that are to be calculated include the Axial Power Distribution (APD) trip setpoints and the Thermal Margin/Low Pressure (TM/LP) setpoints. The APD setpoints protect the fuel from exceeding the fuel centerline melt design limit while the TM/LP trip setpoints protect the fuel from penetrating DNB limits. The TM/LP trip setpoints are related to the APD trip setpoints through the axial power distribution profiles generated during the APD setpoint calculations. These axial power distributions are directly used in the TM/LP trip setpoint determinations. The LCO's are also evaluated on a cycle by cycle basis and are generated through analysis of those reactor transients which are not protected by the LSSS's. These transients include the Dropped Rod and Loss of Flow anticipated operational occurances. The Loss of Coolant accident is also

evaluated and incorporated into the LCO's on a cycle basis. Some of the inputs to the calculation of the LCO values for these transients are determined from the APD and TM/LP LSSS calculations. The axial power distribution profiles determined in the APD calculations and used in the TM/LP calculation are also used in the determination of the values of the LCO's.

The methodology employed at ENC to analytically determine the LSSS' and LCO's are discussed in this report. General methods and programs used in the setpoint calculations are the same as those currently used at Exxon Nuclear Company for thermal-hydraulics, safety and neutronics calculations.

These NRC approved design methods which are described in Section 6.1. for neturonics, Section 6.2, for thermal-hydraulics and Reference 15 for nuclear safety, are directly applicable for the determination of setpoints in Combustion Engineering PWR's. The specific neutronic design tools include the computer codes XPOSE(10) PDQ7/HARMONY(11,12) and XTG(13). The Thermal-Hydraulic computer codes used in the analysis include XCOBRA-IIIC(3) and the PTS-PWR(5) codes.

3.0 REACTOR PROTECTION SYSTEM

The Reactor Protection System (RPS) is designed to assure that the reactor is operated in a safe and conservative manner. Input parameters to the RPS are calculated and denoted as Limiting Safety System Settings (LSSS). In addition to the LSSS the plant is operated within the Limiting Conditions of Operation (LCO's). These parameters, the LSSS's and LCO's, are an integral part of the plant technical specifications and are reviewed for each cycle.

The analog Reactor Protection System provides assurance in addition to administrative and other procedures that the plant is operated within the NSSS design and Technical Specifications. The NSSS design and Technical Specifications governing the reactor operation ensure that the specified acceptable fuel design limits (SAFDL), and other safety limits are not violated as a consequence of any Anticipated Operational Occurance (AOO), or the consequences of any other Postulated Accident (PA). These criteria are met provided that (1) the actual Reactor Protection System Settings (setpoints) are equal to or conservative relative to the LSSS, (2) the actual plant operating conditions are within the LCO's and (3) the equipment not associated with the incident operates as designed. Figure 3.1 is a functional diagram of the Reactor Protection System.

The specified acceptable fuel design limits (SAFDL's) are analytically or experimentally based limits for both the fuel and cladding. These limits are used to establish the LSSS's and LCO's which in turn provide conservative operating limits for the core. The specific SAFDL's used to establish the setpoints are:

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- The maximum linear heat rate (LHR) which is determined to result in fuel centerline melt.
- The DNBR corresponding to the accepted criteria which assumes that DNB will not occur.

These specified acceptable fuel design limits must not be violated during those conditions of normal operation or any anticipated operational occurances which are expected to occur one or more times during the life of the plant. Examples of some anticipated operational occurances are rod drop, excessive load, loss of load, and xenon burnout and decay.

Other postulated accidents that are not expected to occur during the life of the plant are also evaluated to assure that the adequacy of the LSSS's and LCO's limit the accidents to acceptable levels. These postulated accidents are caused by severe natural phenomena or unlikely component defects. Evaluation of those accidents complete the balance of the establishment of the LSSS & LCO's as described in this document.

The RPS consists of automatic protection equipment that is programmed with values from the analysis establishing the LSSS's. The RPS automatically prevents plant operation beyond any SAFDL and automatically initiates action to ensure protection against violation of licensed operating limits.

The RPS monitors specific core parameters and initiates a scram when the limiting values of the following parameters are exceeded:

- Low Steam Generator Pressure
- Low Steam Generator Water Level

- Variable High Power
- Thermal Margin/Low Pressure (TM/LP)
- Axial Power Distribution (APD)
- High Containment Pressure
- Low Reactor Coolant Flow
- High Pressurizer Pressure

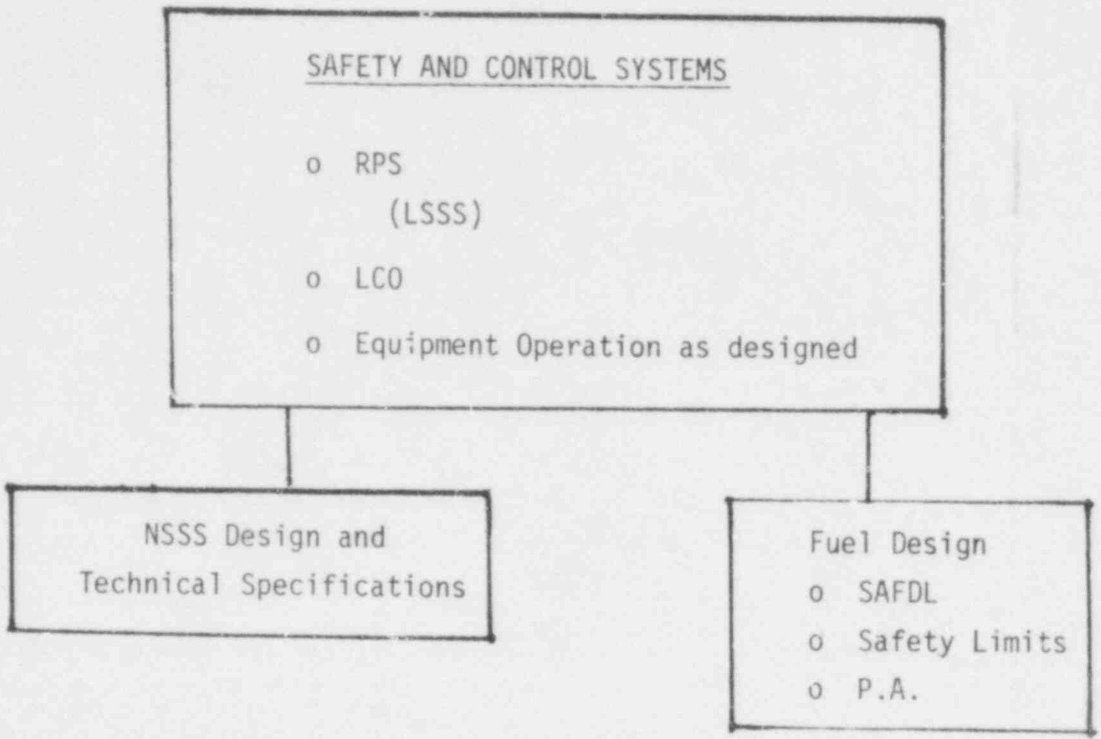
Of the parameters listed above only the Variable High Power, TM/LP and APD trips require review and possible adjustment for changes in fuel design. This report describes the calculational techniques to be used to determine the setpoints for these plant functions.

The Axial Power Distribution trip is based on the linear heat rate at which fuel centerline melt is calculated to occur. Calculations are performed determining the peak linear heat rate as a function of the axial shape index. The axial shape index in the core is measured with the split excore detectors and the APD trip will act to preclude violation of centerline melt limits due to an axial power maldistribution.

The Thermal Margin/Low Pressure (TM/LP) trip is based on departure from nucleate boiling criteria, (DNB) analysis for the reload fuel design. The TM/LP trip will shut the plant down when the core pressure is less than that allowed by the calculated thermal margin limit lines which are functions of core coolant temperature, power and pressure. The TM/LP trip protects the core from attaining a DNBR value below an accepted value.

The variable high power or ΔT overpower trip limits are consistent with values used in calculating the symmetric offset and TM/LP trip setpoints. Values for nuclear peaking are determined by the control rod power dependent insertion limits of the core. The nuclear overpower trip ensures a reactor shutdown before the limiting values of nuclear peaking are exceeded.

Figure 3.1 Safety and Control Systems



4.0 LIMITING SAFETY SYSTEM SETTINGS (LSSS)

4.1 AXIAL POWER DISTRIBUTION (APD) LIMITING SAFETY SYSTEM SETTING (LSSS)

The Reactor Protection System (RPS) axial power distribution (APD) trip limits for CE reactors are the limiting values of monitored parameters that will initiate a reactor trip prior to exceeding fuel centerline melt conditions. The parameters in the core that are monitored for this trip include the core power level and the measured axial shape index. The limiting values of these two monitored parameters are determined from calculations which provide correlations between axial shape index and power peaking. These correlations are used to determine allowable power level as a function of axial shape index.

Following is a detailed description of methods used to determine the APD Limiting Safety System Settings. A typical specific acceptable fuel design limit used in the LSSS APD trip calculation for centerline melt is ∿21 kw/ft. This value may change depending on the design of the fuel or burnup.

4.1.1 Core Axial Shape Index and Nuclear Peaking

The excore axial shape index, Se, is measured with the split excore detectors. It is defined as the difference in signal generated in the lower half of the core (L) and upper half of the core (U) divided by the sum of two signals (L+U). When the measured value of Se equals any one of the calculated axial shape indeces or APD limits, a reactor trip is initiated. The equation for the axial shape index is:

$$Se = \frac{L - U}{L + U}$$

The excore axial shape index, Se, is related to the core average axial shape index, S, through the shape annealing factor (SAF) and the

rod shadowing adjustment factor, F. A description of the shape annealing factor and adjustment factor is presented in Section 4.1.2. During normal operation of the reactor core, the core average axial shape index, S, may vary between $\pm5\%$ throughout the cycle. Figure 4.1 shows the calculated all rods out (ARO) axial shape index for Fort Calhoun Cycle 4.

The peak linear heat rate in the core occurs at the position of the maximum heat flux peaking factor, F_Q^T. The maximum heat flux peaking factor, F_Q^T, is referred to as the "hot spot" in the core. It is the ratio of the maximum linear heat generation rate in the core to the average linear heat generation rate in the core. F_Q^T is therefore the product of F_{xy} (the ratio of the power of the peak fuel pin to the average fuel pin in the core) times the core average axial power peaking factor, F_z, at the same axial position. The heat flux peaking factor with uncertainties is therefore:

$$F_Q^T = F_{xy} \times F_z \times F_u$$

where F_u = uncertainties.

Likewise, the maximum value of F_Q^T occurs in cores with large axial shape indeces; i.e., large values of F_z.

Figure 4.2 shows the variation in the F_Q^T peaking factor as a function of core height for a -7% axial shape index Fort Calhoun fuel rod.

In order to determine the values for axial shape index set-points, pairs of axial shape index, S, and corresponding total nuclear peaking factors, F_Q^T, are determined for each cycle. The values of the axial shape index and maximum power peaking depend significantly upon the core burnup, CEA position, and xenon distribution in the core. By tabulation of the many

possible nuclear heat flux peaking factors, F_Q^T, which result from particular axial shape indices, axial shape index LSSS curves can be determined for each cycle. The following is a description of the methodology used in determining axial power profiles and ordered pairs of axial shape index and nuclear power peaking factors, F_Q^T.

1. Setup core models with PDQ in two dimensions and XTG in three dimensions. Deplete the cycle with the two codes. Use detailed values of peaking factors from PDQ in XTG to determine F_{xy} values for each assembly. Report for rodded and unrodded configurations.
2. Select a burnup (use BOC, MOC, and/or EOC).
3. Setup a one-dimensional XTG axial model of the core at the burnup selected in step 2.
4. Generate a xenon oscillation in the one-dimensional model obtaining the core average axial power distributions and axial shape index. See Figure 4.3.
5. Select an axial power distribution from step 4 and run several cases inserting CEA's to the power dependent insertion limits (PDIL). With the CEA's inserted to a specific power dependent insertion limit, run cases varying power below and up to the permissible power level. Use the 1-D XTG calculation to determine the peak axial power distribution, F(z), and axial shape index, S.

6. Obtain the planar radial peaking factor, F_{xy}, for the distinct CEA configuration selected in step 5 using 2-D PDQ diffusion theory and 3D XTG calculations from step 1. Use the most restrictive (largest) limit of either the desired technical specification value or the calculated value of F_{xy}.
7. Determine the ordered pairs of percent power, p, vs axial offset, S, by solving the equation.

$$P = \% \text{ Power } = \frac{100 \times \text{centerline melt (kw/ft)}}{F_Q^T \times W_{avg}}$$

where:

$F_Q^T = F_{xy} \times F_z \times F_u$ (Reference Figure 4.4)

W_{avg} = core average linear heat generation rate at HFP.

8. Return to 5 and select another axial power distribution. A large number of power distributions is needed to determine an adequate correlation between axial shape indices and corresponding total peaking factors.

The above procedure is repeated for BOC, MOC, and EOC conditions. The resulting data is presented as % power versus axial shape index, S. The most limiting values are used in the LSSS axial shape index trip limits.

4.1.2 Peripheral Axial Shape Index

The axial shape index, S, described in Section 4.1 is the core average axial shape index based on the core average axial power distribution. The measured core axial shape index used with the Reactor Protection System is measured with the excore detectors and is denoted by Se. Generally,

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Se does not equal S. The axial shape index response of the excore detectors is primarily due to the axial shape index of those assemblies located close to the excore detectors, i.e., the peripheral assemblies, which is denoted as Sp. The offset of the peripheral assemblies, Sp, is related to the measured offset of the excore detectors, Se, through the shape annealing Factor, SAF, determined from core measurements:

$$Sp = SAF * Se$$

In the absence of control rods the core average axial shape index, S is equal to the axial shape index of the peripheral assemblies, Sp. However, in rodded core configurations, Sp does not always equal S and adjustment factors must be applied to Sp to derive actual axial shape index, S.

The calculations to determine the rod shadowing effects are made with 3D-XTG and PDQ7. The axial offset of the peripheral assemblies which contribute to the excore detector response and the core average axial offset is determined as a function of CEA insertion. An adjustment factor is then determined and the core average axial shape index is conservatively adjusted to account for the rod shadowing effects.

4.1.3 Uncertainties

Uncertainties, F_u, must be applied to all measurements, calculations, and allowances in a conservative manner such that a reactor trip will occur before any core safety limit is exceeded. This assures that the

specified acceptable fuel design limits will not be exceeded. Typical uncertainties and allowances that may apply to the LSSS trips are listed below:

• Augmentation Factors	(See Table 4.1)
• Physics calculation measurement uncertainty	
• Peak LHGR	7%
• F_r	6%
• Azimuthal tilt allowance	3%
• Engineering tolerance uncertainty	3%
• Power measurement	2% of rated for LCO 5% of rated for LSSS
• Trip overshoot	5% of rated
• Physics uncertainty in Predicting	±.02 axial shape
• CEA distribution effect on excore detectors	index units
• Physics uncertainty in applying shape annealing correction axial shape index limits	±.01 axial shape index units
• Excore detector subchannel calibration using incore detectors	±.01 axial shape index units
• Trip system processing	±.02 axial shape index units

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4.2 THERMAL MARGIN/LOW PRESSURE (TM/LP) LSSS

The TM/LP trip is designed to shut the reactor down when the core pressure falls below calculated thermal margin limit lines determined by inlet temperature, core power and pressure. The thermal-hydraulic design criteria are based to avoid departure from nucleate boiling (DNB). These concerns can be summarized as follows:

(1) The departure from nucleate boiling ratio (DNBR) based on an approved correlation (Reference 1) must be greater than or equal to an approved safety limit, which ensures the integrity of the fuel and that the SAFDL's are not violated.

(2) The thermal-hydraulic conditions of the reactor core must be within the limiting range of the empirical correlation. For example, the W-3 DNB correlation is only valid if local quality is less than 15%.

(3) The bulk coolant temperature at the exit of the core must be less than the saturation temperature.

(4) The coolant void fraction in the limiting coolant channel in the core can not exceed the flow stability limits (Reference 2).

The TM/LP trip is provided to prevent operation from exceeding the above safety limits including allowance for measurement error and uncertainties. A typcial TM/LP LSSS is shown in Figure 4.5 which graphically defines the limiting values of reactor coolant pressure, core

inlet temperature, and reactor power level. The low set point of 1,750 psia will always trip the reactor. The continuous TM/LP LSSS trip function is determined according to the following formula:

$$P_{var} = \alpha * PF(B) * B + \beta * T_{IN} + \gamma$$

where

P_{var} = TM/LP trip (psia)

B = High auctioneered thermal (ΔT) or nuclear power, in % of rated power

$PF(B)$ = A core peaking function that defines the variation of overpower with respect to core power

T_{IN} = Core inlet temperature (^{o}F)

α = the change in primary pressure needed to maintain a given margin to an allowable DNBR for a given change in core power at a constant inlet temperature and over-power margin

β = $\dfrac{P_{var_1} - P_{var_2}}{T_{IN_1} - T_{IN_2}}$ = the change in primary pressure needed to maintain a given margin to an allowable DNBR for a given change in inlet temperature at a constant core power and overpower margin.

γ = A pressure bias term used to adjust the calculated P_{var} to account for system uncertainties and measurement errors.

The TM/LP trip effectively monitors all those NSSS parameters, except mass coolant flow rate and core peaking which affect the thermal-hydraulic safety limits. The calculated P_{var} is then compared to the measured pressure, and a trip signal is generated when P_{var} approaches measured pressure with a specific uncertainty. The APD analysis defines the limiting axial shapes that are to be considered in the TM/LP trip determination. Therefore, the effect of power distribution on TM/LP trip is implicitly considered.

At each thermal-hydraulic safety limit considered, thermal margin limit lines are generated over a wide range of pressure, inlet temperature and power. The variation of these thermal limit Loci with DNB overpower is incorporated in the TM/LP trip by the PF(B) function as indicated above. A procedure to obtain TM/LP LSSS trips using ENC methodology is described in Section 5.2.

4.2.1 Typical Uncertainties

The typical uncertainties associated with the TM/LP trips are described below:

Measurement Errors	Magnitude
(1) Pressure measurement	±22 psia
(2) Coolant inlet temperature	±2°F
Operating Allowances	
(1) Instrument processing error	±5 psia

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- (2) Trip unit allowance for most rapid depressurization transient ± 20 psia
- <u>Power Distribution Uncertainties</u>

  All items in Section 3.1.2 apply except the augmentation factors which are implemented only for the evaluation of fuel centerline melt.

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4.3 VARIABLE HIGH POWER

This trip is designed to protect the core at all power levels against transient initiated core power increases.

Calculations show that CEA insertion causes changes in nuclear peaking throughout the core. In order to avoid excessive peaking due to CEA insertion and exceeding core power limits, control rod insertion during reactor operation is limited by the Power Dependent Insertion Limit (PDIL). This limit assures that the nuclear peaking will be within acceptable values at any power level by limiting control rod insertion as a function of reactor power. The variable nuclear overpower trip together with PDIL assures that the peaking limits at specified power levels will not be violated during operation at the power dependent insertion limits and the allowed nuclear overpower value.

The setpoint calculations for axial shape index and TM/LP are made with the bounding values of nuclear peaking at the power levels which will result in a nuclear overpower trip. These calculations are performed at maximum allowed power levels consistent with the PDIL and variable overpower trip system. The nuclear overpower trip in conjunction with the TM/LP and APD trips prevent the specified acceptable fuel design limits (SAFDL) from being exceeded during a power excursion in any rodded or unrodded core configuration.

Table 4.1 Typical Fuel Augmentation Factors
Reference 14*

Core Height (%)	Core Height (inches)	Non-Collapsed Clad Augmentation Factor
98.5	134.7	1.057
86.6	118.6	1.051
77.9	106.5	1.047
66.2	90.5	1.041
54.4	74.4	1.035
45.6	62.3	1.030
33.8	46.2	1.024
22.1	30.2	1.017
13.2	18.1	1.011
1.5	2.0	1.001

* Augmentation factor for evaluation of fuel centerline melt only.

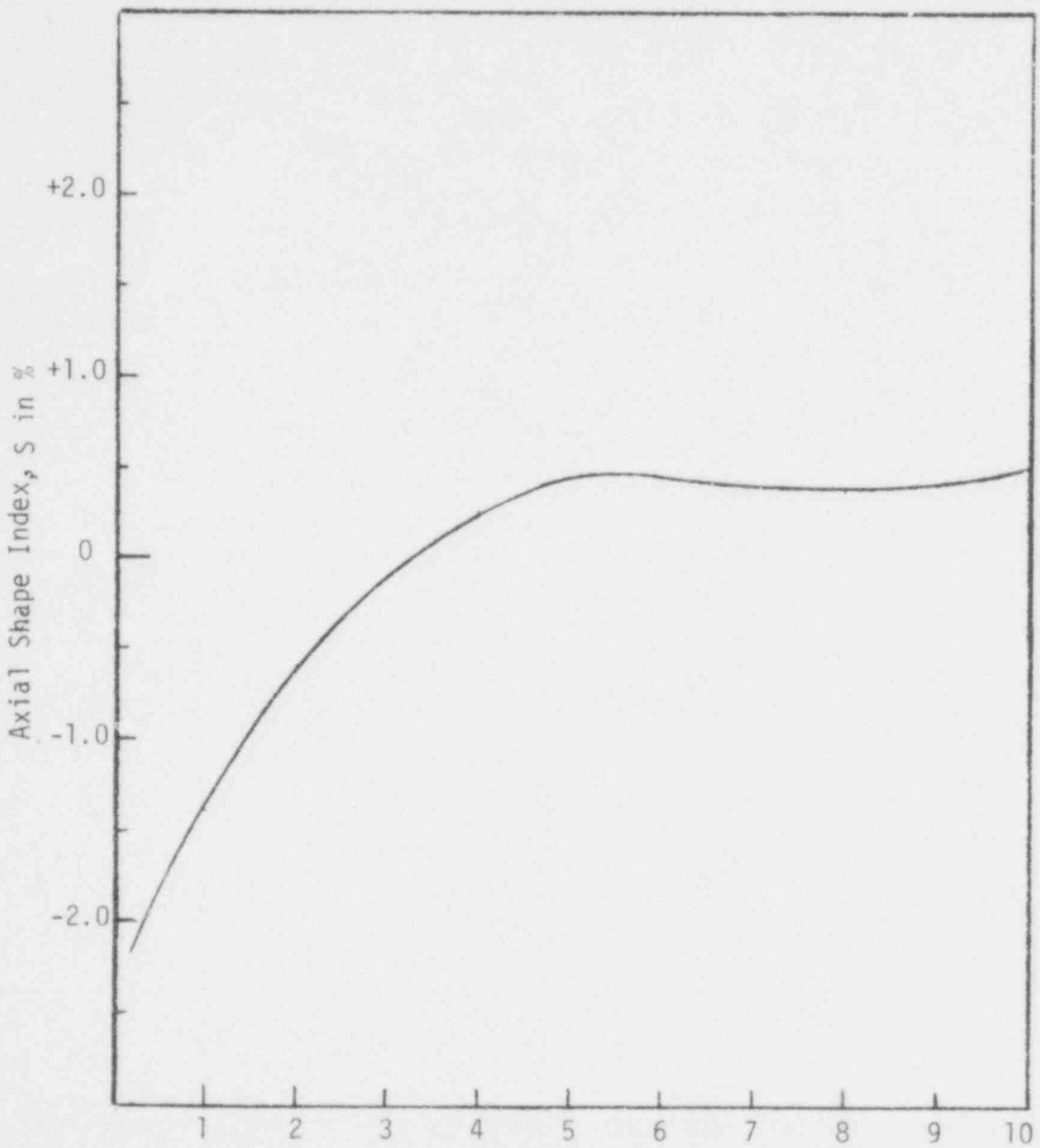

Figure 4.1 Axial Shape Index vs Cycle Exposure
Fort Calhoun Cycle 4,
Calculated D ta, ARO

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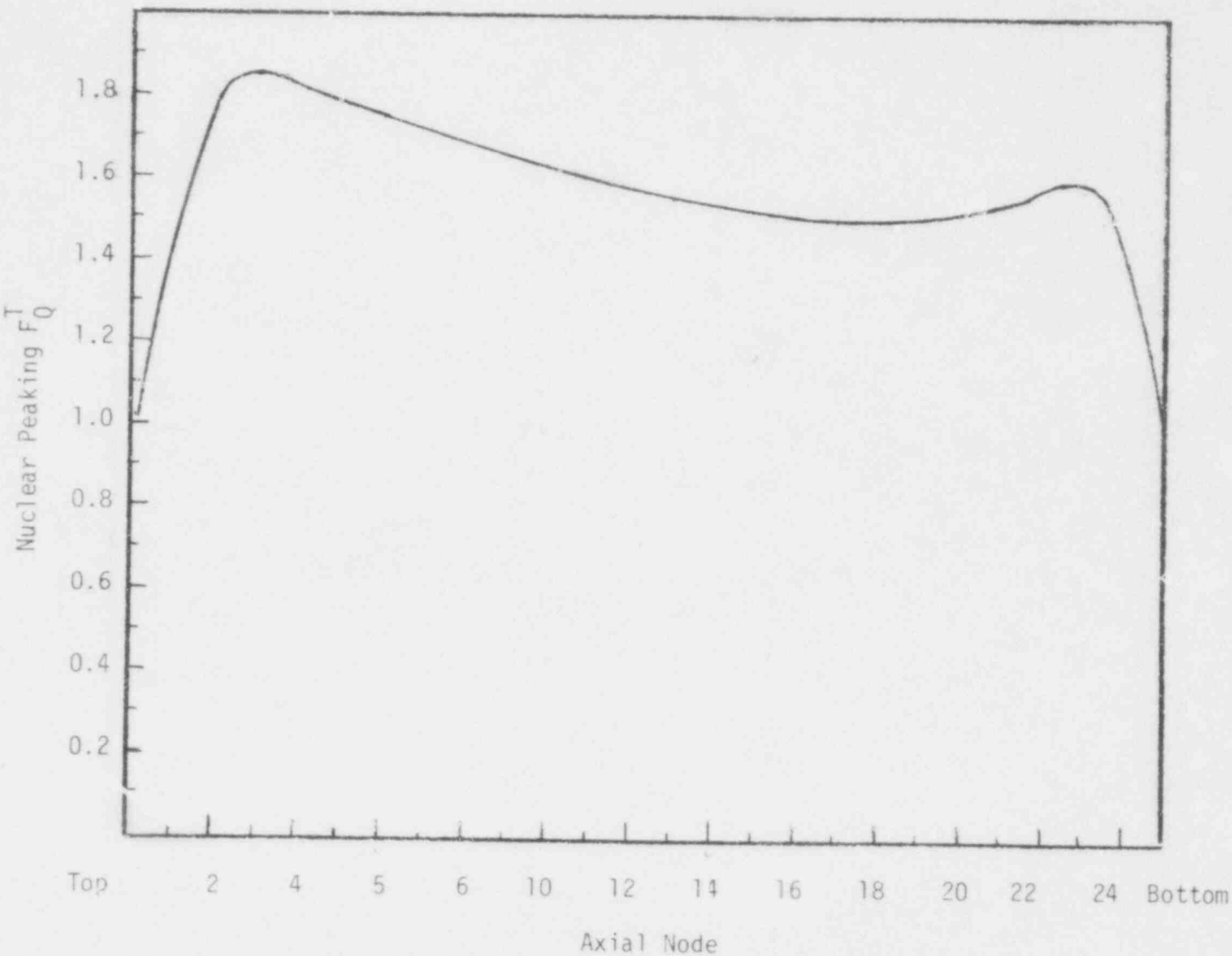

Figure 4.2 Nuclear Peaking F_Q^T, vs Axial Node

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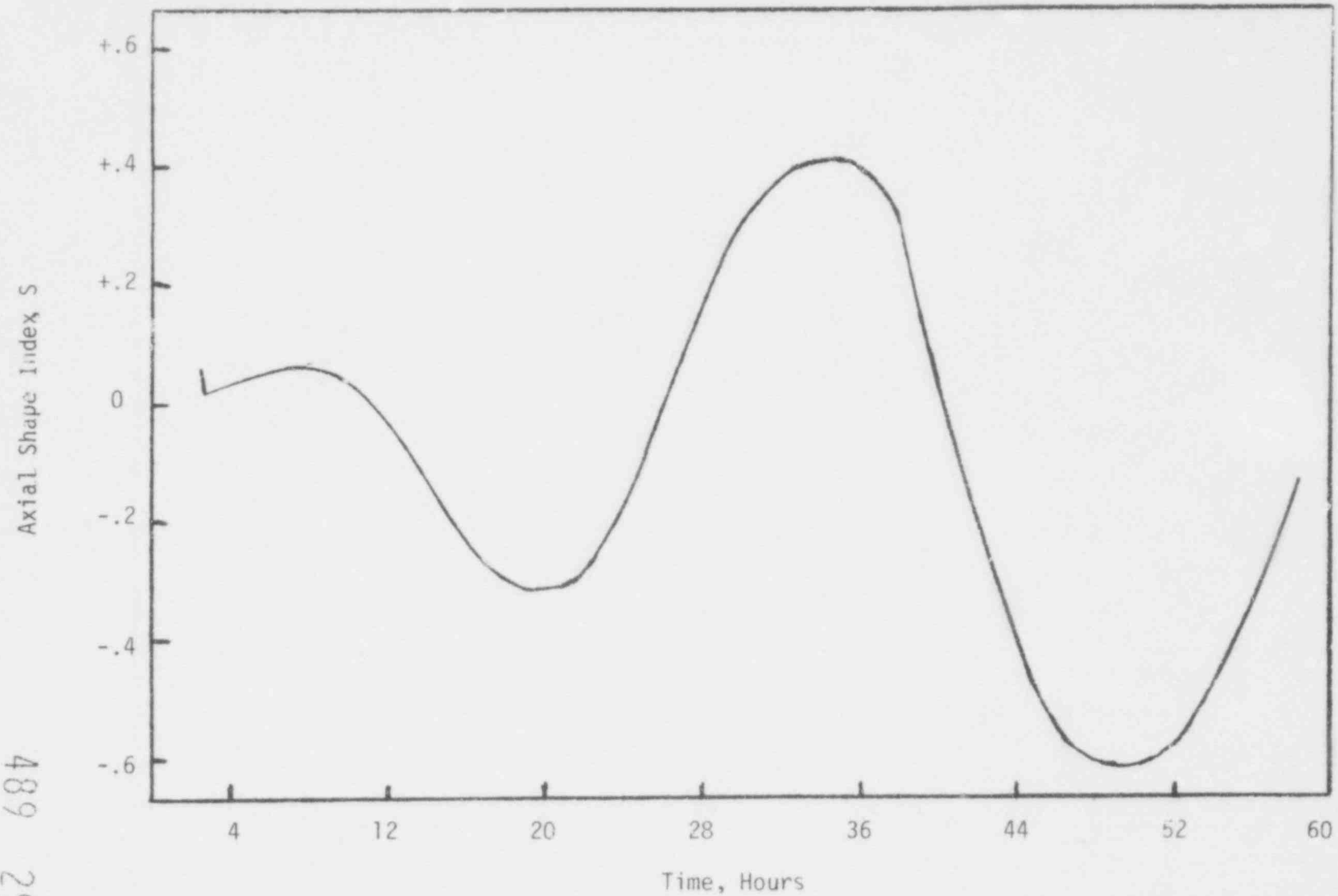

Figure 4.3 Axial Shape Index, Caused By Xenon Oscillation vs Time (Hours)

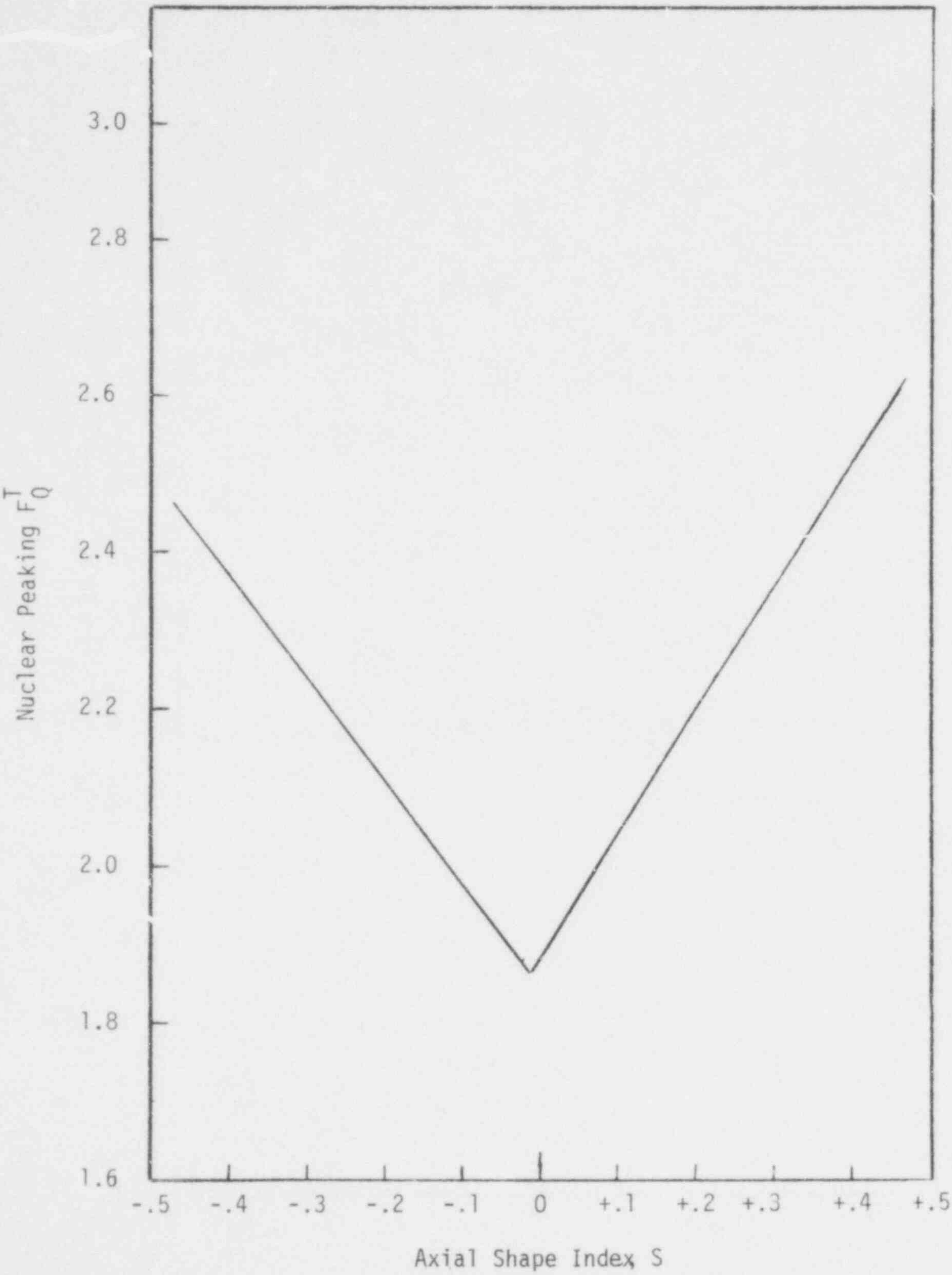

Figure 4.4 Axial Shape Index, S, vs Nuclear Peaking F_Q^T

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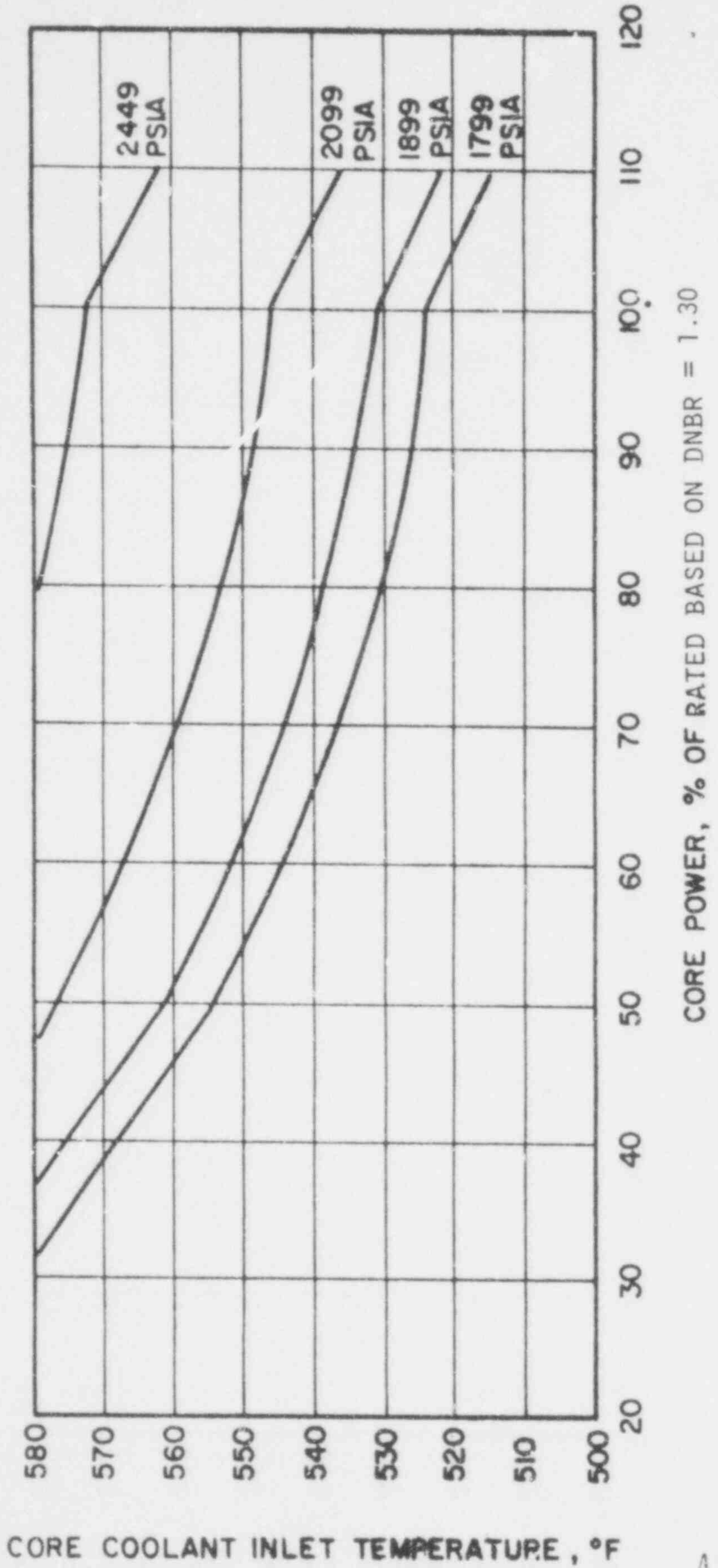

Figure 4.5 TM/LP LSSS

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5.0 LIMITING CONDITIONS FOR OPERATION (LCO)

The Limiting Safety System Settings, (LSSS) are setpoints input into the RPS that cause a reactor trip if the corresponding monitored parameters equal the input values protecting the reactor core from exceeding any Specified Acceptable Fuel Design Limit (SAFDL). SAFDL's must not be violated during those anticipated operational occurances which are expected to occur one or more times during the life of the plant. A list of the transients to be investigated are shown in Table 5.1. All the anticipated operational occurances except the CEA drop and loss of flow are terminated by the RPS. Therefore, only the CEA drop and loss-of-flow have to be considered for the determination of the LCO's. The LCO limit resulting from the LOCA analysis is monitored by the incore detectors. A backup monitoring envelope is provided in the Fort Calhoun Technical Specifications in the event that the incore detectors fail.

During normal operation the peak linear heat rate is monitored through the use of the incore detection system. The peak LHGR is maintained to be below the values which are calculated to result in fuel centerline melt during a CEA drop or a peak clad temperature of 2200^{o}F during the postulated loss of coolant accident. In addition, penetration of DNB limits during steady state operation and anticipated transients including the CEA drop is precluded through the implementaton of the LCO for DNB monitoring. This is accomplished by determining the axial shape index during operation through the excore detectors and

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comparing this value against the allowable shape index as a function of core power. Analysis of the most limiting transients is performed with a variety of axial power shapes. The results of the analysis are reduced to provide allowable core power as a function of axial shape index, (ASI) in a fashion consistent with that presented in Section 4.1.1.

In the event that the incore detectors are not in operation for an extended period of time, the peak linear heat rate will be monitored through the use of a linear heat rate LCO. This LCO is determined in a manner similar to the APD limiting safety system setting (Reference Section 4.1.1) except that the allowable core power as a function of ASI is determined through analysis of the CEA drop and the postulated loss of coolant accident.

The most limiting transient is of primary interest in determining the axial shape index LCO limits. An example of generation of transient axial shape index LCO's expected to be most limiting (CEA drop incident) is described in Section 5.1. In Section 5.2 the axial shape index LCO that implements the LOCA limits is described.

5.1 CEA DROP AXIAL SHAPE INDEX LIMITS

The CEA drop incident is defined as the inadvertent release of a CEA causing it to drop into the reactor core. The absence of a turbine runback following a CEA drop at the EOC boron condition will tend to restore the reactor to near full power with an adversely distorted power distribution. Therefore, it is necessary to maintain the linear heat rate within limits to assure that the SAFDL's are not exceeded during the transient. If the allowable linear heat rate for the CEA drop

incident is limiting, the increase in the three-dimensional power peaking due to the CEA drop must be determined. The APD centerline melt envelope is then reduced by that amount corresponding to that increase in power to that increase in power peaking calculated to result from a CEA drop. The most adverse nuclear peaking and axial power distributions are used in the calculations.

The LCO's are more restrictive than the LSSS, and the plant operation is administratively maintained within the LCO limits. Figure 5.1 shows a typical LCO "barn" for protection of LHR during a CEA drop.

5.2 LOSS OF COOLANT ACCIDENT (LOCA) LCO LIMITS

The consequence of a loss of coolant accident is evaluated in accordance with the Acceptance Criteria as presented in 10CFR 50.46. These criteria maintain that the rated maximum linear heat generation shall be such that:

1) the calculated peak fuel element clad temperature does not exceed 2200^{o}F.
2) the amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total zircaloy associated with the active fuel rod length in the reactor.
3) the cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling and the hot fuel rod cladding local oxidation does not exceed 17%.

4) the system long-term cooling capabilities provided for previous cores remain applicable for ENC fuel.

The allowable linear heat generation rate which satisfies the 10CFR 50 46 is evaluated using F_{xy} and F_z pairs identified in the "Fly Speck" analyses which correlates power peakings and axial shape index. The allowable Linear Heat Generation Rate, (LHGR) or F_Q^T, for the core is thus evaluated as a function of axial peaking. The exposure dependence of the F_Q^T is also evaluated and is superimposed on the F_Q^T versus axial position curve as required.

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Table 5.1 Incidents Considered in Transient and Accident Analysis

Anticipated Operational Occurences for which the RPS Assures no Violation of SAFDLs:

Control Element Assembly Withdrawal

Boron Dilution

Startup of an Inactive Reactor Coolant Pump

Excess Load

Loss of Load of Feedwater Flow

Excess Heat Removal due to Feedwater Malfunction

Reactor Coolant System Depressurization

Loss of Coolant Flow[1]

Loss of AC Power

Anticipated Operational Occurrences which are Dependent on Initial Overpower Margin for Protection Against Violation of SAFDLs:

Loss of Coolant Flow[1]

Loss of AC Power

Full Length CEA Drop

Part Length CEA Drop

Part Length CEA Malpositioning

Transients Resulting from Malfunction of One Steam Generator

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1 Requires Low Flow Trip

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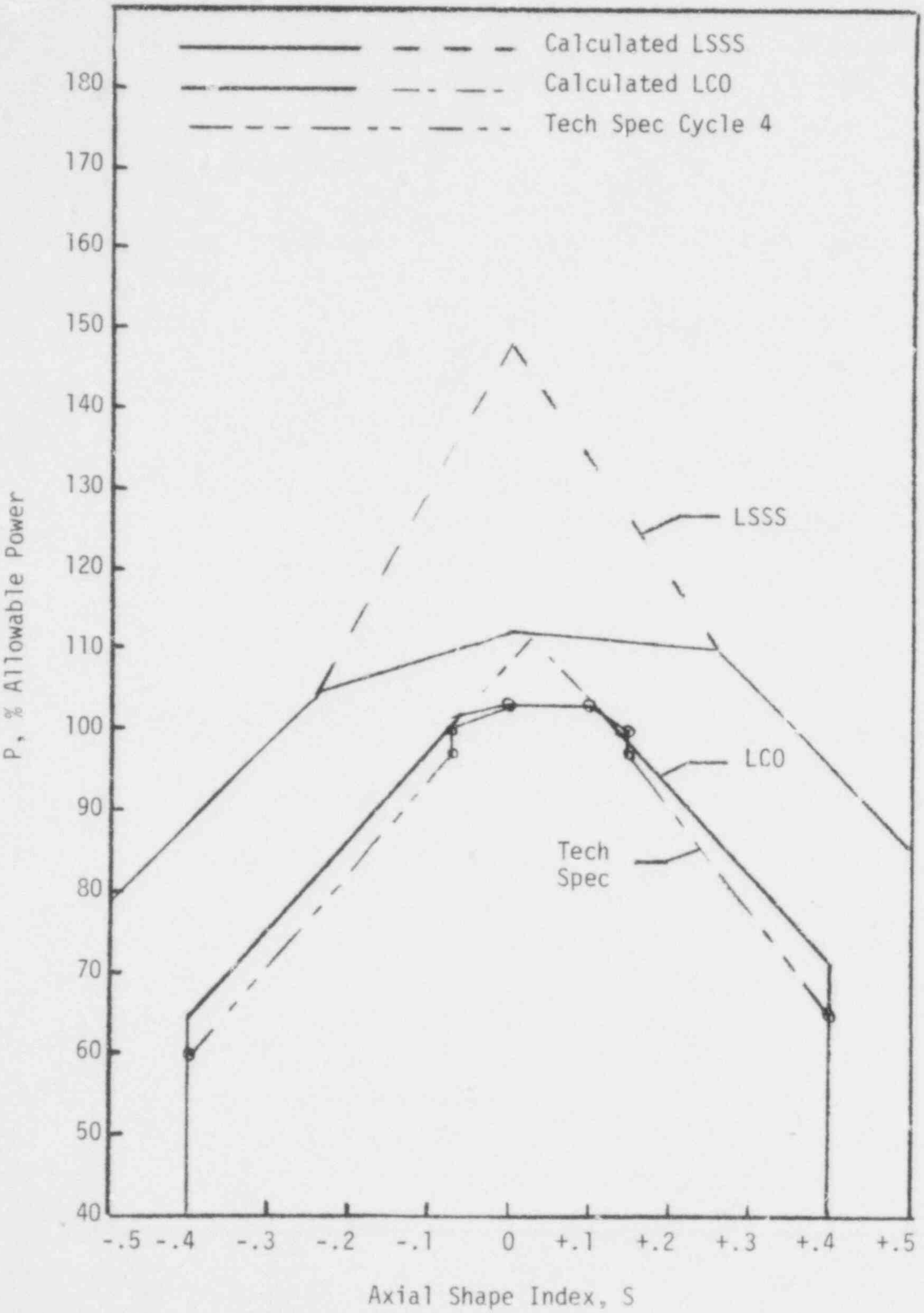

Figure 5.1 Axial Shape Index "Band" for Centerline Melt LSSS at 21 kw/ft and Axial Shape Index "Barn" LCO Calculation at 15.5 kw/ft, Fort Calhoun, Cycle 4

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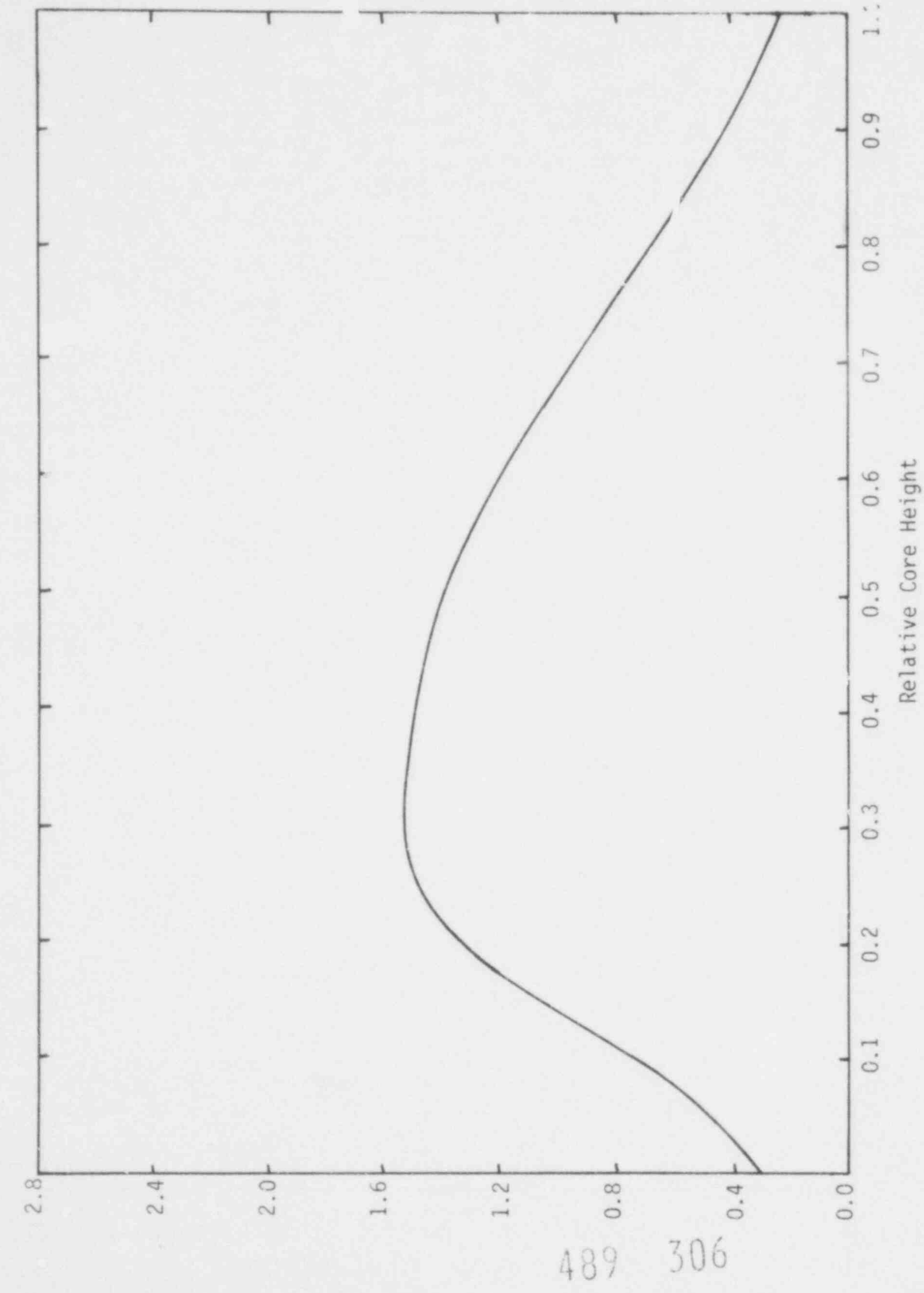

Figure 5.2 A Typical Hot Channel Axial Relative Power Distribution Used in Transient Analyses

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6.0 ENC METHODOLOGY

Standard ENC methods and calculational tools are employed to calculate the limiting safety system settings and limiting conditions of operation for the CE reactor protection system. The analysis requires extensive calculations in the Neutronics, Thermal Hydraulics and Nuclear Safety areas. The plant transients are evaluated through Plant Transient Simulation (PTS) calculations. The following is a general description of ENC methodology employed for determining setpoints for CE reactors.

6.1 NEUTRONICS

The methods and the computer programs used to determine LSSS and LCO setpoints for CE reactors are the standard ENC (NRC approved) methods which are described in References 7, 8, and 9. The neutronic computer programs used to do the analysis consist of the cross section generation code XPOSE, (Reference 10), the pin by pin diffusion theory code PDQ7/HARMONY (References 11 and 12), and the three-dimensional reactor simulator code XTG, (Reference 13). A synthesis code CESPT was written specifically to process data calculated with the PDQ and XTG core models.

6.1.1 Cross Sections, XPOSE

Cross sections for the reactor simulator codes, PDQ7 and XTG, are calculated with the ENC computer code XPOSE. The XPOSE code is a modified version of the industry accepted LEOPARD code. It is used to generate fast and thermal neutron spectra and cross sections. Microscopic cross sections are generated by XPOSE for PDQ7/HARMONY and

macroscopic cross sections are determined by XPOSE for XTG. Non fuel region macroscopic cross sections are determined by XPOSE and reflect the spectrum effects of the surrounding fuel.

6.1.2 Simulator PDQ

Detailed radial calculations of the core are performed with computer code PDQ7/HARMONY in two-dimensions. The core is modeled in PDQ7 on a pin-cell basis; i.e., one mesh block per fuel cell. Each pin cell has the appropriate nuclide concentration of the burnup history for that pin. The PDQ7 model is similar to the model described in Reference 7. Output from the PDQ7/HARMONY calculations include the pin-by-pin radial power distributions. These values are used as input to XTG for the F_{xy} calculation.

6.1.3 Simulator XTG

3DXTG

The reactor core is modeled and depleted in three-dimensions with the reactor simulator code XTG. As mentioned above, cross sections for XTG are generated with PDQ7 and XPOSE. Each assembly in XTG is represented by four radial nodes and twelve axial nodes. Control rod cross sections and assembly local peaking factors are input into the code by assembly. The computer code XTG calculates radial and axial power distributions and determines F_{xy}, the ratio of the hot pin to average pin on a planar basis, i.e., with 12 axial nodes, 12 axial F_{xy} values are determined. Figure 6.1 shows the axial power distribution for Fort Calhoun Cycle 4 at about 500 MWD/MT. Other features of XTG used

- Integral control rod worth calculations, Figure 6.2
- Control rod bank worths
- Core average cross sections by axial node
- Core average exposure by axial node, Figure 6.3.

1-D XTG

Sets of axial power distributions and axial shape indices are calculated with a one-dimensional model of XTG using the axial dependent core average cross sections and exposures determined in the 3-D XTG cycle depletion described above. The reactor core is modeled in one-dimension using one radial node and 24 axial nodes. The 24 node axial exposure distribution is determined from the 12 node 3-D XTG exposure distribution, (Reference Figure 6.3). The integral rod worths are also calculated with the one-dimensional model. A comparison of the Fort Calhoun Cycle 4 integral Rod worth measurements to the 3-D and 1-D XTG integral rod worth calculations are shown in Figure 6.2. The agreement between the measured and calculated integral rod worth is very good.

Using the above inputs, the 1-D XTG procedure to determine ordered pairs of axial power and symmetric offset is:

1. Incite a xenon oscillation at HFP, all control rods out, and with or without thermal hydraulic feedback depending if the calculations are at EOC or BOC, respectively. In the 1-D XTG calculation a xenon

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oscillation can be incited by partially inserting control rods for several hours and the "instantly" removing the control rods. Feedbacks are used in all cases except during the initial xenon oscillation calculation at the BOC.

2. Select 30 to 40 different xenon distributions from Step 1 and rerun several static 1-D XTG calculations with power levels corresponding to the nuclear overpower trip setpoint and control rods at the CEA power dependent insertion limits. Use thermal hydraulic feedback.
3. Determine the axial shape index symmetric offset, S, and axial power distribution, F_z, for the various power levels and corresponding PDIL limits in order to determine the percent allowable power. The percent allowable power is determined with the synthesis computer code, CESPT.

6.1.4 Axial Shape Index Setpoint Code, CESPT

The computer code CESPT, is used to determine the ordered pairs of P, % allowable power, and Sp, peripheral axial shape index. Input for CESPT is taken from the previous calculations made with PDQ7, 3-D XTG and 1-D XTG. CESPT, then, determines the minimum allowable power for each axial power distribution determined with the 1-D XTG calculation described above. The input to CESPT is conservative. Values of F_{xy} are input into the code by axial position. The core is divided into axial

position. The core is divided into axial zones depending on control rod insertion, (each zone has the identical control rod configuration extending throughout the zone). Values for F_{xy} are extracted from PDQ7 and 3-D XTG and input into the corresponding axial position and zone in the synthesis code. A description of the "zone" modeling is shown in Figure 6.4.

The axial power distribution, F_z, determined with the 1-D XTG model are input by axial heigh in CESPT. The fuel augmentation factor, F_{aug}, is also input by axial position. The combination of F_{xy}, and F_{aug} with F_z gives values for the F_Q as follows:

$$F_Q = F_{xy} \times F_z \times F_{aug}$$

without uncertainties

or

$$F_Q^T = F_Q \times F_u^1$$

with uncerta .ties

where

F_{xy} = ratio of hot pin to average pin at core evaluation z

F_z = axial power peaking factor

F_u^1 = uncertainties

F_Q^N = $F_z \times F_{xy}$ = hot spot in core

F_Q^T = $F_u^1 \times F_{aug}$

The percent allowable power is therefore calculated to be:

$$P = \%\ \text{allowable power}, = 100 \times k/F_Q^T \times W_{aug}$$

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where

W_{avg} = average core LHGR for 100% power

k = kw/ft for centerline melt

(a typical value is 21 kw/ft)

The axial power profiles used above are the core average values which give an axial shape index, S, for the overall core. As stated in Section 4.1.2, the core average axial shape index, S, must be modified by an adjustment Factor, F, to account for rod shadowing effects on the excore detectors. The rod shadowing effects are directly applied to S, in the above calculation giving (%power, Sp). These are the sets of ordered pairs used in the setpoint analysis. Figure 6.5 shows typical fly-speck values of the percent allowable power, P, as a function of symmetrical offset for Fort Calhoun Cycle 4, without uncertainties, F_u. Figure 6.6 applied to the axial shape index limiting safety system setting for Fort Calhoun centerline melt at 21 kw/ft. The application of uncertainties to the axial shape index points shown in Figure 6.5 is shown in Figure 6.6. The uncertainties applied to the setpoint analysis for axial shape index centerline melt are discussed in Section 4.1.3.

The above neutronic discussions deal with the methodology used to determine the ordered pairs of symmetric offset vs % allowable power (S,P) for Fort Calhoun. In the course of doing the calculations the axial power profile and peaking limits are determined which are applied to the TM/LP calculation for setpoints.

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6.2 THERMAL-HYDRAULIC

Methods and programs used to determine thermal margin and justify the current ENC DNB correlation for PWR's are described in XN-75-48 (Reference 4) and XN-75-21 (Reference 3). The criteria established to assure that the thermal-hydraulic limits are not exceeded are listed in Section 4.2. The ENC thermal-hydraulic setpoint methodology and related computer codes are described below.

6.2.1 XCOBRA-IIIC Code

XCOBRA-IIIC (Reference 3) is an ENC modified version of COBRA-IIIC computer code (Reference 6). The XCOBRA-IIIC code provides both transient and steady-state calculation capabilities while including the effect of cross flow mixing between fuel assemblies and subchannels. The thermal-hydraulic parameters, such as DNBR, local quality and void fraction, are calculated for each fuel node. The degree of core-wide nodalization and the modeling options available in the code provide calculational flexibility. This code is used for thermal-hydraulic parameter evaluation to generate TM/LP LSSS.

6.2.2 PTS-PWR Code

The PTS-PWR code (Reference 5) is an ENC digital computer program developed to describe the behavior of pressurized water reactors subjected to abnormal operating conditions. The model is based on the solution of the basic transient conservatism equations for the primary and secondary coolant systems of the transient conduction equation for the fuel rods, and of the point kinetics equation for the core neutronics. The program calculates fluid conditions, such as flow, pressure,

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quality, heat flux, DNBP, reactor power, and reactivity during the transient. This code is used for thermal-hydraulic safety limits evaluation to generate LCO's for AOO transients or simply used for checking the behavior of the reactor with existing setpoints during a specified transient.

6.2.3 TM/LP LSSS

The TM/LP trip protects the reactor core from exceeding SAFDL when the core pressure coolant inlet temperature and power deviate from normal. The procedure used to generate TM/LP trips with the XCOBRA-IIIC models is outlined below.

(1) Select a power. The APD offset trip defines the limiting axial shapes that are to be considered in the TM/LP trip.

(2) Select a reactor pressure.

(3) Select a coolant inlet temperature.

(4) Choose the most adverse power distribution based on procedure (1).

(5) Set up an XCOBRA-IIIC model implementing (2), (3) and (4) as input data.

(6) Compare the XCOBRA-IIIC results with the thermal-hydrualic safety limits described in Section 4.2. Return to step (3) adjusting T_{IN} and repeat the process until the results are equal to the T-H limits.

(7) Select another power pressure and repeat the process and so forth.

(8) Construct a set of TM/LP LSSS curves as shown in Figure 5.1.

(9) Based on the finalized TM/LP LSSS curves, one can construct all the constants and PF(B) functions using steps (10) to (12).

(10) Select a reference power (say 100% rated). Constant β can be calculated by computing the following formula.

$$\beta = \frac{P_{var_1} - P_{var_2}}{T_{IN_1} - T_{IN_2}}$$

where, subscripts 1 and 2 represent two XCOBRA-IIIC calculations at the same power level but different T_{IN}.

(11) Normalize PF(B)=1 at B=100 (%) rated power, calculate α and γ.

(12) Then evaluate PF(B) by changing power levels which defines the limiting axial power shapes from APD trips.

The TM/LP curves generated in this manner represent a conservative envelope of the thermal-hydraulic safety limits for all the possible operating pressures at specific power levels and coolant inlet conditions. The TM/LP setpoint will trip the reactor down when the function falls below calculated P_{var}.

6.2.4 Excore Axial Shape Index Monitoring For LCO

The ENC procedure to generate the thermal-hydraulic limits excore ASI monitoring for LCO's using the PTSPWR and the XCOBRA-IIIC models is outlined below.

(1) Use PTSPWR model to evaluate the most limiting Anticipated Operational Occurance, (AOO) transient in order to decide

required overpower margin. All the PTSPWR transient calculations are initiated from steady state conditions

(2) Use the XCOBRA-IIIC model to generate the limiting operating conditions and justify that these operating conditions can be protected during any AOO transient.

(3) Apply similar procedures as that of APD LSSS. An APD LCO limit curve can be obtained. These LCO's provide assurance that the SAFDL will not be violated at those AOO's which are dependent on initial overpower margin for protection.

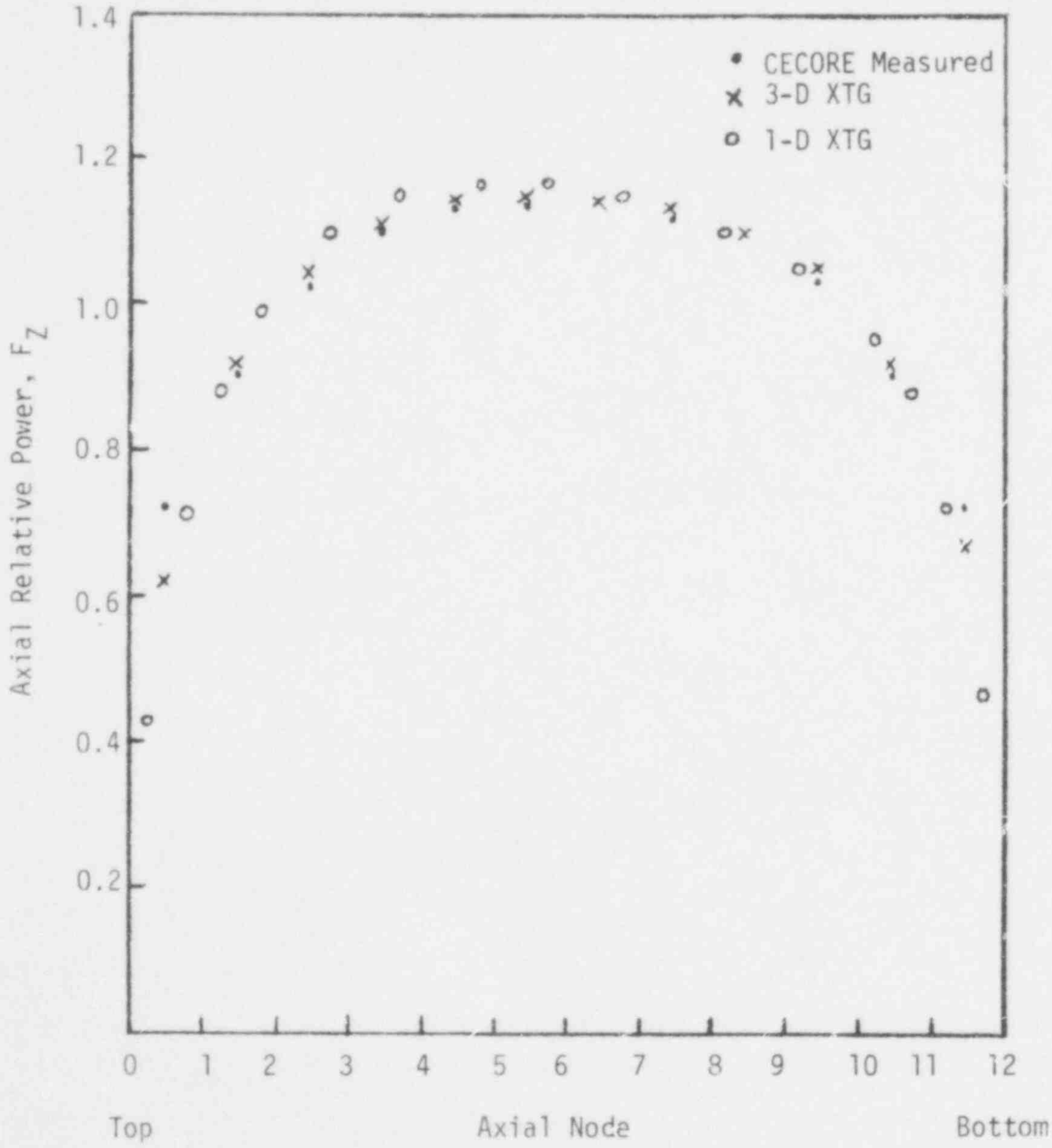

Figure 6.1 Measured and Calculated Axial Power Distributions For Calhoun Cycle 4, 500 MWD/MT

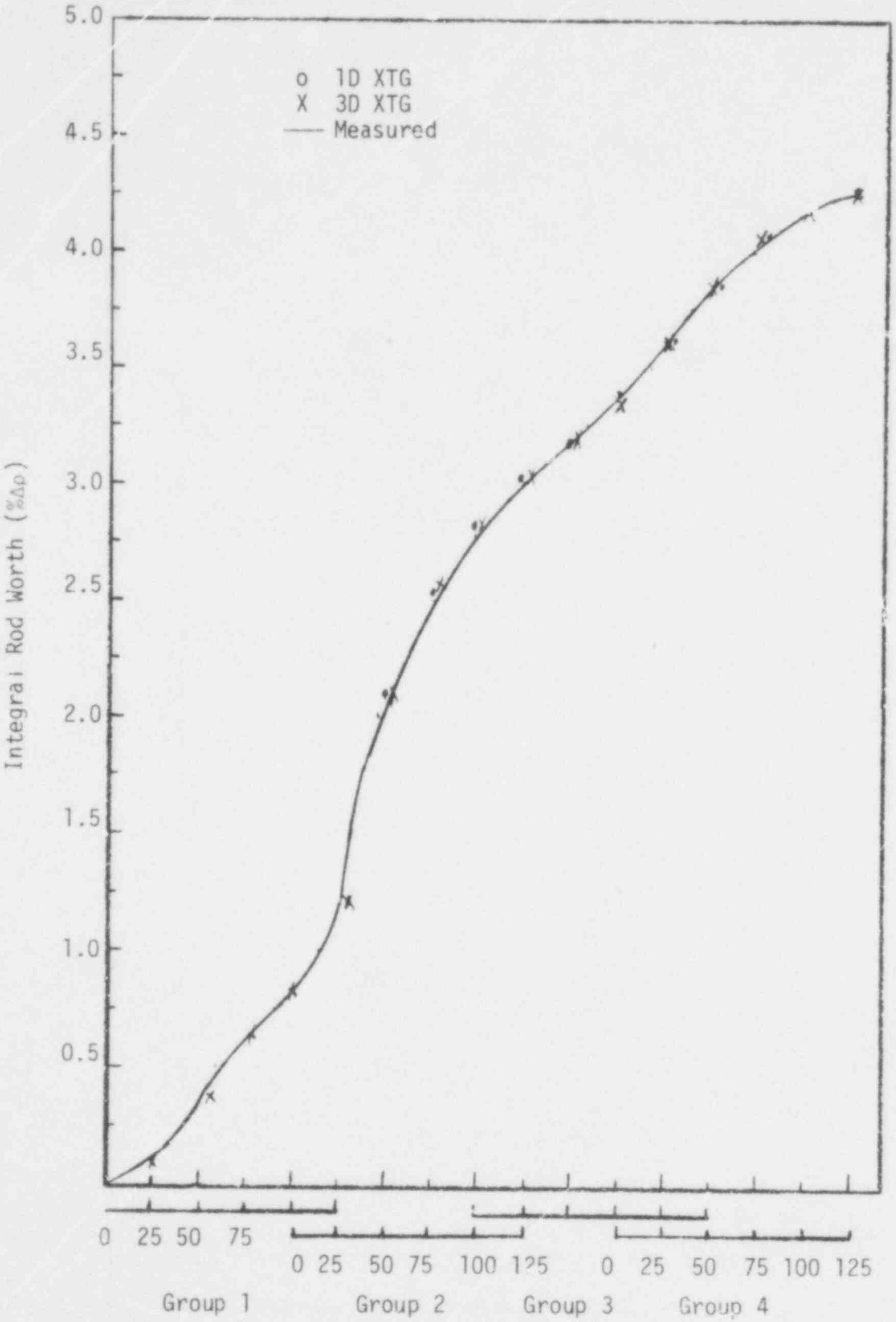

Figure 6.2 Integral Rod worth (Regulating CEA Groups 1, 2, 3, 4) vs Rod Group Position

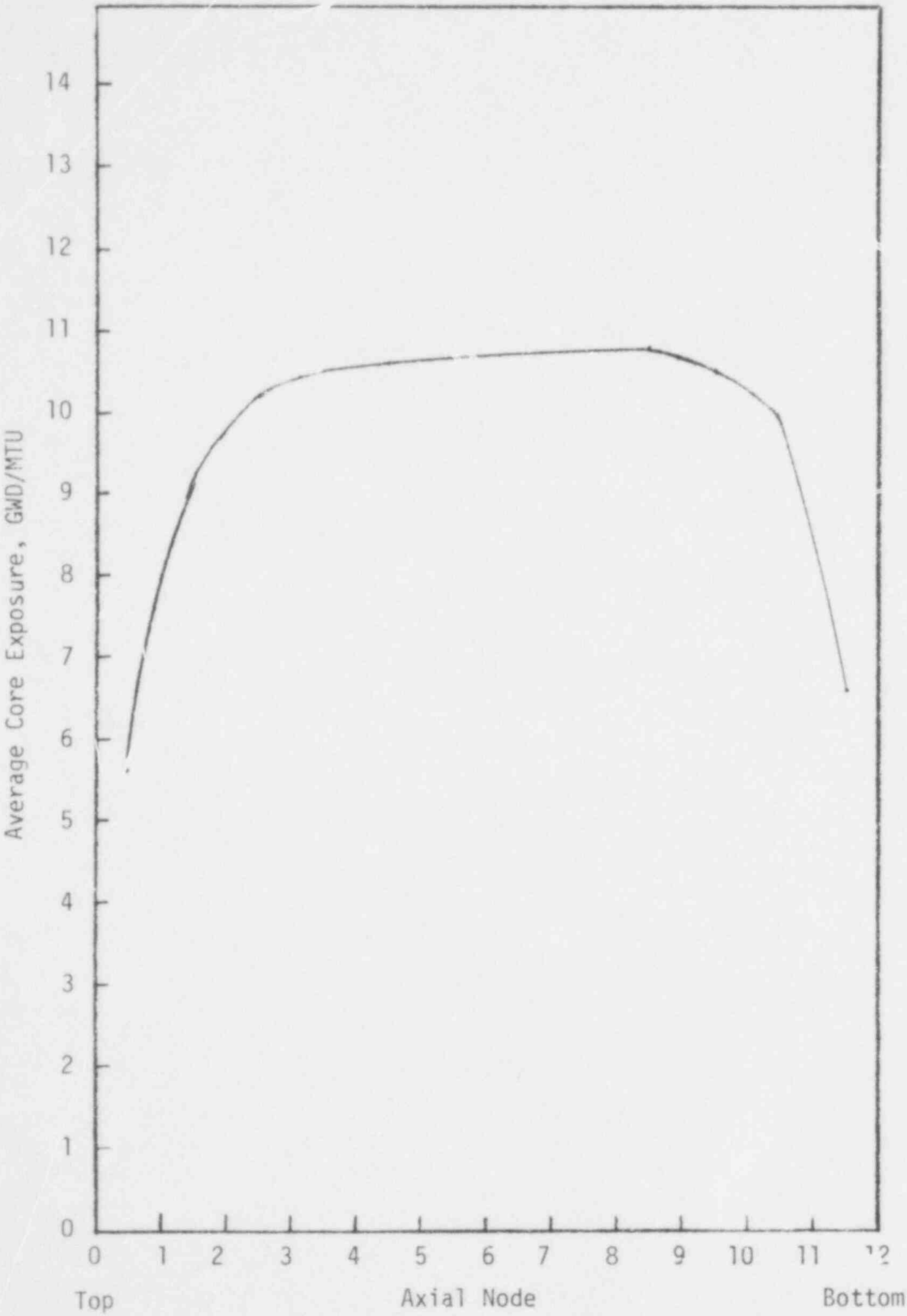

Figure 6.3 Fort Calhoun Cycle 4, Core Average Exposure Distribution, 1-D and 3-D XTG, 9,750 MWD/MTU Core Average Exposure

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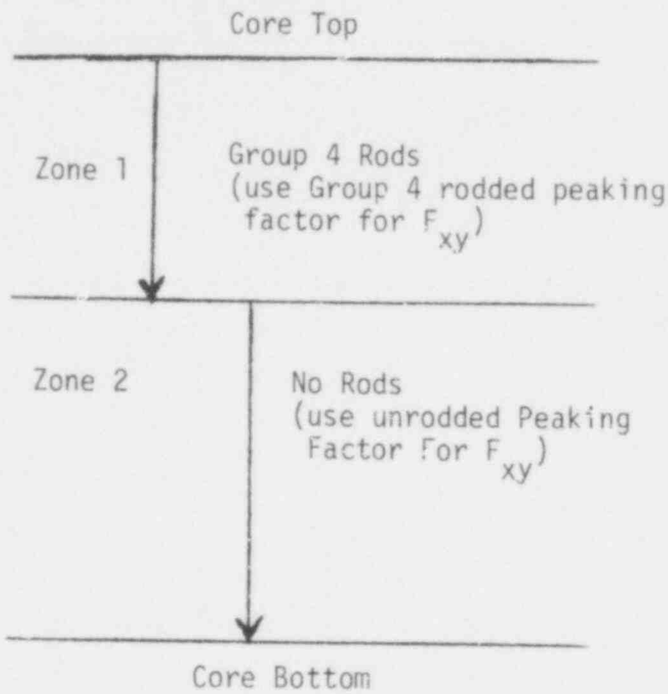

Figure 6.4 CEA Zoning Pattern For Input to CESPT

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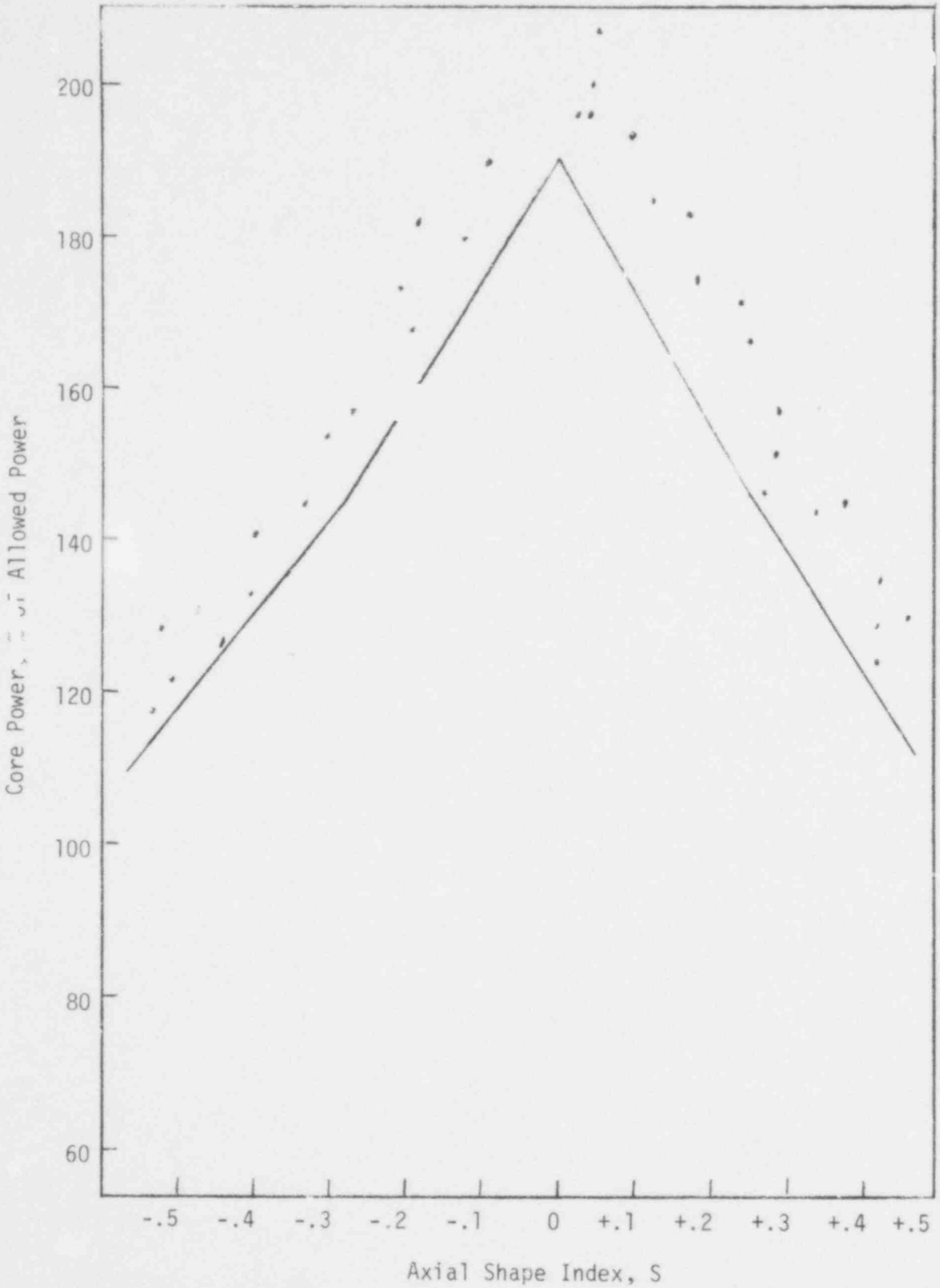

Figure 6.5 Axial Shape Index LSSS for Centerline Melt Without Uncertainties, Fort Calhoun, Cycle 4

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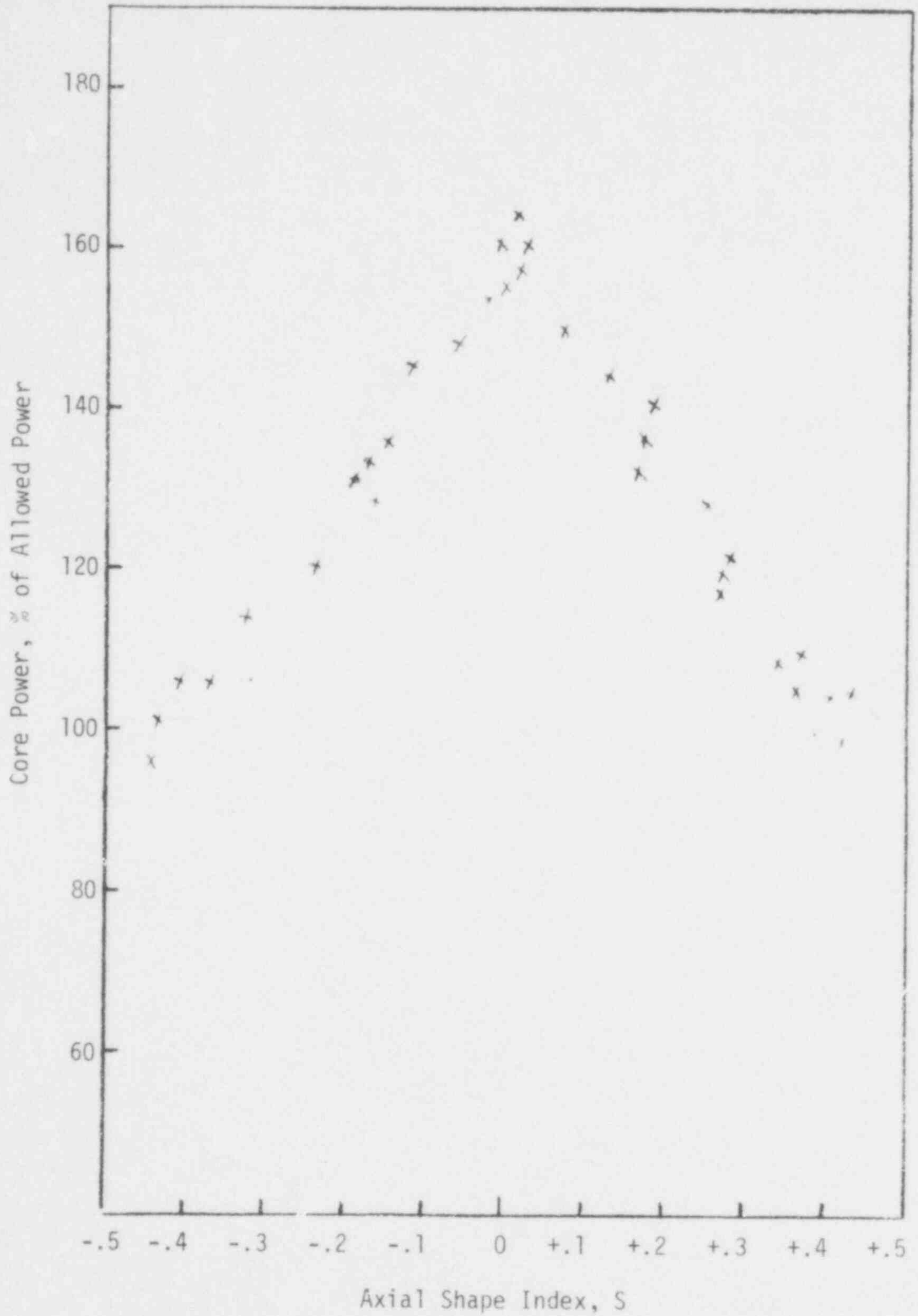

Figure 6.6 **Axial Shape Index LSSS for Fuel Centerline Melt With Uncertainties, Fort Calhoun, Cycle 4**

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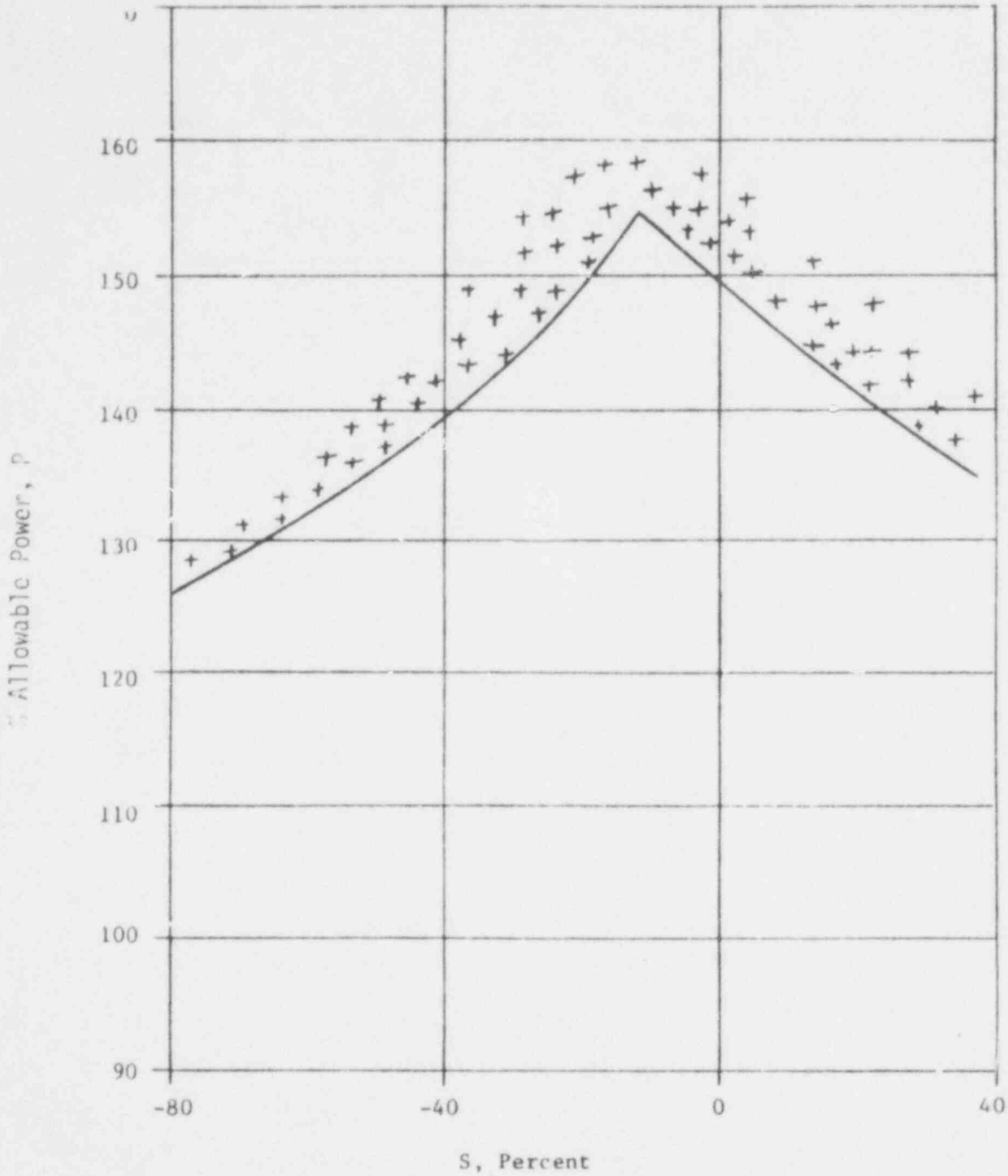

Figure 6.7 Typical DNBR % Allowable Power vs Axial Shape Index, S

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