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THE VEPCO NOMAD CODE AND MODEL

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ABSTRACT

The Virginia Electric and Power Company (Vepco) has developed NOMAD, a one-dimensional (axial), two energy group, diffusion theory computer code with thermal-hydraulic feedback, and a calculational model designated as the Vepco NOMAD model.

The model utilizes the Vepco computer codes NOMAD, XSEDT, XSFIT, XSEXP, NULIF, FXYZ, FDELH, and PCEDT. The model also uses data from the Vepco PDQ07 Discrete, PDQ07 One Zone, and FLAME models. The model is used to perform one-dimensional reactor physics analysis in support of reactor startup and cycle operation of the Vepco Surry and North Anna nuclear reactors. The accuracy of the NOMAD model is demonstrated through comparisons with other codes and with measurements taken at the Surry and North Anna Nuclear Power Stations.

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SECTION 1 - INTRODUCTION

The purposes of this report are to describe a reactor analysis computer code and model which were developed at Virginia Electric & Power Company (Vepco) and to demonstrate the accuracy of this model by comparing analytical results generated by the model to results from other codes and to actual measurements from Surry Units No. 1 and 2 and North Anna Units No. 1 and 2.

The code to be described is a one-dimensional (axial), two energy group, diffusion theory (with thermal-hydraulic feedback) computer code and is named the NOMAD code. The model to be described is designated as the Vepco NOMAD model. In addition to NOMAD, the model uses the Vepco computer codes XSEDT(1), XSFIT(1), XSEXP(1), FXYZ(2), FDELH(3), PCEDT(3), and NULIF(4). The model also utilizes data from the Vepco PDQ07 Discrete(5), PDQ07 One Zone(6), and FLAME(7) models. A detailed description of the input requirements, functioning, physical models, and output capabilities of these codes can be obtained from the referenced code manuals or reports.

The types of reactor physics calculations which can be performed within the general capabilities of the Vepco NOMAD model include:

1. Core average axial power and burnup distributions
2. Axial offset
3. Peaking factors ($F_Q(Z)$, $F_{XY}(Z)$, $F_Z(Z)$)
4. Final Acceptance Criteria (FAC) Analysis

5. Load follow maneuver simulation

6. Criticality searches on boron concentration, control rod bank position, core power level, or hot full power (HFP) inlet enthalpy

7. Differential control rod bank worths

8. Integral control rod bank worths as a function of rod bank position.

The remainder of this report describes the Vepco NOMAD code, the purposes and interrelationships of the other computer codes which comprise the Vepco NOMAD model, the specific modeling of a reactor core with these codes, and the comparisons of calculated results with appropriate results obtained with the Vepco PDQ07 Discrete and One Zone models, the Vepco FLAME model, and with core measurements obtained from the Surry and North Anna Nuclear Power Stations.

SECTION 2 - CODE DESCRIPTION

2.1 Introduction

The Vepco NOMAD (Nuclear Operations Model for Analysis in One Dimension) code provides a relatively simple and inexpensive method for calculating axial power distributions and core reactivity. The calculation contains three levels of iteration: a source or flux calculation is performed during the inner iterations, thermal-hydraulic feedback in the second iteration level, and the xenon concentration in the outer iterations. The two outer levels of iteration are optional.

The neutron flux is calculated by solving the finite difference form of the two-group diffusion equations using Gauss elimination. A Chebyshev polynominal scheme is used to accelerate convergence.

The thermal-hydraulic feedback model accounts for the effect of the nonuniform fuel and moderator temperature distributions on the flux distribution. Single phase flow with no bulk boiling is assumed. Successive relaxation is used to accelerate convergence.

The NOMAD code provides two methods for calculating the axial xenon distribution. The equilibrium xenon calculation is based on the present flux distribution. The xenon depletion calculation is based on the flux distributions from the present and the previous timesteps using an iterative technique. The first method is applicable to fuel cycle design calculations, whereas the second is used for load follow maneuvers or xenon transients.

Using the xenon distribution obtained, flux and thermal-hydraulic calculations are performed again. This process continues until the flux, thermal-hydraulic, and xenon convergence criteria are satisfied.

The NOMAD code also contains the following capabilities:

1. Radial buckling calculation and normalization
2. Criticality search on a selected variable
3. Delta-I control
4. Boration and dilution calculations
5. Final Acceptance Criteria (FAC) Analysis
6. Differential and integral rod worth calculations
7. Xenon worth calculation.

Figure 2-1 is a simplified flow diagram of the calculations performed by NOMAD.

The remainder of this section describes in greater detail the models used in the flux, thermal-hydraulic, and xenon calculations, and the methods employed in the seven calculations listed above.

2.2 Neutron Flux Calculation

NOMAD calculates the neutron flux by solving the following finite difference diffusion equations:

$$-\frac{d}{dz}\left[D_i(z)\frac{d}{dz}\phi_i(z)\right] + \Sigma_i(z)\phi_i(z) = \chi_i\ G(z)/\lambda + \Sigma_{r,i-1}(z)\phi_{i-1}(z), \quad i=1,2, \qquad (2.1\text{-}1)$$

where

i = the neutron energy group,

$D_i(z)$ = diffusion coefficient for group i at position z,

ϕ = neutron flux,

$\Sigma_i(z) = D_i(z)B_i^2(z) + \Sigma_{a,i}(z) + \Sigma_{p,i}(z)$ = the total cross section,

B^2 = radial buckling,

Σ_a = absorption cross section,

Σ_r = removal cross section,

Σ_p = poison cross section,

χ_i = fraction of fission neutrons born in group i (χ_1=1, χ_2=0),

$G(z) = \sum_{j=1}^{2} \nu\Sigma_{f,j}(z)\phi_j(z)$ = the fission source,

Σ_f = fission cross section,

ν = number of neutrons per fission, and

λ = eigenvalue = $K_{eff.}$

Equation 2.1-1 is, of course, simply a set of neutron balance equations, where neutron losses on the left-hand side must be balanced by neutron gains on the right-hand side. The first term on the left-hand side describes the leakage from a unit volume by neutron migration in the z-direction, and the second term represents all other losses. These include absorption (Σ_a and Σ_p), leakage in the radial directions (DB^2), and scattering out of the energy group ($\Sigma_{r,i}$). The sources on the right-hand side are fission ($\chi_i\ G(z)/\lambda$), and scattering into the group from the next-higher energy group ($\Sigma_{r,i-1}$).

The z-axis in the region of solution, $0 \leq z \leq Z$, is subdivided into axial regions by mesh points at which the solution is to be determined. At some interior point z_n, integrate from

$$z_{n-}=z_n - \frac{1}{2} h_{n-} \quad \text{to} \quad z_{n+}=z_n + \frac{1}{2} h_{n+},$$

where h_{n-} and h_{n+} are the heights of the axial regions below and above point z_n, respectively. (See Figure 2-2.)

The approximations

$$\left.\frac{d\phi_i(z)}{dz}\right|_{z_{n-}} = \frac{\phi_i(z_n)-\phi_i(z_{n-1})}{h_{n-}}, \qquad \left.\frac{d\phi_i(z)}{dz}\right|_{z_{n+}} = \frac{\phi_i(z_{n+1})-\phi_i(z_n)}{h_{n+}}, \tag{2.1-2}$$

$$\int_{z_{n-}}^{z_{n+}} \Sigma_i(z)\phi_i(z)dz = \Sigma_i(z_{n-})\int_{z_{n-}}^{z_n} \phi_i(z)dz + \Sigma_i(z_{n+})\int_{z_n}^{z_{n+}} \phi_i(z)dz$$

$$= \left[\Sigma_i(z_{n-})\frac{h_{n-}}{2} + \Sigma_i(z_{n+})\frac{h_{n+}}{2}\right]\phi_i(z_n) = \overline{\Sigma}_i^n \phi_i^n, \tag{2.1-3}$$

and

$$\frac{D_i(z_{n\pm})}{h_{n\pm}} = \overline{D}_i^{n\pm} \tag{2.1-4}$$

are used to obtain the difference equation

$$-\overline{D}_i^{n+} \phi^{n+1} + (\overline{D}_i^{n+} + \overline{D}_i^{n-} + \overline{\Sigma}_i^n)\phi_i^n - \overline{D}_i^{n-} \phi_i^{n-1}$$

$$= \frac{\chi_i}{\lambda} \sum_{j=1}^{2} \overline{\nu\Sigma}_{f,j}^n \phi_j^n + \overline{\Sigma}_{r,i-1}^n \phi_{i-1}^n. \qquad (2.1\text{-}5)$$

From the form of the equations it can be seen that the macroscopic cross-sections are required to be constant in each half-interval. In fact, the code assumes these parameters are constant in each axial region.

The code also assumes that the three top and bottom axial regions are reflector regions and the flux at the outer boundaries of these regions is zero. The system of equations given by Equation 2.1-5 and the zero flux boundary conditions, may be expressed in matrix form as

$$\underline{\underline{M}}\,\underline{\phi} = \frac{1}{\lambda} \underline{\underline{F}}\phi + \underline{\underline{R}}\phi$$

where $\underline{\underline{M}}$, $\underline{\underline{F}}$, and $\underline{\underline{R}}$ are tridiagonal matrices.

NOMAD solves this system of equations by the method of power iterations using Gaussian elimination. The rate of convergence is accelerated by replacing the pth iterate of $\underline{\phi}$ (i.e. $\underline{\phi}^p$) with a linear combination of $\underline{\phi}^p$ and the previous iterate, $\underline{\phi}^{p-1}$,

$$\underline{\tilde{\phi}}^p = \underline{\phi}^p (1 + \theta^p) - \underline{\phi}^{p-1}\theta^p, \qquad (2.1\text{-}7)$$

where θ^p is an acceleration parameter computed on the basis of Chebyshev polynomials.

The eigenvalue for iteration n is calculated as

$$\lambda^n = \lambda^{n-1} \frac{\sum_{k=1}^{N} \sum_{j=1}^{2} \nu\Sigma_{f,j}(z_k)\phi_j^n(z_k)}{\sum_{k=1}^{N} \sum_{j=1}^{2} \nu\Sigma_{f,j}(z_k)\phi_j^{n-1}(z_k)}, \qquad (2.1\text{-}8)$$

where N is the total number of mesh points.

The converged solution to Equation 2.1-6 gives the fast and thermal fluxes at the mesh points between each axial region. Next, NOMAD calculates the fluxes at the center of each region. For a region of height 2m, label the top and bottom mesh points as t and b, respectively, and the region center as c. (See Figure 2-3.)

Using the approximations

$$\left.\frac{d\phi_i(z)}{dz}\right|_b = \frac{\phi_i(c)-\phi_i(b)}{m}, \qquad \left.\frac{d\phi_i(z)}{dz}\right|_t = \frac{\phi_i(t)-\phi_i(c)}{m}, \qquad (2.1\text{-}9)$$

and

$$\int_b^t \Sigma_i(z)\phi_i(z)dz = \Sigma_i(c)\phi_i(c)2m, \qquad (2.1\text{-}10)$$

and integrating Equation 2.1-1 yields

$$D_i(c)\left[\frac{2\phi_i(c) - \phi_i(t) - \phi_i(b)}{2m^2}\right] + \Sigma_i(c)\phi_i(c)$$

$$= \frac{\chi_i}{\lambda} \sum_{j=1}^{2} \nu\Sigma_{f,j}(c)\phi_j(c) + \Sigma_{r,i-1}(c)\phi_{i-1}(c) \qquad (2.1\text{-}11)$$

Since the fluxes at each region boundary, $\phi_i(b)$ and $\phi_i(t)$ are known, NOMAD solves for the region center flux $\phi_i(c)$ directly. The code then integrates $\phi_i(b)$, $\phi_i(c)$, and $\phi_i(t)$ using Simpson's method to obtain the average fast and thermal fluxes in the region.

Finally, the relative power is calculated for each region Z

$$P(Z) = \frac{K\Sigma_{f1}(Z)\bar{\phi}_1(Z) + K\Sigma_{f2}(Z)\bar{\phi}_2(Z)}{\sum_{Z=1}^{R} K\Sigma_{f1}(Z)\bar{\phi}_1(Z) + K\Sigma_{f2}(Z)\bar{\phi}_2(Z)} * R \qquad (2.1\text{-}12)$$

where P(Z) = relative power in axial region Z

K = energy per fission (watt-sec)

$\bar{\phi}_i(Z)$ = average group i flux in region Z

R = total number of axial regions with fuel

2.3 Thermal-Hydraulic Feedback

The thermal-hydraulic feedback model uses an energy balance to calculate the moderator enthalpy as a function of axial position

$$Enthalpy_{out} = Enthalpy_{in} + Power / Flow\ Rate, \qquad (2.3\text{-}1)$$

where $Enthalpy_{out}$ = moderator enthalpy exiting the region (BTU/lbm)

$Enthalpy_{in}$ = moderator enthalpy entering the region (BTU/lbm)

Power = power produced in the region (BTU/hr)

Flow Rate = core moderator flow rate (lbm/hr).

Single phase, homogeneous flow is assumed with no bulk boiling or void formation. The moderator enthalpy and system pressure are input to the HOH subroutine[8], which calculates the corresponding moderator temperature.

The thermal-hydraulic feedback model calculates the fuel temperature rise above the moderator temperature as a function of relative power and burnup

Fuel temp(Z) = Mod. temp(Z) + (DGEFPD * Burnup(Z) + FTFO) * RPD(Z) * PR, (2.3-2)

where

Fuel temp(Z) = fuel temperature in axial region Z

Mod. temp(Z) = moderator temperature in region Z

DGEFPD = fuel temperature vs. burnup coefficient (^{o}R/EFPD * Relative power)

Burnup(Z) = burnup of region Z (EFPD)

FTFO = fuel temperature vs. relative power coefficient

(^{o}R/Relative power)

RPD(Z) = relative power density in region Z

PR = core relative power (fraction of full power).

This fuel temperature fit is based on the one used in the Vepco PDQ07 thermal-hydraulic feedback model.[9]

NOMAD then recalculates the macroscopic cross sections based on the new fuel and moderator temperatures, performs another flux calculation, and performs another thermal-hydraulic calculation. This process continues until the thermal-hydraulic convergence criteria is satisfied.

2.4 Xenon Calculation

NOMAD calculates the iodine and xenon concentrations for each axial region using an analytic solution to the iodine and xenon rate equations. This solution is simply an integration of the iodine and xenon rate equations which assumes that the flux and the cross-sections remain constant over the time interval for which the calculation is performed. Prior to calculating the iodine and xenon concentrations, NOMAD normalizes the fast and thermal fluxes (in neutrons /cm^2-sec) to the core power level

$$\phi_2^N(Z) = \frac{\text{POWDEN} * \text{PR} * \text{RPD}(Z)}{K\Sigma_{f1}(Z) * \phi_1(Z)/\phi_2(Z) + K\Sigma_{f2}(Z)} \quad (2.4\text{-}1)$$

$$\phi_1^N = \phi_2^N(Z) * \phi_1(Z)/\phi_2(Z) \quad (2.4\text{-}2)$$

where $\phi_i^N(Z)$ = normalized flux for group i in region Z (neutrons/cm^2-sec)

POWDEN = power density (watts/cc)

PR = core relative power (fraction of full power)

RPD(Z) = relative power density in region Z

$\phi_i(Z)$ = region average relative flux for group i in region Z.

NOMAD then uses these normalized fluxes to calculate the iodine and xenon concentrations

$$I(Z)_{i+1} = \left[I(Z)_i - \frac{\gamma_I \Sigma_f \phi}{\lambda_I}\right] e^{-\lambda_I (t_{i+1} - t_i)} + \frac{\gamma_i \Sigma_f \phi}{\lambda_I} \quad (2.4\text{-}3)$$

$$Xe(Z)_{i+1} = \left[Xe(Z)_i - \frac{(\gamma_I + \gamma_{Xe}) \Sigma_f \phi}{LX} + \frac{\lambda_I I(Z)_i - \gamma_I \Sigma_f \phi}{\lambda_I - LX}\right] e^{-LX (t_{i+1} - t_i)}$$

$$- \left[\frac{\lambda_I I(Z)_i - \gamma_I \Sigma_f \phi}{\lambda_I - LX}\right] e^{-\lambda_I (t_{i+1} - t_i)} + \frac{(\gamma_I + \gamma_{Xe}) \Sigma_f \phi}{LX} \quad (2.4\text{-}4)$$

where $I(Z)_{i+1}$ = iodine concentration in region Z at step i+1

γ_I = iodine fission yield

λ_I = iodine decay constant

t_{i+1} = time (seconds) at step i+1

$\Sigma_f \phi = K\Sigma_{f1}(Z)\phi_1^N(Z) + K\Sigma_{f2}(Z)\phi_2^N(Z) / K_{AVG}$

K_{AVG} = average energy per fission

$Xe(Z)_{i+1}$ = xenon concentration in region Z at step i+1

γ_{Xe} = xenon fission yield

$$LX = \lambda_{Xe} + \sigma_{a1}^{Xe} \phi_1(Z) + \sigma_{a2}^{Xe} \phi_2(Z)$$

λ_{Xe} = xenon decay constant

σ_{aj}^{Xe} = xenon absorption cross section, group j.

To calculate the equilibrium iodine and xenon concentrations, the exponential terms in Equations 2.4-3 and 2.4-4 are set to zero.

Each xenon depletion is actually performed in two substeps. During the first substep, the xenon is depleted for 55% of the depletion time using the flux from the previous timestep. The second substep, which depletes the remaining 45% of the depletion time, is performed iteratively with the flux calculation at the present timestep. Thus, the xenon is depleted with the flux from the previous timestep for 55% of the depletion and with the flux from the present timestep for 45% of the depletion.

After NOMAD calculates the iodine and xenon concentrations, the macroscopic cross sections are adjusted and the flux and thermal-hydraulic calculations are performed again. The iodine and xenon concentrations are calculated with the new fluxes. This process continues until the xenon distribution converges.

2.5 Radial Buckling Coefficient Model

The radial buckling model accounts for radial leakage and compensates for the radial dimensions which are neglected in a one-dimensional axial model. The buckling coefficients, which are used to calculate the radial buckling as a function of core height, are adjusted to obtain agreement with a three-dimensional code for axial offset and relative power at the core midplane. The equations for the radial buckling function expressed in terms of the buckling coefficients are:

$$BUK1(Z) = BO * (1 + BTILT * (Z - ZO) / HT) * CZ(Z) \quad (2.5\text{-}1)$$

$$BUK2(Z) = BTH * BUK1(Z) \quad (2.5\text{-}2)$$

$$CZ(Z) = \begin{cases} \cos(BMID * PI * (Z - ZO) / HT) & \text{if } BMID > 0.05 \\ 1.8 - \cos(BMID * PI * (Z - ZO) / HT) & \text{if } BMID < -0.05 \\ 1.0 & \text{otherwise,} \end{cases} \quad (2.5\text{-}3)$$

where

BUK1(Z) = Fast group buckling

BUK2(Z) = Thermal group buckling

BO = Buckling amplitude coefficient

BMID = Buckling curvature coefficient

BTILT = Buckling tilt coefficient

BTH = Thermal-to-fast group buckling ratio

HT = Active core height

Z = Axial position

ZO = HT / 2

PI = 3.1415927.

For a positive BMID, the function is a convex curve. For a negative BMID, the function is a concave curve. When BMID is near zero, the function is a straight line. BTILT adjusts the slope of the curve. If it is positive, the buckling is greater in the top half of the core. If it is negative, the buckling is greater in the bottom half of the core. BO and BTH must always be positive so that the buckling function is positive.

NOMAD has an automated buckling coefficient search option. The search iterates on BTILT, BMID and BO, respectively, until the axial offset, midplane power, and eigenvalue converge on the target values (eigenvalue target is 1.0) or until a maximum number of iterations have been performed and a warning is printed. When a buckling coefficient search is performed, the coefficients are written to a data set with the core average burnup (in EFPH). This buckling coefficient data set may be read and used in subsequent calculations by NOMAD. If the core average burnup lies between two burnups in the buckling coefficient table, linear interpolation is performed to determine the coefficients for that step.

2.6 Criticality Search

The criticality search option in NOMAD searches for the value of a selected variable (e.g., boron concentration, control rod bank position, core power level, HFP inlet enthalpy) which will give the desired target eigenvalue. The search takes the errors from the two previous guesses and uses linear interpolation or extrapolation to guess what value of the selected variable will give an error of zero. NOMAD performs another eigenvalue calculation with the new value of the search variable. The search continues until the eigenvalue converges on the target value.

Since a control rod bank can only be inserted in discrete steps, the criticality convergence criterion may not be satisfied when a control rod search is performed. In this case, the code optionally performs a critical boron search after the control rod bank position is adjusted as near to critical as possible.

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

2.7 Delta-I Control

In order to simulate a load follow or other maneuver where the reactor is required to operate within a certain delta-I band, a delta-I control option is available that automatically adjusts the control rods to keep the delta-I within its operating band. Delta-I is defined as:

$$\text{Delta-I}(\%) = \frac{\text{Power(top)} - \text{Power(bottom)}}{\text{Power(top)} + \text{Power(bottom)}} * PR * 100, \qquad (2.7\text{-}1)$$

where

Power(top) = Relative power in top half of core

Power(bottom) = Relative power in bottom half of core

PR = core relative power level.

When delta-I is outside the operating band, NOMAD moves D-bank from the bite position (216 steps) to the rod insertion limit by increments of 20 steps. Each of these points is used to determine the delta-I as a function of D-bank position by a cubic least squares fit. The cubic equation is then solved to find the D-bank position which will adjust delta-I to the designated value. If the desired delta-I can not be achieved, then delta-I is adjusted to the nearest possible value. The user may request that the code adjust delta-I to (1) the target delta-I (the center of the operating band) or (2) the nearest edge of the operating band. Adjusting delta-I to the edge of the band requires less boration or dilution to accomplish. The eigenvalue calculation is then performed at the adjusted D-bank position. NOMAD automatically performs a critical boron search after the rods are moved to re-establish criticality.

2.8 Boration and Dilution Calculations

Boration and dilution calculations are important when studying a possible load follow maneuver to insure that the water processing system can handle the rapid changes in boron concentration. NOMAD performs boration and dilution calculations for every step following the first criticality search if the boration/dilution parameters are input. NOMAD solves the following equations[10]:

$$WATER(J) = -SYSMAS / H_2ODEN \; * \ln\left[1 + \frac{(BOR(J) - BOR(J-1))}{(BOR(J-1) - C_{in})}\right] \qquad (2.8\text{-}1)$$

$$TIMEMN = WATER(J) / LDRATE, \qquad (2.8\text{-}2)$$

where

WATER(J) = Water processed (gallons) at step J

SYSMAS = Total primary coolant system mass (lbm)

H_2ODEN = Water density in letdown line (8.2 lbm/gal)

BOR(J) = Boron concentration at step J

BOR(J-1) = Boron concentration at step J-1

C_{in} = Boron concentration in letdown line

TIMEMN = minimum time required to perform boration/dilution

LDRATE = Letdown rate.

WATER(J) is multiplied by -1 in a dilution case in order to distinguish between boration and dilution cases.

If TIMEMN is greater than the time between steps J-1 and J, NOMAD calculates the maximum achievable change in the boron concentration and prints a warning message to indicate that the minimum time required for the boration or dilution is greater than the time allowed. If instructed, the code then performs another criticality search on control rod bank position, core power level, or inlet enthalpy at the maximum (or minimum) boron concentration achievable in the boration (or dilution).

2.9 Final Acceptance Criteria (FAC) Analysis

NOMAD is capable of performing Final Acceptance Criteria (FAC) analysis. One part of this capability is the average power distribution calculation for a load follow depletion. The code integrates the axial flux distributions over the timesteps specified to obtain the average flux distribution:

$$\overline{\phi}_i(Z) = \frac{\sum_{j=1}^{T} \phi_{ij}(Z) t_j}{\sum_{j=1}^{T} t_j}, \qquad i=1,2, \qquad (2.9\text{-}1)$$

where

$\phi_{ij}(Z)$ = group i flux in region Z for timestep j (neutrons/cm^2 - sec)

t_j = length of timestep j (hours)

T = total number of timesteps specified.

This flux distribution is substituted into Equation 2.1-12 to obtain the average power distribution and Equations 2.4-3 and 2.4-4 to obtain the average iodine and xenon distributions. The load follow depletion is then performed using these power and xenon distributions.

To perform a FAC analysis, NOMAD combines the axial power distributions that it calculates in the load follow calculations with the $F_{XY}(Z)$ data input to the code to determine $F_Z(Z)$, $F_{XY}(Z)_{calc.}$, $F_{XY}(Z)_{allowable}$, and $F_Q(Z)_{calc.}$ NOMAD performs the following sequence at each step in the FAC analysis case. First, it determines the rodded configuration (i.e., ARO, D in, D+C in) for each axial region. Then the code selects the $F_{XY}(Z)$ that corresponds to the axial level and the rodded configuration of each region. NOMAD checks each $F_{XY}(Z)$ to insure that it is not less than the minimum $F_{XY}(Z)$ allowed for that rod configuration as specified in the user input. If the reactor is not at full power, NOMAD adjusts the $F_{XY}(Z)$ as follows:

$$FXYREL = F_{XY}(Z) * (1 + ADJUST * (1 - PR)), \qquad (2.9\text{-}2)$$

where

FXYREL = $F_{XY}(Z)$ adjusted for the core relative power level

$F_{XY}(Z)$ = F_{XY} at axial region Z for 100% power

ADJUST = F_{XY} power adjustment factor

(e.g., 0.3 for North Anna, 0.2 for Surry)

PR = Core relative power level.

Next, NOMAD calculates the $F_Q(Z)$ for this case:

$$FQTEST = FXYREL * RPD(Z) * PR * FQGRID, \qquad (2.9\text{-}3)$$

where

FQTEST = $F_Q(Z)$ * PR for this step

RPD(Z) = Relative axial power in region Z

FQGRID = Correction factor for grids * uncertainty factor

= 1.025 * UF

UF = 1.03 (3 case FAC analysis)

UF = 1.00 (18 case FAC analysis).

If FQTEST is greater than the previous $F_Q(Z)$, then the following values are saved:

$$F_Q(Z)_{calc.} = FQTEST \quad (2.9\text{-}4)$$

$$F_{XY}(Z)_{calc.} = FXYREL \quad (2.9\text{-}5)$$

$$F_{XY}(Z)_{allowable} = F_Q(Z)\ limit/\ (F_Z(Z) * PR * FQGRID) \quad (2.9\text{-}6)$$

$$F_Z(Z) = RPD(Z). \quad (2.9\text{-}7)$$

Once the entire load follow simulation has been completed and the final values for $F_Z(Z)$, $F_{XY}(Z)_{calc.}$, $F_{XY}(Z)_{allowable}$, and $F_Q(Z)_{calc.}$ have been obtained, NOMAD checks for any limit violations for $F_{XY}(Z)_{calc.}$, $F_{XY}(Z)_{allowable}$, and $F_Q(Z)_{calc.}$ and flags them in the FAC ANALYSIS RESULTS output.

2.10 Differential and Integral Rod Worth Calculations

Differential and integral rod worth calculations are available in NOMAD. The control rod banks may be inserted or withdrawn in any order chosen by the user (e.g., single bank, multiple banks in overlap, multiple banks together, etc.) The differential worth is calculated as follows:

$$DIFF(I) = (1/RKEF3 - 1/RKEF1) * 1E+5 / ISTEPS, \quad (2.10\text{-}1)$$

where

DIFF(I) = Differential rod worth for case I (pcm/step)

RKEF3 = K_{eff} for case I+1

RKEF1 = K_{eff} for case I-1

ISTEPS = Number of steps rods moved from case I-1 to I+1.

The differential worth is not calculated for the first or last case of the rod worth sequence.

NOMAD calculates the integral worth as follows:

$$RINT(I) = (1/RKEF2 - 1/RKEF) * 1E+5, \qquad (2.10\text{-}2)$$

where

RINT(I) = integral rod worth for case I (pcm)

RKEF2 = K_{eff} for case I

RKEF = K_{eff} for 1st case of rod worth sequence.

The integral worth is calculated for every step of the rod worth sequence.

2.11 Xenon Worth Calculation

NOMAD can automatically calculate the xenon worth at any selected timestep(s). When the xenon worth option is on, NOMAD saves the calculated xenon distribution in a separate array, resets the xenon distribution to zero and performs another eigenvalue calculation. It calculates the xenon worth from the two eigenvalues:

$$\text{Xenon worth} = \frac{K_{eff}\text{ (no xenon)} - K_{eff}\text{(w/xenon)}}{K_{eff}\text{ (no xenon)} * K_{eff}\text{ (w/xenon)}} * 1E+5. \qquad (2.11\text{-}1)$$

The program then restores the saved xenon distribution. Thus, the xenon worth at any timestep of a problem can be determined without interrupting the flow of the other calculations being performed.

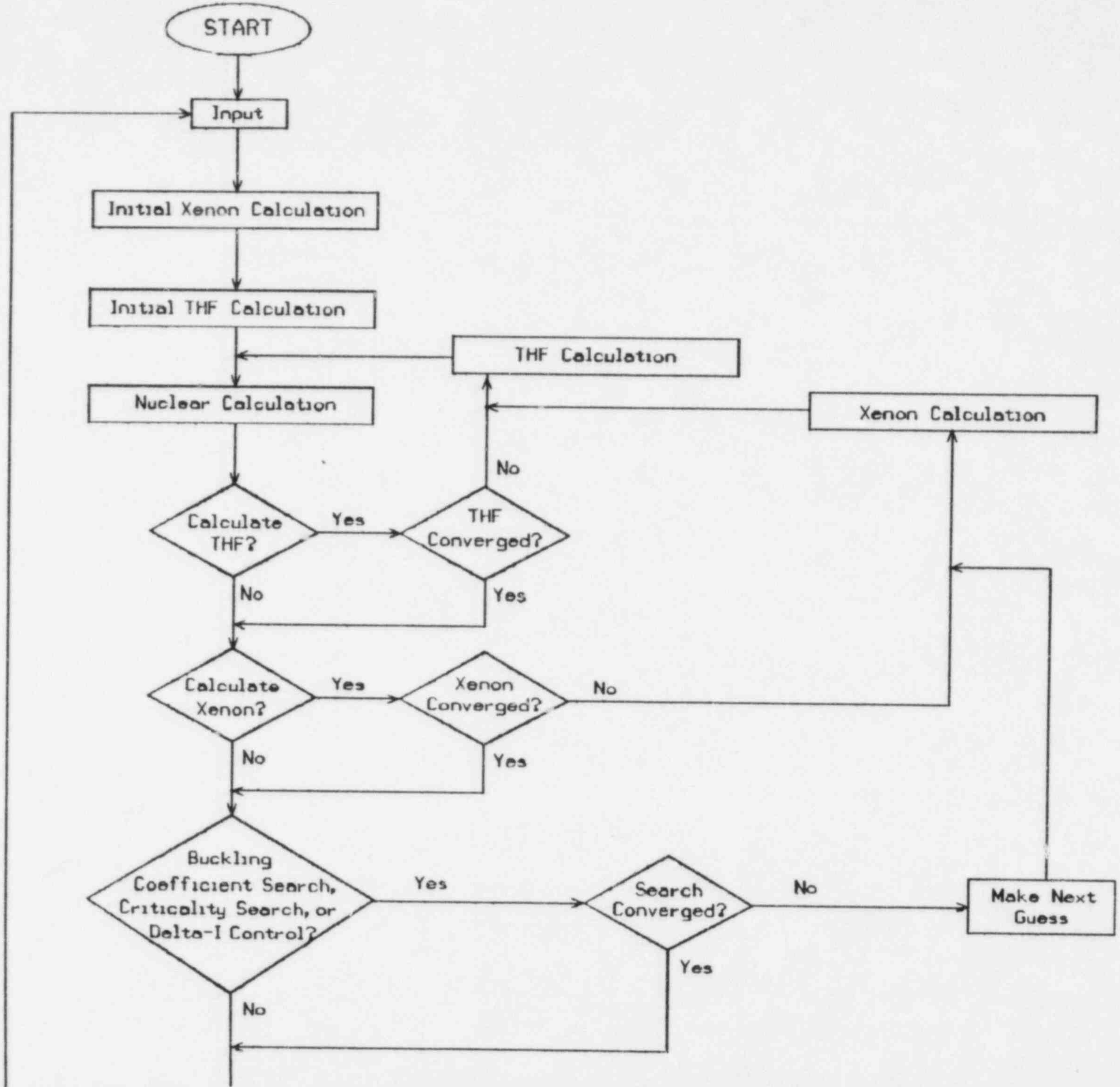


FIGURE 2-1
NOMAD Code Flow Diagram

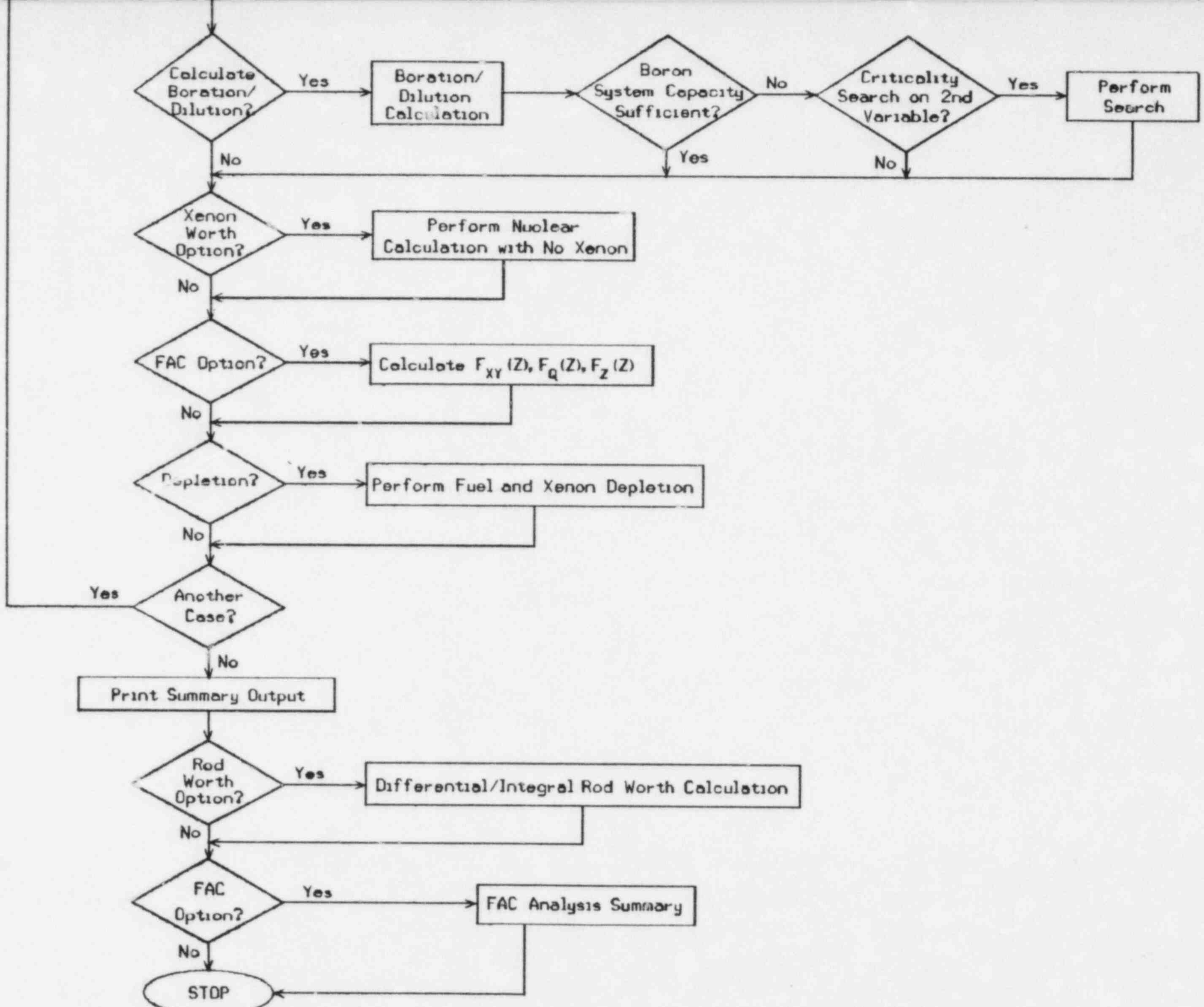


FIGURE 2-1 (Continued)

FIGURE 2-2

Axial Mesh Points and Regions

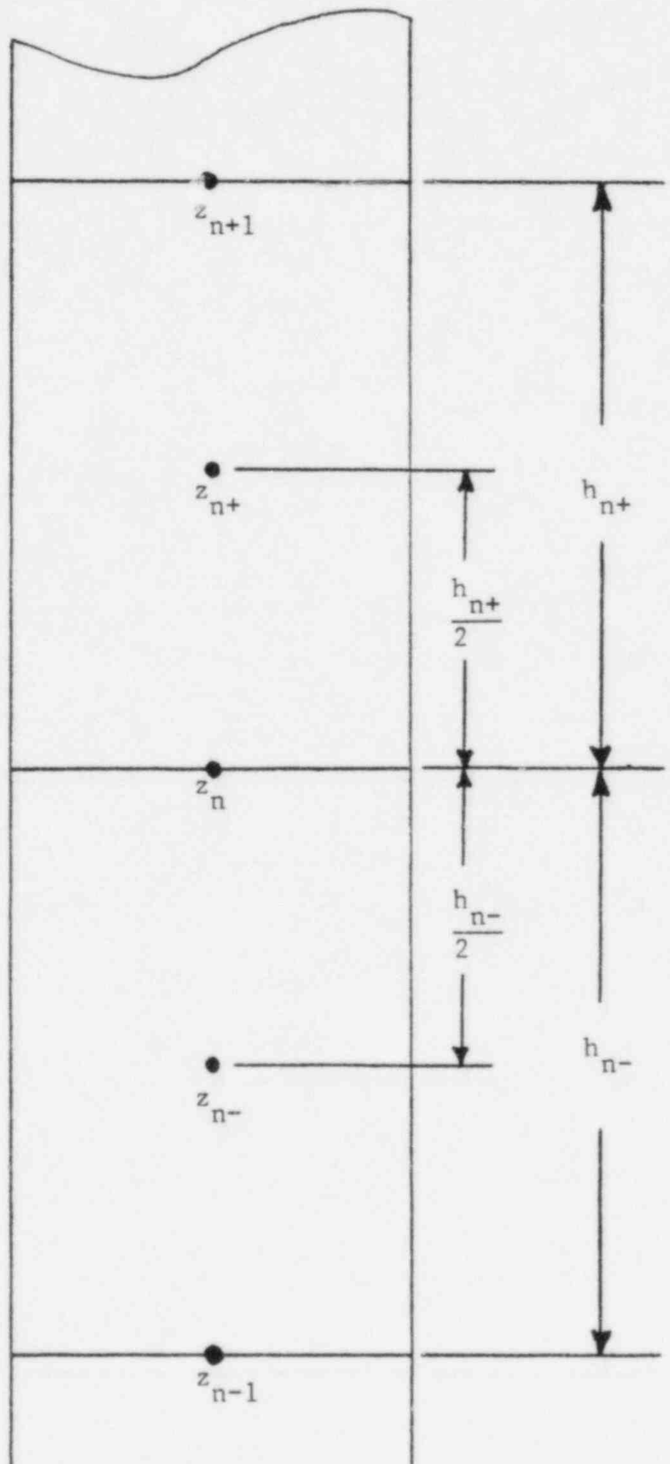
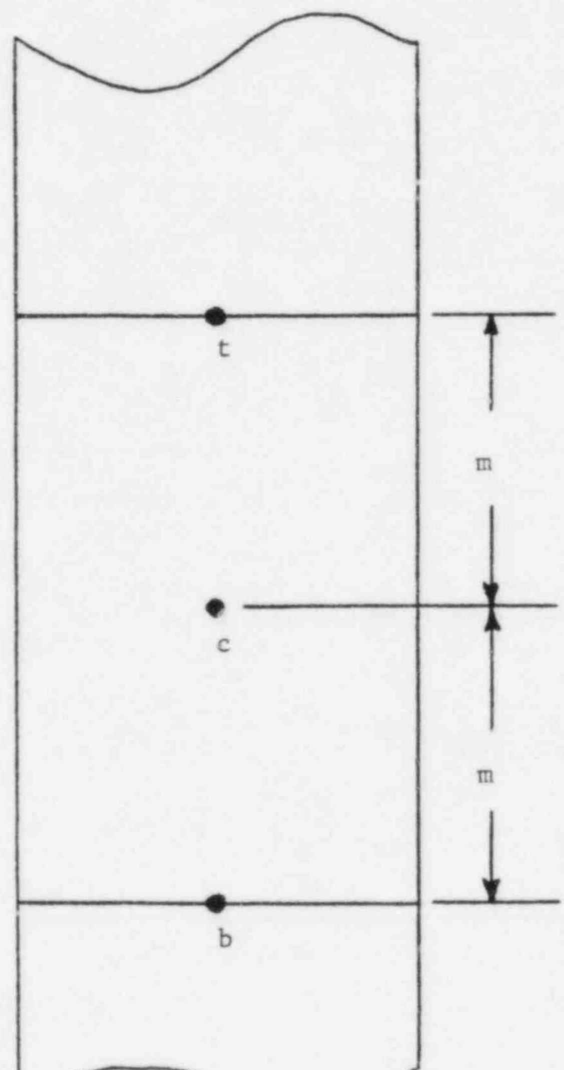


FIGURE 2-3

Axial Region Center and Boundary Mesh Points



SECTION 3 - MODEL DESCRIPTION

3.1 Introduction

The Vepco NOMAD model is used to calculate axial power distributions and core reactivity for one-dimensional geometries in which the core is represented by 32 axial fuel regions and three top and three bottom reflector regions. The method used by the Vepco NOMAD model to perform these calculations is a finite difference solution of the two energy group diffusion theory equations. Moderator and fuel temperature effects are accounted for by thermal-hydraulic feedback.

The Vepco NOMAD model incorporates several calculational steps. First, a quarter core PDQ07 One Zone(6) depletion is performed and the flux and concentration files at each burnup step are saved for the particular unit and cycle being studied. Then, PDQ performs flux-weighted macroscopic cross section calculations for a series of change cases at each burnup step from 150 MWD/MTU to EOC. The core average macroscopic cross sections from these calculations are then processed for input into the NOMAD computer code. These cross-sections, as well as the cycle normalization data (i.e., BOC core average axial burnup distribution, equilibrium iodine and xenon concentrations, integral control rod bank worths, and axial offset and core midplane power at each depletion step), are then used by NOMAD to perform an iterative, two-group finite difference diffusion theory calculation for the neutron flux as a function of core height. The method of solution comprises three levels of iteration: neutron flux, thermal-hydraulic feedback, and xenon concentration. The neutron flux calculation is performed first based on initial guesses of fuel and moderator temperatures and xenon concentrations. Then a new set of fuel and moderator temperatures are calculated. Using these new temperatures, another flux calculation is performed. Once the flux and temperatures have both converged, a new set of xenon concentrations is calculated. Using the new xenon concentrations, the flux and thermal-hydraulic feedback calculations are performed again. This process continues until the convergence criteria for all three levels are satisfied.

Several interrelated computer codes are used to perform the calculations outlined above. The computer codes comprising the Vepco NOMAD model and their interrelationships are presented in the flow diagram in Figure 3-1. The NOMAD computer code is the principal analytical tool in the Vepco NOMAD model. The other codes provide either input data or data manipulation. The PDQ07 One Zone model and the XSEDT(1) code are used to generate core average macroscopic cross sections at different core conditions. The XSFIT(1) and XSEXP(1) codes process these data for use by NOMAD. NULIF(4) is used to calculate the top and bottom reflector macroscopic cross sections. These generally remain the same from cycle to cycle. The Vepco PDQ07 One Zone, PDQ07 Discrete(5), and FLAME(7) models supply cycle normalization data. The FXYZ(2) code provides $F_{XY}(Z)$ input to NOMAD for FAC analysis. FDELH(3) and PCEDT(3) perform 1-D/2-D synthesis of NOMAD and PDQ07 Discrete or One Zone results.

The remainder of this section describes in greater detail the input to and functioning of the computer codes used in the Vepco NOMAD model.

3.2 Cross Section Generation

The Vepco NOMAD code requires the following two group macroscopic cross sections for the solution of the axial flux and power distributions:

D_1, Σ_{a1}, Σ_{r1}, $\nu\Sigma_{f1}$, $\kappa\Sigma_{f1}$, D_2, Σ_{a2}, $\nu\Sigma_{f2}$, and $\kappa\Sigma_{f2}$.

These cross sections actually consist of base macroscopic cross sections and polynomial coefficients that adjust the base cross sections for changes in the fuel and moderator temperatures and the boron and xenon concentrations. The cross sections are generated from the PDQ07 One Zone model for that unit and cycle.

The PDQ07 One Zone model is depleted to EOC and the flux and concentration files are saved at each burnup step. A series of restart calculations are performed at each burnup step from 150 MWD/MTU to EOC. (The BOC step is not included because there is no xenon present). Using the input flux and concentration files, PDQ only performs the flux-weighted macroscopic cross section calculations (i.e., no flux or eigenvalue calculation). The cases contain variations in the fuel and moderator temperatures and the boron and xenon concentrations which should include all core conditions encountered during reactor operation. The range covered for each variable is:

Fuel Temperature	331 to 2052 degrees Fahrenheit (N. Anna)
	487 to 2326 degrees Fahrenheit (Surry)
Moderator Temperature	543 to 613 degrees Fahrenheit (N. Anna)
	526 to 596 degrees Fahrenheit (Surry)
Boron Concentration	0 to 1800 ppm
Xenon Concentration	0 to 2.00 E-08 atoms/bn-cm.

The XSEDT code copies the core average macroscopic cross sections obtained from each of these calculations to a data set which is read by the XSFIT code.

The variables upon which each macroscopic cross section have been found to be dependent are listed in Table 3-1. The XSFIT code analyzes the PDQ07 macroscopic cross sections and generates base macroscopic cross sections and polynomial coefficients which express these cross sections in terms of these variables. It then compares the PDQ07 cross sections to those calculated with the polynomial coefficients to verify the accuracy of the coefficients.

These base cross sections and polynomial coefficients are passed to the XSEXP code which smooths several of the fast group polynomial coefficients and calculates base cross sections and polynomial coefficients beyond the lower and upper burnup extremes using a linear least squares extrapolation. These extrapolated cross sections are needed to account for axial regions with burnups less than the core average at BOC and greater than the core average at EOC. The XSEXP codes writes this final set of base cross sections and polynomial coefficients to a data set which is read by NOMAD.

3.3 Model Normalization

The Vepco NOMAD model for a particular unit and cycle must be normalized to the Vepco PDQ07 Discrete, PDQ07 One Zone, and FLAME models for the same unit and cycle.

The BOC axial burnup distribution from the Vepco FLAME model is input to the NOMAD model in the cycle/geometry deck. There is a one-to-one correspondence between the fueled axial regions in the NOMAD model and the axial nodes in the FLAME model. The NOMAD BOC axial power distributions at

HZP and HFP are normalized to the FLAME BOC axial power distributions using the buckling coefficient search option in the NOMAD code. This search finds a combination of buckling coefficient values which give a buckling distribution that forces the axial offset and the power at the core midplane to match those from FLAME for the same conditions.

The NOMAD model xenon parameters are normalized to the PDQ07 One Zone model. The NOMAD model is depleted from BOC to 150 MWD/MTU using the xenon parameters from the previous cycle. The fast and thermal xenon microscopic absorption cross sections are assumed to remain constant. The Iodine 135 and Xenon 135 fission yields are modified to force the NOMAD equilibrium iodine and xenon concentrations to agree with those from the PDQ07 One Zone model at 150 MWD/MTU:

$$\gamma_I(\text{new}) = {}_I(\text{old}) * I^{135}(\text{PDQ})/I^{135}(\text{NOMAD}) \qquad (3.3\text{-}1)$$

$$\gamma_{Xe}(\text{new}) = \left[\gamma_{Xe}(\text{old}) + \gamma_I(\text{old})\right] * \frac{Xe^{135}(\text{PDQ})}{Xe^{135}(\text{NOMAD})} - \gamma_I(\text{new}) \qquad (3.3\text{-}2)$$

The NOMAD model is depleted again from BOC to 150 MWD/MTU with the new fission yields to verify that the concentrations now agree with the PDQ07 One Zone model.

The axial power distributions obtained from the NOMAD model are normalized to the FLAME model results for the remainder of the cycle by performing buckling coefficient searches at each depletion step from 150 MWD/MTU to EOC. These buckling coefficients are saved in a table which is input to the NOMAD code for all subsequent calculations.

The NOMAD model control rod cross sections are normalized by forcing agreement between the NOMAD and PDQ07 Discrete rod bank integral worths. In the case of rod swap worth calculations, the NOMAD bank worths are normalized to the PDQ07 Discrete bank worths for each bank inserted alone. Otherwise, they are normalized to the PDQ07 worths for the banks inserted in sequence (e.g., D in, D+C in, D+C+B in., etc.)

The core average cross sections Σ_{a1}, Σ_{a2}, and Σ_{r1} from the PDQ07 "Rod bank out" case are subtracted from the same cross sections for the "Rod bank in" case. These values are input to the NOMAD code for that bank, the control rod normalization is set to 1.0, and the integral bank worth is calculated with NOMAD. The PDQ07 bank worth is divided by the NOMAD bank worth, and the control rod normalization is multiplied by that ratio. The bank worth is calculated again with NOMAD to verify that the worth now agrees with the PDQ07 Discrete model. This process is repeated for each rod bank.

3.4 1-D/2-D Synthesis

Prior to the execution of the 1-D/2-D synthesis option in NOMAD, a 2-D PDQ07 case is run for each of the rodded configurations present in the synthesis case (e.g., ARO, D in, D+C in, etc.). The IFM average power files from these PDQ07 cases are saved for input to the FDELH code. An input data set is created for FDELH, omitting the axial power sharing values. The job and case ID's for the IFM files are listed in the order that the rodded configurations occur from bottom to top of core. NOMAD reads this input data set and re-writes it with the axial power sharings which it calculates. FDELH subsequently reads this data and performs the 1-D/2-D synthesis:

$$F_p(x, y) = P_1 * F_{p1}(x, y) + P_2 * F_{p2}(x, y) + P_3 * F_{p3}(x, y), \qquad (3.4\text{-}1)$$

where

$F_p(x, y)$ = relative power for fuel in location (x, y)

P_n = axial power sharing from NOMAD for rod configuration n

$F_{pn}(x, y) = F_p(x, y)$ for rod configuration n.

The PCEDT code then performs a power census edit which provides the percentage of pins in the core whose relative power is greater than the specified value for percentage values of 1%, 2%,..., 9%, 10%, 20%,..., 80%, and 90%.

3.5 FAC Analysis Model

The Vepco FXYZ code provides NOMAD with the $F_{XY}(Z)$ values at each burnup step for each different rodded configuration which appears in the load follow calculations. It calculates these based on the three-dimensional power distributions obtained from FLAME and the $F_{\Delta H}$ data from the PDQ07 Discrete model

$$F_{XY}(Z) = \left[\frac{F_Q(X,Y,Z)}{P(Z)} * \frac{F_{\Delta H}(X,Y)}{RPD(X,Y)} * F_{CON} * F_{XEN} \right]_{\text{Maximum.}} \quad (3.5\text{-}1)$$

where

$F_Q(X,Y,Z)$ = relative power in node (X,Y,Z) from FLAME

P(Z) = core average axial power in plane Z from FLAME

$F_{\Delta H}(X,Y)$ = peak pin power in assembly (X,Y) from PDQ07 Discrete

RPD(X,Y) = relative power in assembly (X,Y) from FLAME

$F_{CON} = F_E * F_U$

F_E = engineering heat flux hot channel factor = 1.03

F_U = measurement uncertainty factor = 1.05

F_{XEN} = radial xenon re-distribution correction factor = 1.03.

If $F_{\Delta H}(X,Y)$ is less than RPD (X,Y), a value of 1.0 is substituted for that ratio. NOMAD reads the $F_{XY}(Z)$ data from a file where FXYZ stores them. Usually only data for ARO and D in rod configurations are necessary for FAC analysis. NOMAD then performs the FAC analysis calculation as previously described in Section 2.9.

3.6 Deep Insertion Control Rod Model

The radial buckling distribution must be adjusted to effectively predict the results of an unusual load follow or slow transient case where the control rods are deeply inserted (i.e., D-bank below 170 steps). This adjustment is not necessary for the load follow calculations performed in FAC analysis, since the rods usually are not very deep. Adjustment of the buckling distribution accounts for the radial flux re-distribution that occurs during deep control rod insertion.

Four steps are necessary for determining the buckling coefficients to compensate for the rod insertion:

1. Examine the anticipated (if predicting axial power behavior for a possible operational strategy) or actual (if analyzing measured data) control rod and core power history for the case in question and

select several control rod positions, roughly 20 steps apart, that cover the range of rod movement. (An ARI case is not necessary, because the ARO buckling coefficients are sufficient for this case.) Then select core power levels that generally correspond to those rod positions.

2. Use NOMAD to perform a critical boron search at the appropriate cycle burnup with the HFP, ARO equilibrium xenon distribution for each of the rod positions and corresponding power levels selected above.

3. Perform a FLAME calculation at the same burnup for each of the above cases with the same rod positions, power levels, and critical boron concentrations.

4. Perform a buckling coefficient search with NOMAD for each of these cases to normalize to FLAME results. Because FLAME is a three dimensional code, it accounts for the radial flux re-distribution. Therefore, the new radial buckling coefficients obtained by this method also account for that effect. The xenon distribution is frozen at HFP, ARO equilibrium conditions in both the NOMAD and FLAME calculations. This insures consistent results with both codes.

TABLE 3-1

MACROSCOPIC CROSS SECTION VARIABLE DEPENDENCE

Cross Section	Fuel Temp.	Mod. Temp.	Boron Conc.	(Boron * Mod. Temp.)	Xenon	$Xenon^2$	(Boron * Xenon)
D_1	X	X	X				
Σ_{r1}	X	X	X	X			
Σ_{a1}	X	X	X	X			
$\nu\Sigma_{f1}$	X	X	X				
$\kappa\Sigma_{f1}$	X	X	X				
D_2	X	X	X	X	X		
Σ_{a2}	X	X	X	X	X	X	X
$\nu\Sigma_{f2}$	X	X	X	X			
$\kappa\Sigma_{f2}$	X	X	X	X	X		

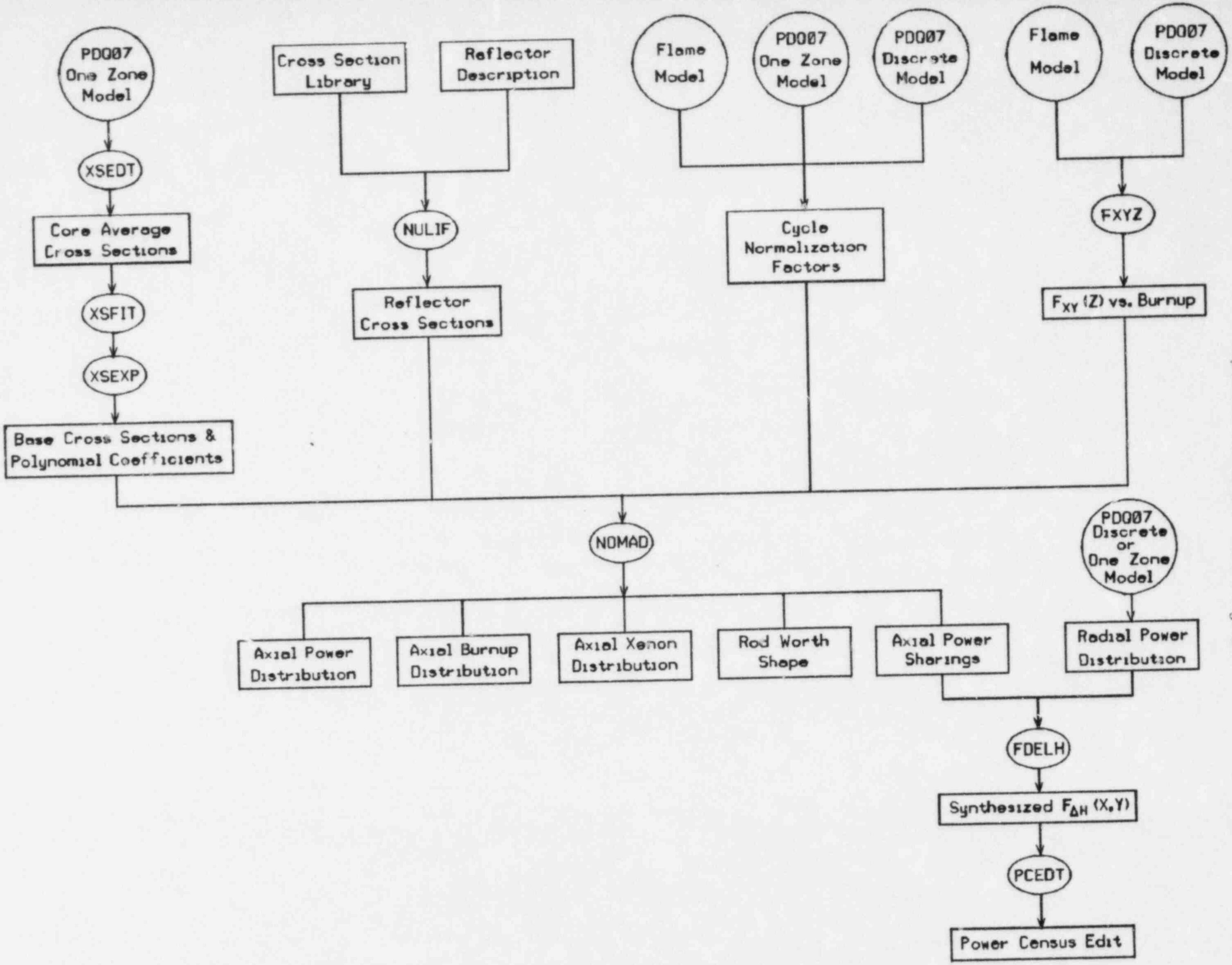


FIGURE 3-1
Vepco NOMAD Model Flow Diagram

SECTION 4 - USER INFORMATION

4.1 Input Description

NOMAD input is read as a series of numbered and categorized cards. The 010 through 060 cards are required at the beginning of every job. However, the 040 card can be omitted if a buckling coefficient table is input. The 070 through 090 cards are optional and should only be input when needed. Each card is free format. The data for a single card number may be continued onto subsequent cards, but the card number only appears on the first card. A 500 card is always the last card for each case (or step), and the user may run up to 749 dependent cases following the independent (i.e., first) case. Any of the numbered input cards may appear in a dependent case, but each case must end with a 500 card.

During the delta-I control and the criticality search options, the power, control rod positions, or boron concentration may be changed by the code. To use the values calculated by the code in the previous step, the user inputs a negative value for the appropriate variable(s) on the 500 card.

A detailed description of each input card begins on the following page.

010 CARD --- Title (1 line, 80 characters maximum)

020 CARD --- General Parameters

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	NRGNS	INTEGER	Number of axial regions
2	EPS	REAL	Eigenvalue convergence limit
3	IWRITE(1)	INTEGER	Temp., power, and burnup edit: 0 = Do not write edit 1 = Print & fiche edit 2 = Fiche edit
4	IWRITE(2)	INTEGER	Macroscopic cross section edit
5	IWRITE(3)	INTEGER	Iodine, xenon, and flux edit
6	IWRITE(4)	INTEGER	Axial xenon and RPD plot file
7	IWRITE(5)	INTEGER	Flux squared/power sharing edit (must =1 for 1-D/2-D synthesis)

030 CARD --- Thermal-Hydraulic Data

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	ITMAX	INTEGER	Max. no. of thermal iterations
2	TAU	REAL	Thermal relaxation parameter
3	THCON	REAL	Thermal convergence criterion
4	TMO	REAL	Moderator ref. temperature (° Farenheit)
5	TFO	REAL	Fuel reference temperature (° Rankine)
6	DGEFPD	REAL	Fuel temperature vs. burnup coefficient (°R/(EFPD*Rel. power))
7	FTFO	REAL	Fuel temp. vs relative power coeff. (°R / Relative power)
8	ENTHIN(1)	REAL	HZP inlet enthalpy (BTU/lbm)
9	ENTHIN(2)	REAL	HFP inlet enthalpy (BTU/lbm)
10	HEIGHT	REAL	Core height (centimeters)
11	POWER	REAL	Total core power level (watts)
12	FLORAT	REAL	Core flow rate (lbm/hour)
13	SYSPR	REAL	System pressure (psia)
14	POWDEN	REAL	Power density (watts/cc)

040 CARD -- Buckling Coefficient Input

(Not required if buckling coefficient table is used)

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	BO	REAL	Buckling amplitude
2	BMID	REAL	Buckling curvature coefficient
3	BTILT	REAL	Buckling tilt coefficient
4	BTH	REAL	Thermal buckling fraction
5	RPDMID	REAL	Target power at midplane
6	AXOFFT	REAL	Target axial offset
7	ILOCK	INTEGER	Buckling coeff. table flag. Use this card instead of table: 0 = For this step only 1 = For subsequent steps

050 CARD -- Control Rod Cross Sections

(Seven cards: control D-A, shutdown B&A, PL rods)
(Number 050 appears on the first card only)

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	RODWTH(1, I)	REAL	Fast macroscopic absorption cross section
2	RODWTH(2, I)	REAL	Thermal macroscopic absorption cross section
3	RODWTH(3, I)	REAL	Fast macroscopic removal cross section
4	RODWTH(4,I)	REAL	Control rod normalization
5	IOVRLP(I)	INTEGER	Bank overlap (1st 3 cards only)
6	RDBANK(I)	REAL	Bank name (8 characters in '')

060 CARD -- Delta-I Control and Criticality Search Parameters

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	IDCNTL	INTEGER	Delta-I control option. If delta-I out of allowed band: -1 = adjust to edge of band 0 = do not adjust +1 = adjust to target value
2	TARGET	REAL	Target delta-I (%)
3	DELTIL	REAL	Lower delta-I band (%)
4	DELTIH	REAL	Upper delta-I band (%)
5	RILHFP	REAL	D-bank HFP insertion limit (steps)
6	RILHAF	REAL	D-bank insertion limit at 50% power (steps)
7	EIGEN	REAL	Criticality search target eigenvalue
8	EPS2	REAL	Criticality search convergence limit
9	ICRIT	INTEGER	Criticality search variable: 1 = Boron concentration (ppm) 2 = D-bank position (steps) -2 = D-bank position followed by boron concentration 3 = Power (%) 4 = HFP inlet enthalpy (Btu/lbm)
10	CRTMAX	REAL	Maximum value of ICRIT
11	CRTMIN	REAL	Minimum value of ICRIT

070 CARD --- Core Average Fixed Parameters (Optional Card)

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	IFX1	INTEGER	Fixed fuel and moderator temperature flag: 0 = Do not fix 1 = Fix
2	IFX2	INTEGER	Fixed burnup flag
3	IFX3	INTEGER	Fixed xenon flag
4	IFX4	INTEGER	Time flag: -1 = continuous clock +1 = 24 hour clock
5	TMFX	REAL	Moderator temperature (°F)
6	TFFX	REAL	Fuel temperature (°R)
7	EFPDFX	REAL	Burnup (EFPD)
8	XENFX	REAL	Xenon conc. (Atoms/bn-cm)
9	TIMEFX	REAL	Initial clock time (hours)

080 CARD --- Boration/ Dilution Input (Optional Card)

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	SYSMAS	REAL	Primary system mass (lbm)
2	CBBOR	REAL	Boration line boron conc. (ppm)
3	CBDIL	REAL	Dilution line boron conc. (ppm)
4	FLOBOR	REAL	Boration line flow rate (gpm)
5	FLODIL	REAL	Dilution line flow rate (gpm)
6	NXCRIT	REAL	If boron system incapable of maintaining criticality: 0 = Print warning only 1 = Adjust D-bank position 2 = Adjust power level 3 = Adjust inlet enthalpy

090 CARD --- FAC Analysis Input (Optional Card)

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	FQGRID	REAL	Grid correction factor: 1.056 = 3 case 1.025 = 18 case
2	ADJUST	REAL	F_{XY} power adjustment factor: 0.3 = North Anna 0.2 = Surry
3	NMAPS	INTEGER	Number of $F_{XY}(Z)$ maps input
4	FXYMIN(I) (I = 1, NMAPS)	REAL	Minimum $F_{XY}(Z)$ values ($* F_U * F_E$) for each $F_{XY}(Z)$ map
5	MPBURN(I) (I=1, NMAPS)	INTEGER	Burnup (MWD/MTU) at which each $F_{XY}(Z)$ map is printed
6	FQLIMX(J)	REAL	X-coordinate for F_Q limit curve (core height in feet)
7	FQLIMY(J)	REAL	Y-coordinate for F_Q limit curve ($F_Q(Z)$ limit)
	(Input variables 6 & 7 in pairs for J=1, 4)		
8	FXYLMX(J)	REAL	X-coordinate for F_{XY} limit line (core height in feet)
9	FXYLMY(J)	REAL	Y-coordinate for F_{XY} limit line ($F_{XY}(Z)$ limit)
	(Input variables 8 & 9 in pairs for J=1, 4)		

100 CARD --- Recovery File Options (Optional Card)

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	DFILE	REAL	Recovery file flag: SAVE = Save new recovery file RESTORE = Restore old file
2	FNAME	REAL	8 character file name

(Note: 1 space between variable 1 and variable 2)

500 CARD -- Case Card

VARIABLE	VARIABLE NAME	TYPE	DESCRIPTION
1	TIME	REAL	Depletion time interval (EFPH)
2	PCTPOW	REAL	Core average power (%) **
3	IRDPOS(1)	INTEGER	D-bank position (steps) **
4	IRDPOS(2)	INTEGER	C-bank position (steps) **
5	IRDPOS(3)	INTEGER	B-bank position (steps) **
6	IRDPOS(4)	INTEGER	A-bank position (steps) **
7	IRDPOS(5)	INTEGER	SB-bank position (steps) **
8	IRDPOS(6)	INTEGER	SA-bank position (steps) **
9	IRDPOS(7)	INTEGER	PL-Rods position (steps) **
10	BORON	REAL	Boron concentration (ppm) **
11	IXEN	INTEGER	Xenon Option: -1 = Xenon from previous step 0 = Xenon depletion 1 = No xenon 2 = Equilibrium xenon
12	IOPT	INTEGER	Case Option: 0 = Static case 1 = Depletion 2 = 1st step of rod worth sequence 3 = Criticality search 4 = 1st step of FAC analysis 5 = Xenon worth calculation 6 = Buckling coefficient search 7 = Frozen THF 8 = 1st step of average power distribution calculation -8 = Final step of average power distribution calculation; perform load follow depletion

**If a negative value is input for any of these variables, the value from the end of the previous time step is used.

NOMAD also reads a cycle/geometry deck. The data in this deck remains constant for a particular unit and cycle. The cycle deck contains reflector macroscopic cross sections, xenon parameters, axial region dimensions, and the BOC axial burnup distribution.

The cycle/geometry deck is read in free format in the following order:

D_1	Σ_{a1}	Σ_{r1}		(Bottom reflector fast cross sections)
D_2	Σ_{a2}			(Bottom reflector thermal cross sections)
D_1	Σ_{a1}	Σ_{r1}		(Top reflector fast cross sections)
D_2	Σ_{a2}			(Top reflector thermal cross sections)
σ_{a1}^{Xe}	σ_{a2}^{Xe}	γ_I	γ_{Xe}	(Iodine and xenon parameters)

Region height (cm)	BOC Burnup (=0.0)	(Region 1 - bottom reflector)
Region height (cm)	BOC Burnup (=0.0)	(Region 2 - bottom reflector)
Region height (cm)	BOC Burnup (=0.0)	(Region 3 - bottom reflector)
Region height (cm)	BOC Burnup (EFPD)	(Region 4 - bottom fuel region)
.	.	.
.	.	.
.	.	.
Region height (cm)	BOC Burnup (EFPD)	(Region (NRGNS-3) - top fuel region)
Region height (cm)	BOC Burnup (=0.0)	(Region (NRGNS-2) - top reflector)
Region height (cm)	BOC Burnup (=0.0)	(Region (NRGNS-1) - top reflector)
Region height (cm)	BOC Burnup (=0.0)	(Region (NRGNS) - top reflector)

4.2 Error & Warning Messages

NOMAD prints error messages whenever it detects an error in input or execution. If it is an input error, it will terminate after checking the remainder of the input for that case for errors. If the error occurs during execution, the job is terminated.

The code also prints warning messages for less severe problems, such as a criticality search not converging. Execution continues after a warning message is printed.

NOMAD returns a condition code of zero for a successful job completion in most cases. Other programmed return codes are:

11 - Power sharing calculated for 1-D/2-D synthesis

333 - Job terminated, recovery file could not be found

444 - Job terminated, thermal-hydraulic feedback did not converge

555 - Job terminated, maximum number of cases exceeded

666 - Job terminated, buckling search did not converge

777 - Job terminated, cross section file error

838 - Job terminated, FAC analysis input error

999 - Job terminated, input error.

4.3 Execution Time

Approximate execution (CPU) times for NOMAD for several types of cases are given below:

HZP, no xenon	0.1 seconds
HFP, eq. xenon	1.1 seconds
HFP, xenon depletion, criticality search	0.7 seconds
HFP, eq. xenon, criticality search	3.4 seconds
Buckling coefficient search (10 depletion steps)	4 - 6 minutes
FAC analysis (72 hour load follow)	5 - 8 minutes

4.4 Output

NOMAD offers the user flexibility in its output control. The output options are IWRITE(1) - IWRITE(5) on the 020 card. All five of these options are turned off and on by values of 0 and 1, respectively. All edits are written to the printer except IWRITE(4), which writes to a plot file.

The temperature, power, and burnup edit is a one page edit which includes the k-effective, delta-I, axial offset, number of iterations to reach convergence, and cycle average burnup. It also prints for each axial region and the core average the values of each of the following parameters: k-infinity, moderator enthalpy (BTU/lbm), moderator temperature (degrees F), fuel temperature (degrees R), relative power density (RPD), and fuel burnup (EFPH). This edit is generally the most useful one in design calculations. The macroscopic cross section edit is a two page edit which provides the fast and thermal cross sections for each axial region and the core average. This edit is recommended only for debugging purposes or special applications.

The iodine, xenon, and flux edit lists the xenon and iodine concentrations and the fast and thermal fluxes by axial region and core average. This edit is required for normalization of the xenon model. The flux squared/power sharing edit gives the fraction of the flux squared and the power which occurs in each axial segment with a different rodded configuration. This edit must be on to perform 1-D/2-D synthesis or rod swap worth calculations, and it may be useful for other calculations.

These edits are normally printed once per case, but they may appear more than once when a criticality search or a delta-I control adjustment is performed. The first occurrence of each edit contains the results of the nuclear calculation prior to any variable adjustments. The final occcurrence contains the results of the final nuclear calculation after all searches have been completed.

In addition to the output options selected, NOMAD always prints the following output:

(1) Input card image listing

(2) Cycle/geometry deck card image listing

(3) Buckling coefficients & distribution and reflector cross sections

(4) Input summary

(5) 1-D analysis case summary.

NOMAD offers two options for output to a data set for creating plots. The first is IWRITE(4), which writes the axial xenon and relative power distributions to a file. Each distribution is labeled with the cycle burnup for that step, so that several distributions may be saved in the same file. The user is advised to not use this option in any job where IOPT = 2, 4, or 5 (rod worth, FAC analysis, or xenon worth). These options also write data to the same plot file. The plot output for these three options is similar to the printed output for them.

The second plot option writes most of the data which appears in the 1-D Analysis Case Summary to a file. The data includes the case number, time, power, boron, D-bank position, delta-I, peak power, k-effective, and xenon. If the boration/dilution calculations are performed, the water processed per step is written instead of the xenon concentration. This option is exercised by assigning a data set to the appropriate unit (see Section 4.5).

4.5 I/O Units

The I/O units used by NOMAD are:

Unit No.	Function
2	Buckling coefficient table (input)
3	Cycle/geometry input deck
4	Cross section input
5	Card input
6	Printer and microfiche
8	$F_{XY}(Z)$ input
9	Case summary plot file
10	Plot file for RPD, xenon, rod worth, FAC analysis, xenon worth
11	Buckling coefficient table (output)
12	Microfiche
13	Recovery file
14	Scratch disk space
21	FDELH code input for 1-D/2-D synthesis.

Data is written to unit no. 11 only when a buckling coefficient search is performed.

SECTION 5 - RESULTS

5.1 Introduction

The purpose of this section is to demonstrate the predictive capability of the Vepco NOMAD model for calculations of axial power distributions, delta-I and axial offsets, critical boron concentrations, differential and integral control rod worths, load follow maneuvers, and core peaking limits for FAC analysis. This section presents: 1) reactivity parameter comparisons to measured data from the Surry and North Anna Nuclear Power Stations and to other Vepco codes; 2) a thermal-hydraulic feedback calculation comparison to COBRA[11]; 3) axial power distribution comparisons to measured data from Surry and North Anna; 4) differential and integral control rod worth comparisons to measured data and the Vepco FLAME model; 5) comparisons of load follow maneuver simulations of delta-I, axial offset, and critical boron concentration to measured data from North Anna; and 6) FAC analysis results obtained with NOMAD.

5.2 Reactivity Parameters

Differential boron worths were calculated with NOMAD at BOC, HZP and EOC, HFP. The BOC, HZP values are compared to measured data and the EOC, HFP values are compared to the Vepco PDQ07 Discrete model in Table 5-1. Isothermal temperature coefficients obtained with NOMAD at BOC, HZP also are compared to measured data in Table 5-1.

Figures 5-1 through 5-6 compare NOMAD xenon worths after startup, orderly shutdown, and trip to results obtained with XETRN[12], Vepco's zero-dimensional xenon transient code. Note that XETRN assumes the xenon worth is a linear function of the xenon concentration, whereas NOMAD calculates the xenon worth directly from the eigenvalues.

5.3 Thermal-Hydraulic Feedback

The accuracy of the thermal-hydraulic feedback model has been verified by direct comparison with the COBRA code. Both NOMAD and COBRA were run for a North Anna 120% overpower case. The COBRA model used four 17 x 17 fuel assemblies to represent the core. COBRA also used as input the axial power distribution calculated by NOMAD. The system pressure, core flow rate, and inlet enthalpy were identical for both COBRA and NOMAD. Table 5-2 shows a comparison of the moderator enthalpy and moderator temperature distributions calculated by NOMAD and COBRA.

5.4 Axial Power Distribution

Axial power distribution comparisons between the Vepco NOMAD model and measurements are presented in Figures 5-7 through 5-16 for BOC, HZP, no xenon and BOC, HFP, equilibrium xenon conditions. Representative axial power distributions comparisons are shown for North Anna 1 Cycle 2, North Anna 1 Cycle 3, North Anna 1 Cycle 4, North Anna 2 Cycle 2, and Surry 1 Cycle 6.

The NOMAD predictions attempt to simulate the actual core conditions. However, NOMAD does not represent the spacer grids in order to increase calculation efficiency. The accuracy which is compromised is insignificant.

5.5 Differential and Integral Rod Worths

The Vepco NOMAD model predictions and startup physics measurements for differential and integral control rod bank worths for B-bank are compared in Figures 5-17 through 5-24. B-bank was the rod swap reference bank for each of these cycles and its worth was measured by boron dilution.

In addition, NOMAD results for Banks A through D moving in overlap are compared to the Vepco FLAME model in Figures 5-25 through 5-32. No measurements were performed at these conditions.

Integral control rod bank worths were calculated for banks measured by rod swap using a 1-D/2-D synthesis technique. NOMAD performed a critical boron search with the reference bank fully inserted. Another bank was then fully inserted, and NOMAD performed a criticality search on the reference bank position and calculated the flux-squared sharings. These flux-squared sharings were then used to determine a weighted average of the PDQ07 Discrete bank worths for the bank inserted alone and the bank inserted with the reference bank. These synthesized bank worths and the measured worths are compared in Table 5-3. The critical position of the reference bank predicted by NOMAD for each bank fully inserted is compared to measurement in Table 5-4.

5.6 Load Follow Maneuver Simulation

NOMAD's load follow simulation capability has been verified by comparison to three sets of measured data for load follow type cases. The first set of data consists of hourly delta-I readings and two critical boron measurements from a 70% load reduction test performed near the end of North Anna Unit 1 Cycle 2. The power and D-bank history for this case is listed in Table 5-5. Figures 5-33 and 5-34 compare NOMAD results to the measured delta-I and critical boron concentrations.

Two additional sets of measured data were recorded near the end of North Anna Unit 1 Cycle 3 during power escalations following reactor trips. The first incident occurred on April 16-20, 1982, and the second on April 30 - May 2, 1982. The power and D-bank histories for these two cases are given in Tables 5-6 and 5-7. The negative times listed in Table 5-6 are simply the number of hours before the comparisons in Figures 5-35 and 5-36 begin. The data prior to 0.0 hours was not plotted because it was at HFP, equilibrium conditions (delta-I is virtually constant) or low power levels (no delta-I data available). Hourly readings of delta-I and eight critical boron measurements were taken during the first case. Results from the NOMAD simulation are plotted versus these data in Figures 5-35 and 5-36, respectively. During the second case, both ex-core delta-I readings and INCORE axial offset measurements were performed, since delta-I cannot be measured accurately at low power levels. (The INCOREs were performed on only a limited number of assemblies each time.) The delta-I readings have been converted to axial offsets in order to compare NOMAD results to both types of data in Figure 5-37. Figure 5-38 plots the NOMAD critical boron concentrations versus thirteen measured values for this case.

All three of these cases were simulated with NOMAD using the deep control rod insertion model described in Section 3.6. Buckling coefficient searches to normalize to FLAME were performed at several different conditions for each case. The final column of Tables 5-5 through 5-7 lists the normalization conditions for the buckling coefficients which were used at each step of the calculation. For example, in Table 5-5, the buckling coefficients normalized at 100% power and D-bank at 228 steps are used for the first three steps. The remainder of the calculation is performed with the buckling coefficients normalized at 30% power and D-bank at 139 steps.

The necessity of this model can be seen in Table 5-8, which compares FLAME to NOMAD results obtained without any compensation for deep rod insertion. Notice the difference between the axial offset results when the rods are deep in the core. These parallel calculations are a simplified simulation of the history listed in Table 5-7. The time 0.0 hours in Table 5-8 corresponds approximately to 22.63 hours on 4/30/82 in Table 5-7.

5.7 FAC Analysis

Three case and eighteen case FAC analyses were performed with the Vepco NOMAD model for North Anna 1 Cycle 4 and North Anna 2 Cycle 2. The cases were performed for the following conditions:

BOL (150 MWD/MTU)	Base Load Depletion	100%-70%-100% 100%-50%-100% 100%-30%-100%
85% EOL	Base Load Depletion	100%-70%-100% 100%-50%-100% 100%-30%-100%
85% EOL	Load Follow Depletion	100%-70%-100% 100%-50%-100% 100%-30%-100%

These nine cases were performed twice, with IDCNTL set to +1 and -1. The three case analyses consisted of the 100%-50%-100% cases with IDCNTL= -1.

The Vepco NOMAD model results were found to be consistent with the results from an accepted and verified vendor model which has been used in the design and licensing of the Surry and North Anna reactors. Both models indicated minor technical specification violations near the core bottom in the three case analyses and no violations in the eighteen case analyses.

Figures 5-39 through 5-42 show the NOMAD results for the eighteen case analyses. The $F_Q(Z)$ plots in Figures 5-40 and 5-42 contain an uncertainty factor of 10.9%.

TABLE 5-1

REACTIVITY COEFFICIENTS COMPARISON

Differential Boron Worth, BOC HZP (pcm/ppm)

Unit/Cycle	NOMAD	PDQ07 Discrete	Measured	% Difference[1]
N1C2	-9.15	-9.10	-8.88	3.04
N1C3	-8.15	-8.08	-8.54	-4.57
N1C4	-7.70	-8.04	-8.25	-6.67
N2C2	-8.97	-8.91	-8.46	6.03
S1C6	-8.27	-8.31	-8.78	-5.81
S1C7	-8.43	-8.44	-8.44	-0.12

Differential Boron Worth, EOC HFP (pcm/ppm)

Unit/Cycle	NOMAD	PDQ07 Discrete	% Difference[2]
N1C2	-9.06	-9.43	-3.92
N1C3	-8.17	-8.47	-3.54
N1C4	-7.90	-8.62	-8.35
N2C2	-8.91	-9.40	-5.21
S1C6	-8.81	-9.31	-5.37
S1C7	-8.93	-9.16	-2.51

Isothermal Temperature Coefficient BOC, HZP (pcm/^{o}F)

Unit/Cycle	NOMAD	PDQ07 One Zone	Measured	Difference[3]
N1C2	-5.07	-3.87	-2.36	-2.71
N1C3	-3.99	-3.40	-4.36	0.37
N1C4	-4.90	-3.52	-4.92	0.02
N2C2	-5.29	-3.27	-2.27	-3.02
S1C6	-4.28	-3.79	-2.32	-1.96
S1C7	-6.49	-5.68	-5.85	-0.64

[1] % Difference = (NOMAD - Measured) / Measured x 100

[2] % Difference = (NOMAD - PDQ07)/PDQ07 x 100

[3] Difference = NOMAD - Measured

TABLE 5-2

COMPARISON OF NOMAD AND COBRA
MODERATOR ENTHALPY AND TEMPERATURE DISTRIBUTIONS

Position (inches)	Enthalpy (BTU/lbm) NOMAD	Enthalpy (BTU/lbm) COBRA	Moderator Temperature (°F) NOMAD	Moderator Temperature (°F) COBRA
4	544.9	545.1	548.4	548.0
8	546.8	546.9	549.9	549.5
12	549.1	549.3	551.8	551.4
16	551.8	552.0	554.0	553.5
20	554.7	554.8	556.3	555.8
24	557.7	557.8	558.7	558.2
28	560.8	560.9	561.1	560.6
32	563.9	564.0	563.6	563.0
36	567.0	567.1	566.0	565.5
40	570.2	570.2	568.4	567.8
44	573.3	573.4	570.9	570.3
	576.5	576.6	573.3	572.6
52	579.6	579.7	575.7	575.0
56	582.8	582.9	578.1	577.3
60	586.0	586.1	580.4	579.7
64	589.2	589.3	582.8	582.1
68	592.4	592.5	585.2	584.5
72	595.7	595.8	587.5	586.9
76	598.9	599.0	589.9	589.3
80	602.2	602.3	592.2	591.7
84	605.5	605.6	594.6	594.1
88	608.8	608.9	596.9	596.5
92	612.2	612.3	599.2	598.9
96	615.5	615.6	601.5	601.3
100	618.9	619.0	603.8	603.7
104	622.3	622.3	606.1	606.4
108	625.6	625.7	608.4	608.3
112	629.1	629.1	610.7	610.6
116	632.5	632.5	612.9	612.9
120	635.8	635.9	615.2	615.1
124	639.1	639.2	617.3	617.4
128	642.4	642.4	619.4	619.5
132	645.3	645.4	621.3	621.4
136	648.0	648.0	623.0	623.0
140	650.1	650.1	624.4	624.3
144	651.6	651.5	625.3	625.2

TABLE 5-3

ROD SWAP COMPARISON, PART 1

Integral Bank Worths (pcm)

North Anna Unit 1 Cycle 3

Bank	NOMAD/PDQ	Measured	% Difference
D	1048	1089	-3.76
C	839	777	7.98
A	620	722	-14.13
SB	1006	919	9.47
SA	1096	1238	-11.47

North Anna Unit 1 Cycle 4

Bank	NOMAD/PDQ	Measured	% Difference
D*	N/A	N/A	N/A
C	808	843	-4.15
A	479	562	-14.77
SB	980	1023	-4.20
SA	997	1094	-8.87

North Anna Unit 2 Cycle 2

Bank	NOMAD/PDQ	Measured	% Difference
D	1010	1015	-0.49
C	780	757	3.04
A	757	812	-6.77
SB	713	664	7.38
SA	912	948	-3.80

Surry Unit 1 Cycle 6

Bank	NOMAD/PDQ	Measured	% Difference
D	1228	1234	-0.49
C	819	815	0.49
A	538	551	-2.36
SB	1018	1013	0.49
SA	1093	1137	-3.87

*D-bank worth was measured by a combination of rod swap & dilution.

TABLE 5-4

ROD SWAP COMPARISON, PART 2

Reference Bank Critical Position (Steps)

North Anna Unit 1 Cycle 3

Bank	NOMAD	Measured	Difference
D	143	143	0
C	117	103	14
A	97	97	0
SB	143	119	24
SA	158	170	-12

North Anna Unit 1 Cycle 4

Bank	NOMAD	Measured	Difference
D*	N/A	N/A	N/A
C	154	163	-9
A	114	122	-8
SB	182	190	-8
SA	186	201	-15

North Anna Unit 2 Cycle 2

Bank	NOMAD	Measured	Difference
D	186	195	-9
C	171	168	3
A	167	175	-8
SB	166	155	11
SA	181	189	-8

Surry Unit 1 Cycle 6

Bank	NOMAD	Measured	Difference
D	179	175	4
C	123	110	13
A	93	83	10
SB	154	138	16
SA	166	157	9

*D-bank worth was measured by a combination of rod swap & dilution.

TABLE 5-5

N1C2 70% LOAD REDUCTION TEST
POWER AND D-BANK HISTORY

Time (Hours)	Power (%)	D-bank (steps)	Buckling Normalization Power / D-bank
00.00	98.8	228	100% / 228
01.00	82.9	190	
02.00	56.2	186	
03.00	32.8	150	30% / 139
04.00	29.5	152	
05.00	29.2	166	
06.00	29.2	162	
07.00	29.6	157	
08.00	30.0	156	
09.00	30.6	145	
10.00	30.5	143	
11.00	30.3	145	
12.00	30.2	141	
13.00	30.2	140	
14.00	29.9	138	
15.00	29.5	137	
16.00	29.5	137	
17.00	28.9	136	
18.00	29.3	136	
19.00	29.0	136	
20.00	27.9	136	
21.00	28.4	137	
22.00	28.4	137	
23.00	27.5	145	
24.00	27.3	151	
25.00	27.2	155	
26.00	27.2	157	
27.00	26.7	160	
28.00	26.7	160	
29.00	26.7	160	
30.00	26.7	160	

TABLE 5-6

N1C3 SHUTDOWN/RETURN TO POWER CASE 1
POWER AND D-BANK HISTORY

Time (Hours)	Power (%)	D-bank (steps)	Buckling Normalization Power / D-bank
-23.00	100.9	215	100% / 228
-22.00	100.4	215	
-21.00	100.6	215	
-20.00	100.5	215	
-19.00	100.6	215	
-18.00	100.8	215	
-17.00	100.7	215	
-16.00	100.6	215	
-15.00	100.5	215	
-14.00	100.0	215	
-13.00	99.9	214	
-12.00	100.6	215	
-11.00	100.5	215	
-10.00	100.6	215	
-9.00	0.0	0	
-8.00	0.0	0	
-7.00	0.0	0	
-6.00	0.2	47	2% / 95
-5.00	1.2	156	5% / 182
-4.00	1.9	179	
-3.00	0.0	0	
-2.00	0.2	185	
-1.00	4.9	193	
0.0	17.7	197	
1.00	23.5	186	
2.00	27.8	160	30% / 160
3.00	29.7	160	
4.00	30.1	160	
5.00	30.2	160	
6.00	29.8	160	
7.00	30.7	160	
8.00	29.3	158	
9.00	29.2	151	
10.00	29.4	143	30% / 139
11.00	29.4	143	
12.00	29.4	142	

TABLE 5-6 (Continued)

13.00	30.9	141	
14.00	45.7	141	
15.00	48.9	144	
16.00	49.0	159	48% / 175
17.00	48.7	170	
18.00	48.4	178	
19.00	48.0	181	
20.00	47.5	182	
21.00	47.3	183	
22.00	47.6	184	
23.00	47.7	184	
24.00	47.3	182	
25.00	47.7	180	
26.00	47.7	178	
27.00	48.8	177	
28.00	49.0	175	
29.00	49.0	172	
30.00	48.4	170	
31.00	48.9	169	
32.00	48.4	168	
33.00	48.4	168	
34.00	47.7	168	
35.00	48.3	168	
36.00	48.4	167	
37.00	48.8	165	
38.00	55.6	172	
39.00	69.9	181	100% / 228
40.00	73.4	183	
41.00	73.2	183	
42.00	76.9	190	
43.00	87.7	205	
44.00	92.0	209	
45.00	99.6	211	
46.00	99.7	211	
47.00	100.2	211	
48.00	99.6	211	
49.00	99.8	211	
50.00	100.0	211	
51.00	100.1	211	
52.00	99.8	211	
53.00	100.0	211	
54.00	100.3	211	

TABLE 5-6 (Continued)

55.00	100.2	211	
56.00	100.2	211	
57.00	100.2	211	
58.00	0.0	0	
59.00	0.0	0	
60.00	0.0	0	
61.00	1.4	88	2% / 95
62.00	5.1	128	
63.00	26.9	200	5% / 182
64.00	28.3	182	30% / 160
65.00	28.1	172	
66.00	30.1	160	
67.00	30.0	160	
68.00	30.0	160	
69.00	29.8	160	
70.00	44.3	170	
71.00	50.4	161	
72.00	49.5	161	
73.00	47.2	161	
74.00	47.6	161	
75.00	47.2	161	
76.00	47.1	158	
77.00	47.3	158	

TABLE 5-7

N1C3 SHUTDOWN/RETURN TO POWER CASE 2
POWER AND D-BANK HISTORY

Date	Time (Hours)	Power (%)	D-bank (steps)	Buckling Normalization Power / D-bank
4 30 82	16.20	100.3	218	100% / 228
4 30 82	18.60	99.7	218	
4 30 82	18.85	99.8	218	
4 30 82	20.72	100.2	218	
4 30 82	21.50	95.2	209	
4 30 82	21.78	92.1	206	
4 30 82	22.06	88.3	200	
4 30 82	22.35	83.6	190	
4 30 82	22.63	75.8	180	75% / 182
4 30 82	23.15	66.0	171	
4 30 82	23.70	58.2	158	60% / 160
5 1 82	0.50	44.0	127	30% / 117
5 1 82	1.30	30.1	110	
5 1 82	2.08	18.2	110	
5 1 82	2.85	3.2	90	
5 1 82	3.87	2.3	145	60% / 160
5 1 82	4.70	2.0	162	
5 1 82	5.77	2.1	173	
5 1 82	6.78	2.3	177	
5 1 82	7.81	2.5	174	
5 1 82	8.84	2.2	169	
5 1 82	9.88	0.0	0	100% / 228
5 1 82	10.88	0.0	0	
5 1 82	11.90	0.0	0	
5 1 82	12.95	0.0	0	
5 1 82	13.20	0.0	91	5% / 81
5 1 82	13.53	0.0	91	
5 1 82	13.95	1.3	91	
5 1 82	14.27	1.3	91	
5 1 82	14.95	1.7	81	
5 1 82	15.73	1.7	81	
5 1 82	15.95	1.3	69	
5 1 82	16.85	1.8	61	
5 1 82	17.58	1.8	61	
5 1 82	17.85	1.8	59	

TABLE 5-7 (Continued)

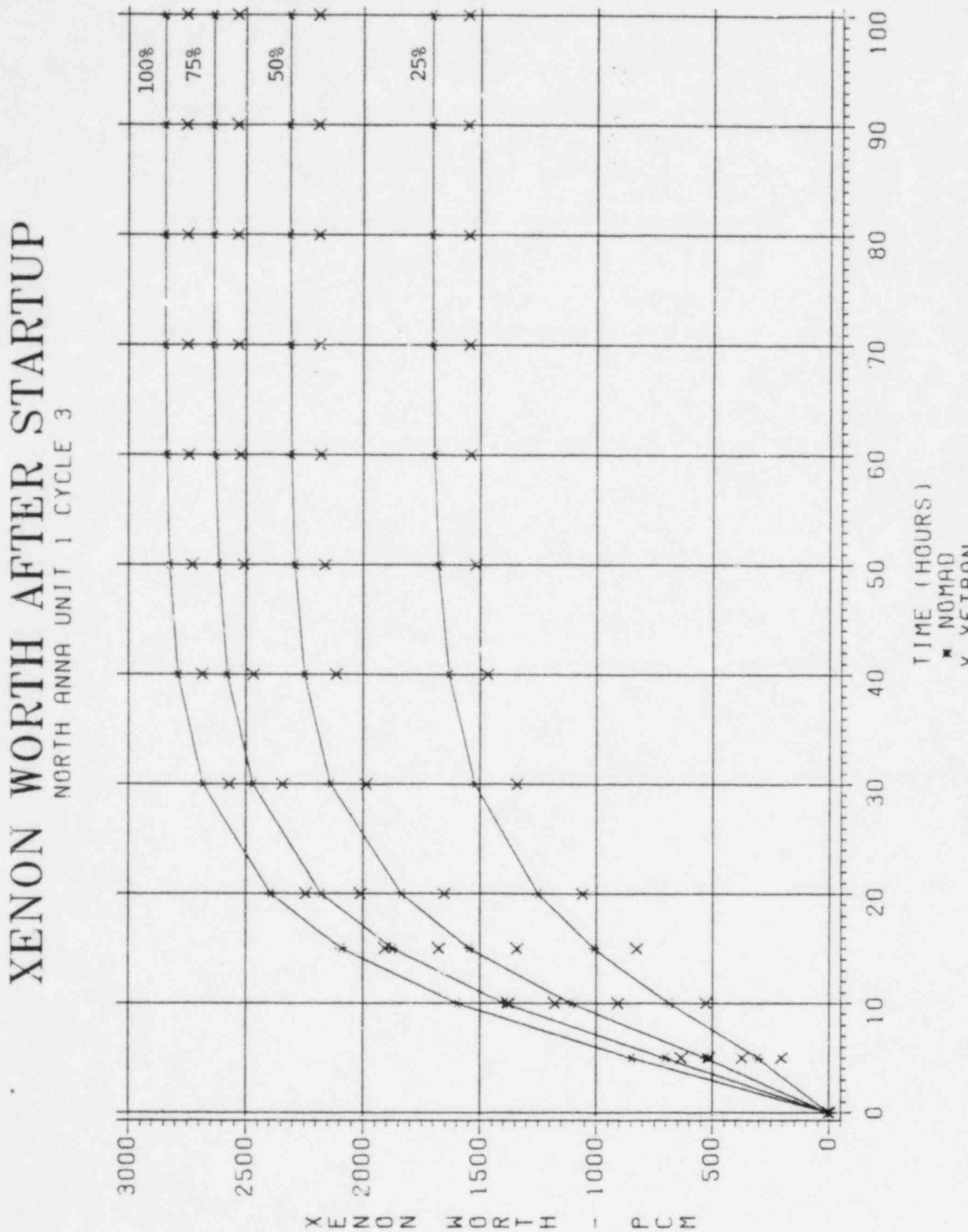
5	1 82	18.95	1.6	59	
5	1 82	19.95	2.1	59	
5	1 82	21.00	1.8	45	2% / 45
5	1 82	22.07	1.8	45	
5	1 82	23.07	1.7	45	
5	1 82	23.97	2.1	45	
5	2 82	1.00	2.2	45	
5	2 82	2.02	2.3	45	
5	2 82	3.07	2.2	45	
5	2 82	3.95	2.4	45	
5	2 82	4.37	2.4	45	
5	2 82	4.98	2.2	45	
5	2 82	6.01	2.3	45	
5	2 82	6.35	2.3	45	
5	2 82	6.95	2.3	45	
5	2 82	8.00	2.8	46	
5	2 82	8.17	7.3	50	
5	2 82	8.53	7.3	50	
5	2 82	9.05	7.3	64	
5	2 82	9.17	17.6	63	
5	2 82	9.44	22.6	63	
5	2 82	10.14	29.8	68	
5	2 82	10.55	29.8	68	
5	2 82	11.00	31.8	69	
5	2 82	12.25	31.8	71	
5	2 82	13.05	31.8	71	
5	2 82	13.25	29.1	71	
5	2 82	14.22	39.9	108	30% / 117
5	2 82	15.19	51.6	155	60% / 160
5	2 82	16.21	64.8	174	75% / 182
5	2 82	17.31	67.1	181	
5	2 82	18.21	68.7	185	
5	2 82	19.21	69.2	185	
5	2 82	20.21	73.9	193	100% / 228
5	2 82	21.24	83.0	201	
5	2 82	22.24	91.1	208	
5	2 82	23.24	96.1	214	
5	2 82	23.87	98.7	214	

TABLE 5-8

COMPARISON OF FLAME AND UNCORRECTED NOMAD RESULTS
FOR N1C3 SHUTDOWN/RETURN TO POWER CASE 2

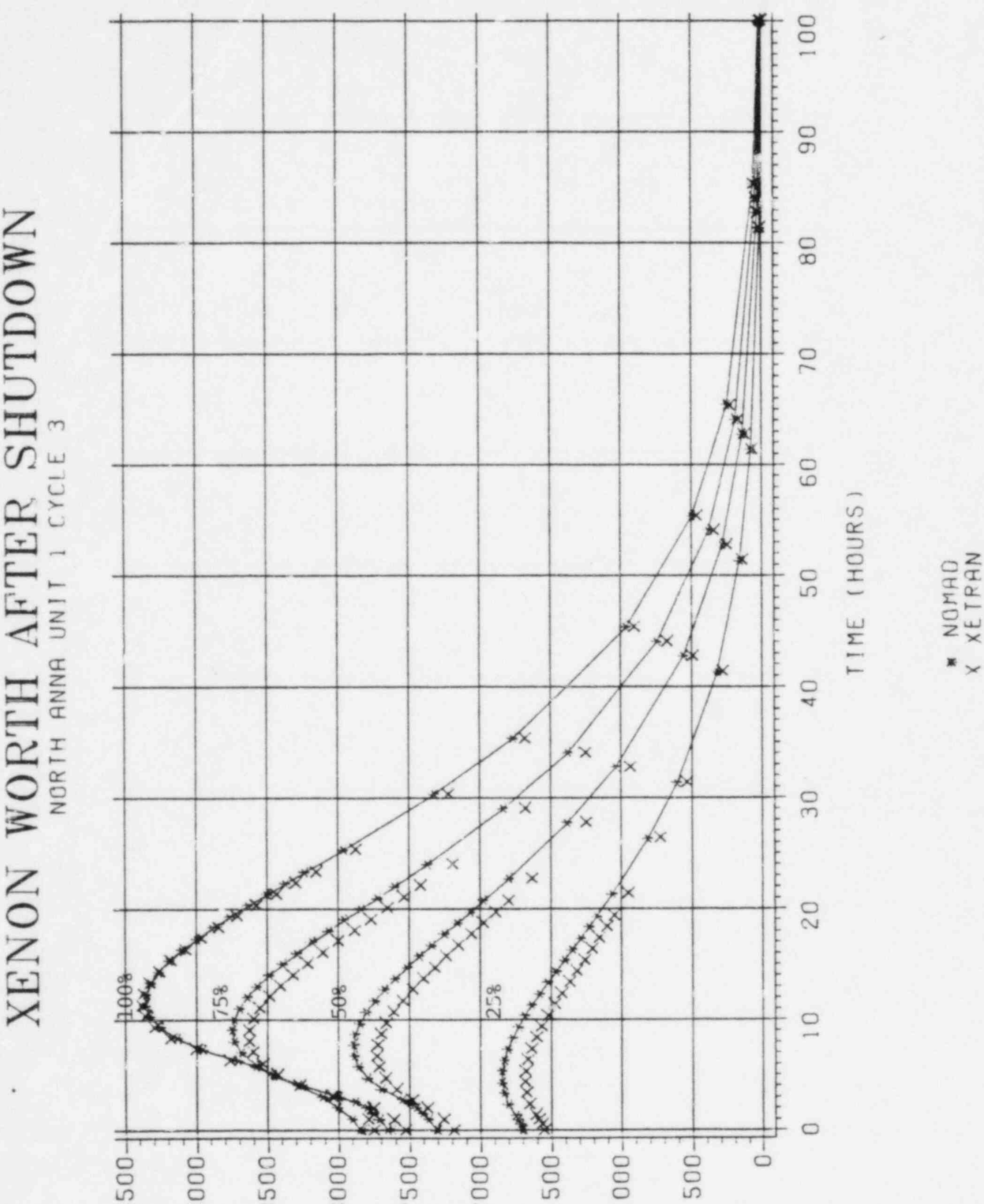
Time (Hours)	Power (%)	D-bank (steps)	Boron Conc. (ppm)	Axial Offset(%) FLAME	NOMAD
0.00	75.0	182	100	-5.60	-5.64
3.00	30.0	110	120	2.09	-14.89
5.35	2.5	167	125	53.18	32.68
12.39	0.0	0	63	78.21	73.77
15.71	1.5	81	100	70.68	49.42
20.36	2.2	45	200	57.35	27.63
34.67	25.0	67	300	20.91	-7.92
40.72	65.0	175	320	-4.23	-6.99

FIGURE 5-1



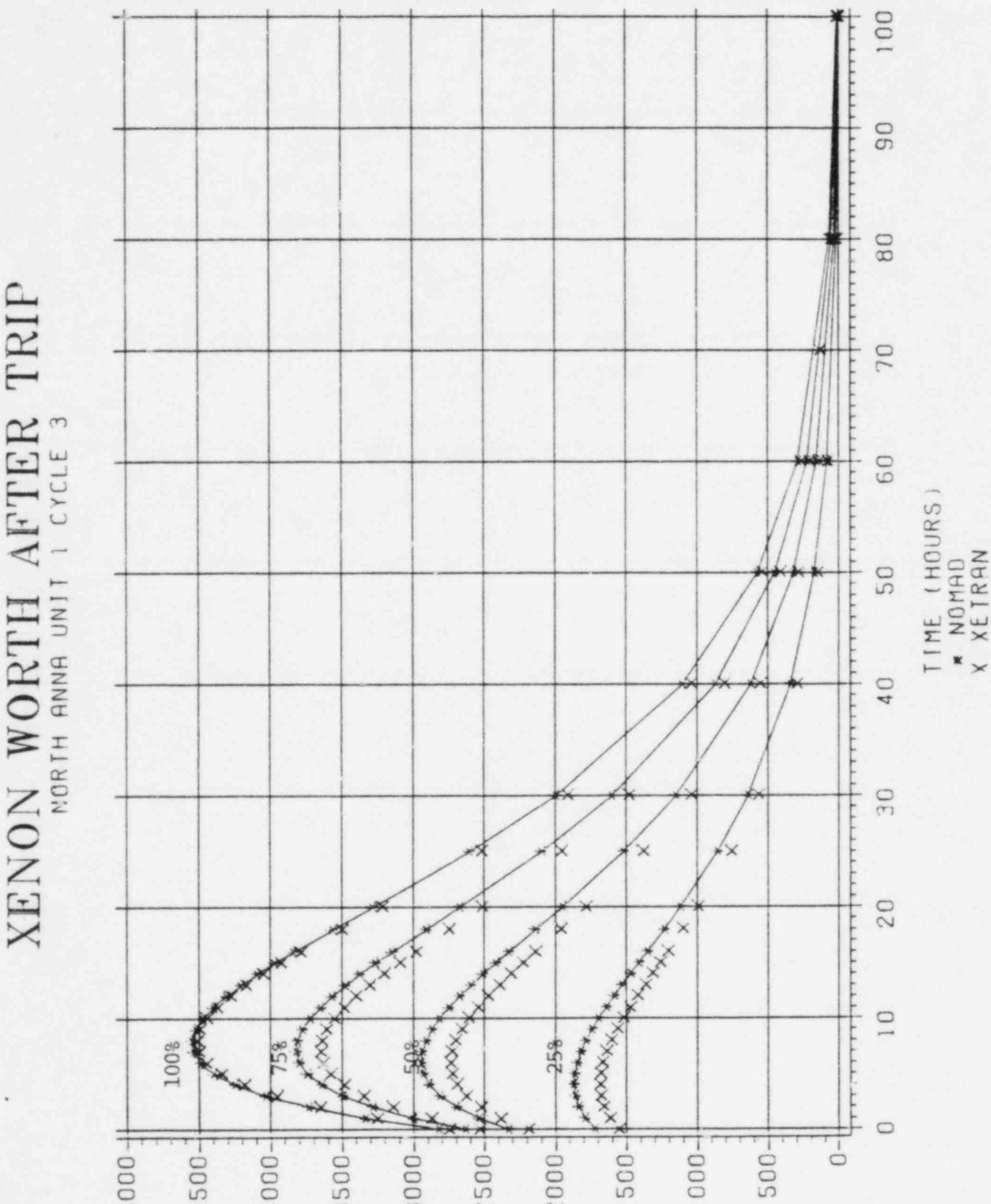
XENON WORTH AFTER STARTUP
NORTH ANNA UNIT 1 CYCLE 3
100%
75%
50%
25%
XENON WORTH - PCM
TIME (HOURS)
NOMAD
X XETRAN

FIGURE 5-2



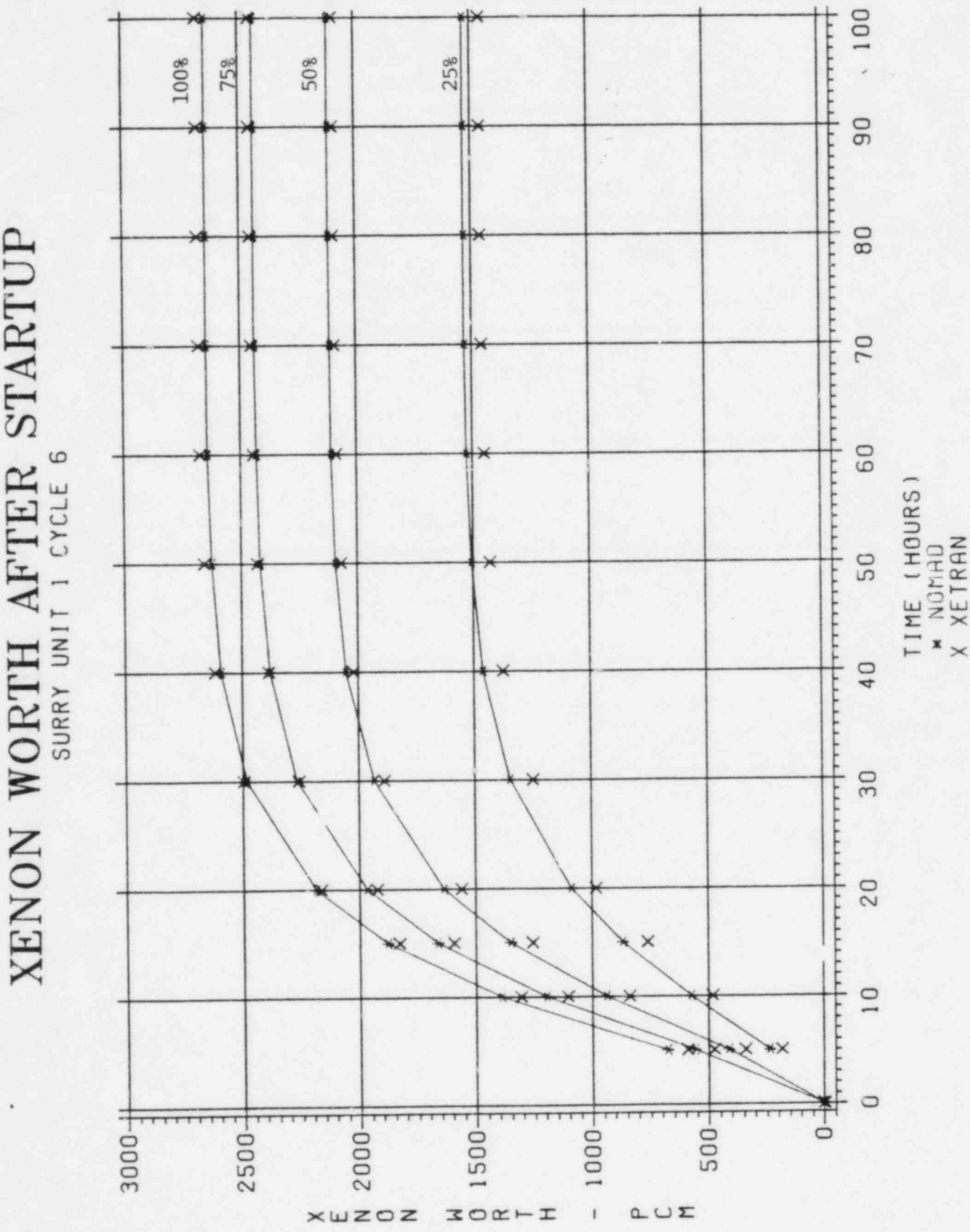
XENON WORTH AFTER SHUTDOWN
NORTH ANNA UNIT 1 CYCLE 3
XENON WORTH - PCM
4500
4000
3500
3000
2500
2000
1500
1000
500
0
100%
75%
50%
25%
0
10
20
30
40
50
60
70
80
90
100
TIME (HOURS)
* NOMAD
X XETRAN

FIGURE 5-3



XENON WORTH AFTER TRIP
NORTH ANNA UNIT 1 CYCLE 3
XENON WORTH - PCM
TIME (HOURS)
* NOMAD
X XETRAN
100%
75%
50%
25%
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
0
10
20
30
40
50
60
70
80
90
100

FIGURE 5-4


XENON WORTH AFTER STARTUP
SURRY UNIT 1 CYCLE 6
100%
75%
50%
25%
XENON WORTH - PCM
3000
2500
2000
1500
1000
500
0
0
10
20
30
40
50
60
70
80
90
100
TIME (HOURS)
* NOMAD
X XETRAN

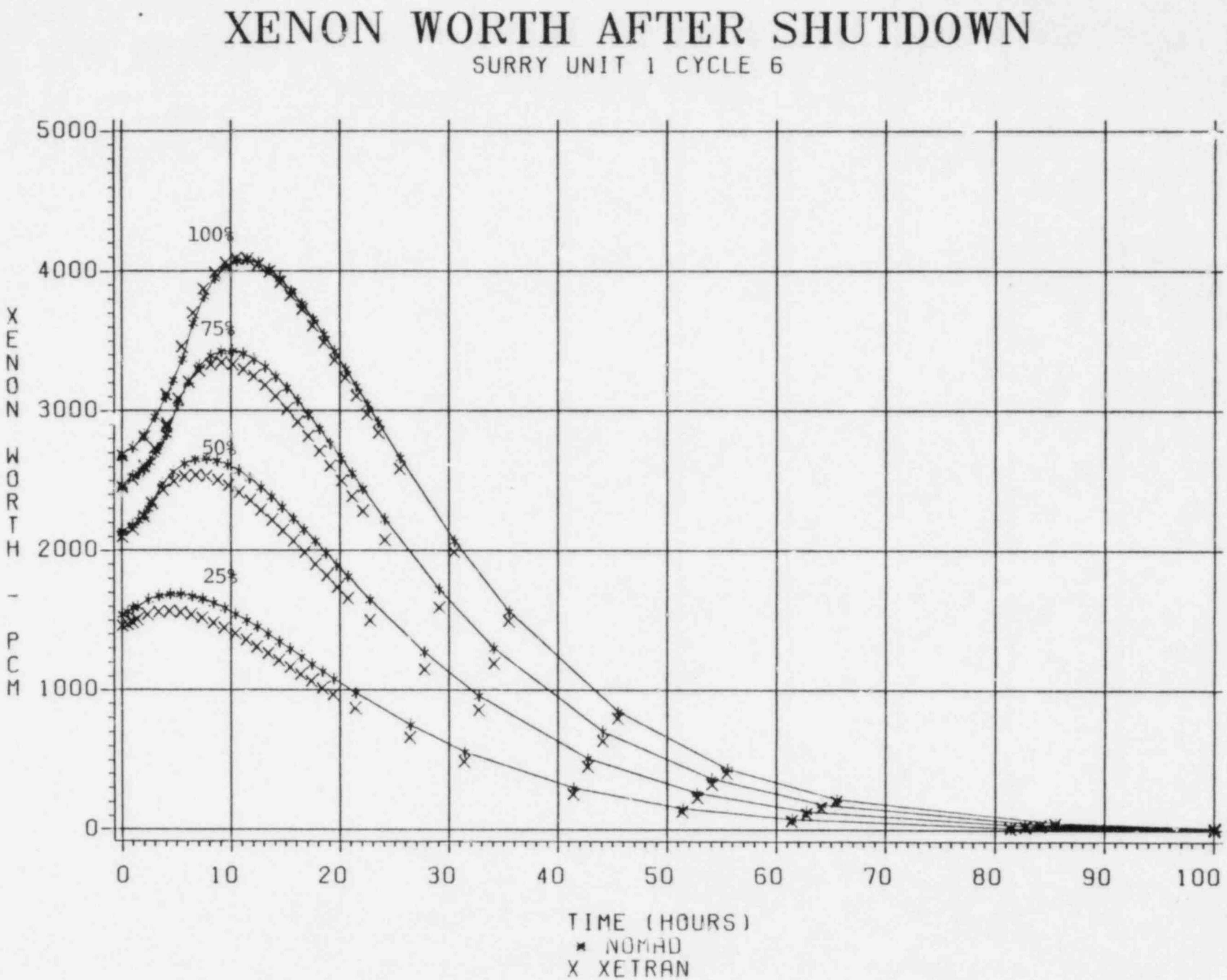
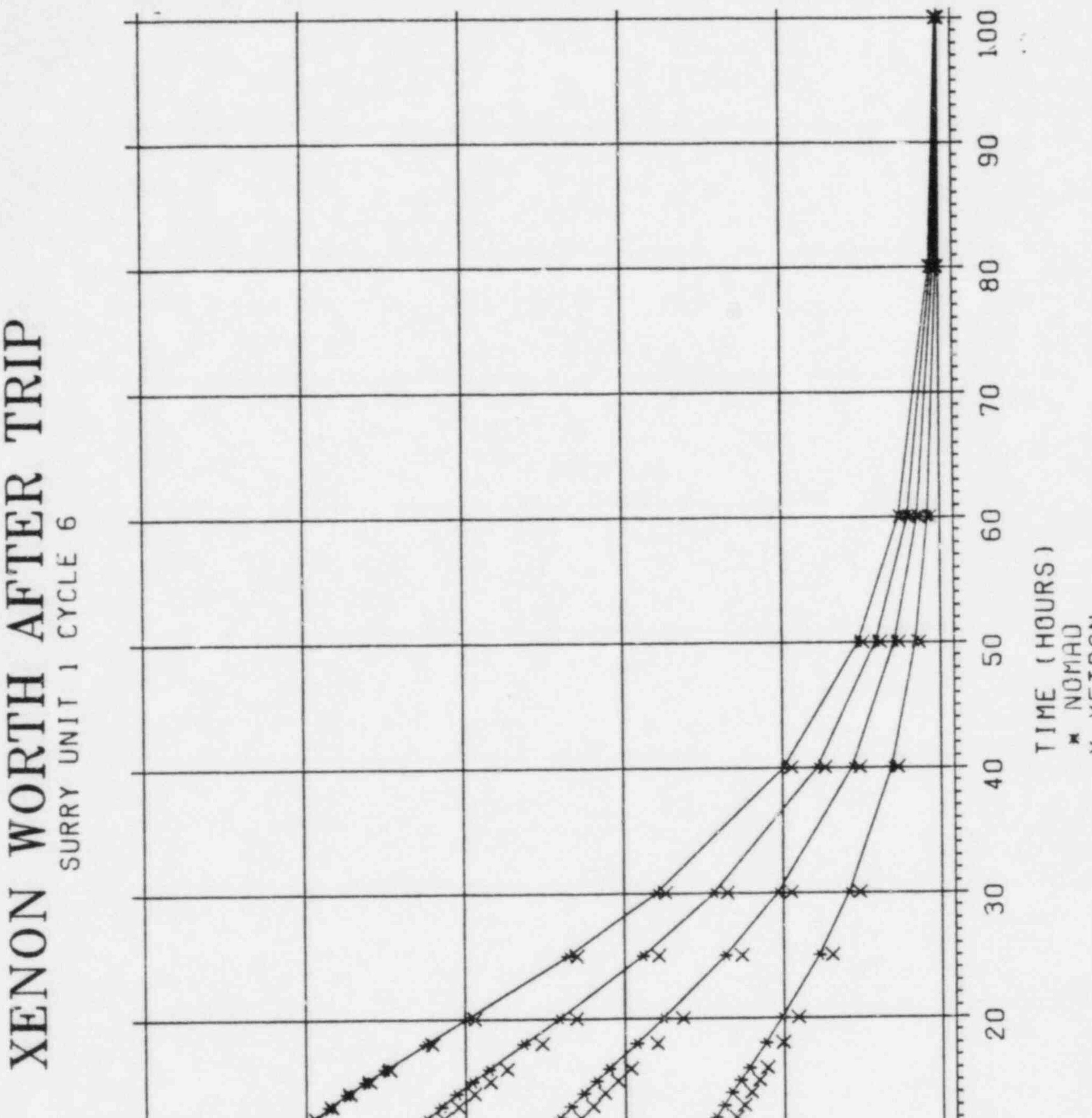

XENON WORTH AFTER SHUTDOWN
SURRY UNIT 1 CYCLE 6
100%
75%
50%
25%
5000
4000
3000
2000
1000
0
XENON WORTH - PCM
0
10
20
30
40
50
60
70
80
90
100
TIME (HOURS)
* NOMAD
X XETRAN

FIGURE 5-5

FIGURE 5-6


XENON WORTH AFTER TRIP
SURRY UNIT 1 CYCLE 6
XENON WORTH - PCM
5000
4000
3000
2000
1000
0
TIME (HOURS)
0
10
20
30
40
50
60
70
80
90
100
100%
75%
50%
25%
* NOMAD
X XETRAN

AXIAL POWER COMPARISON

N1C2 HZP BOC

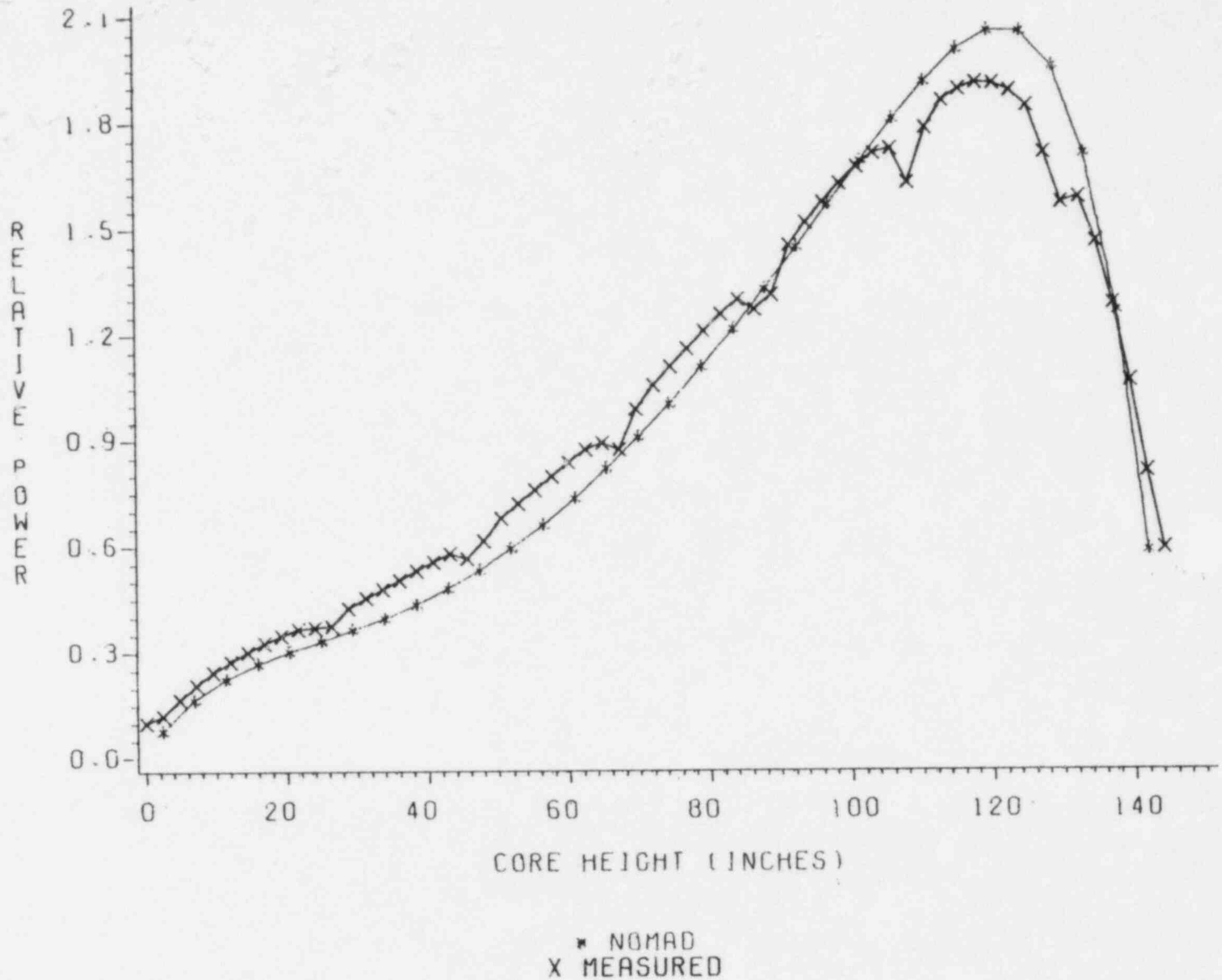


FIGURE 5-7

AXIAL POWER COMPARISON

N1C2 HFP ARO EQ. XE. BOC

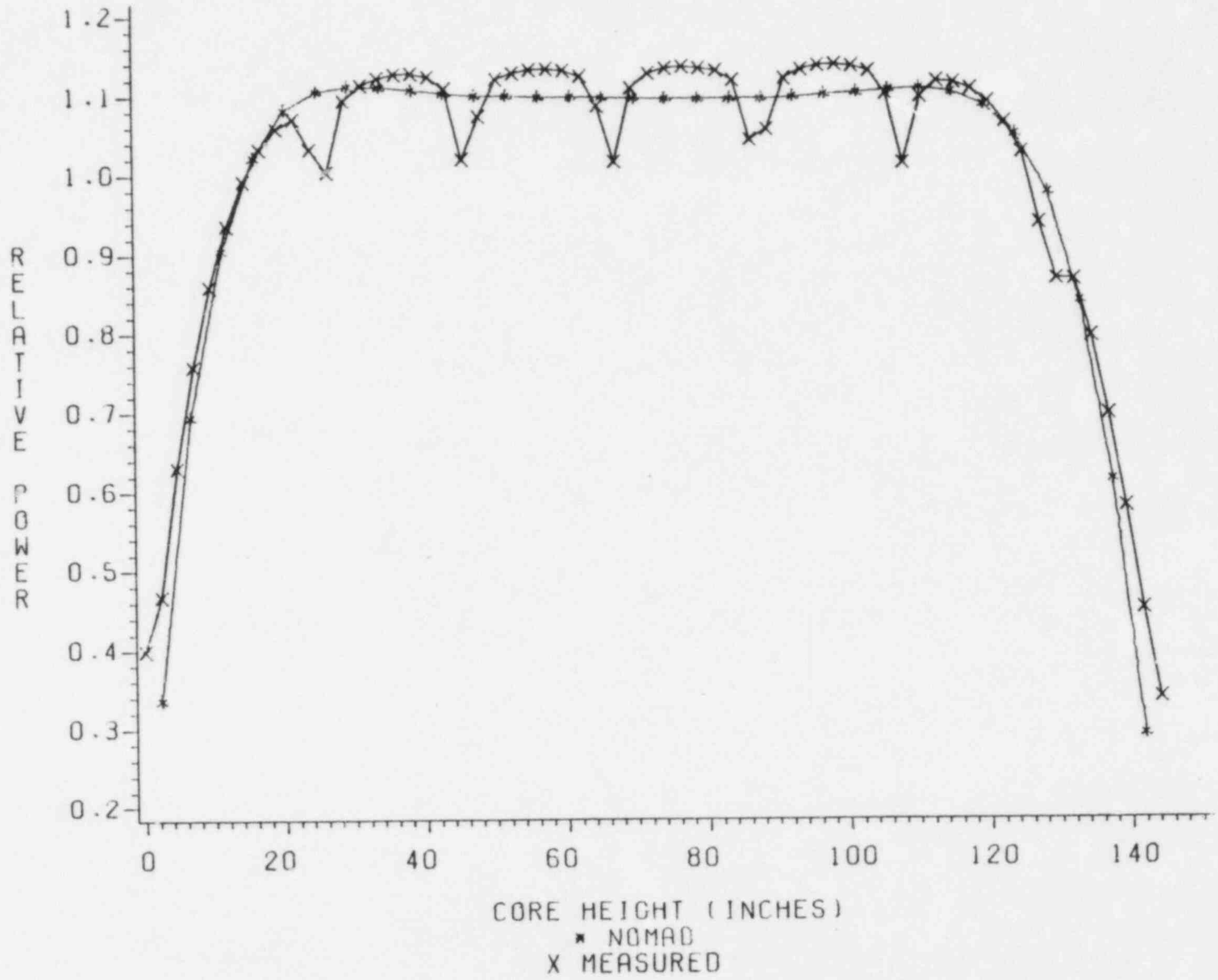
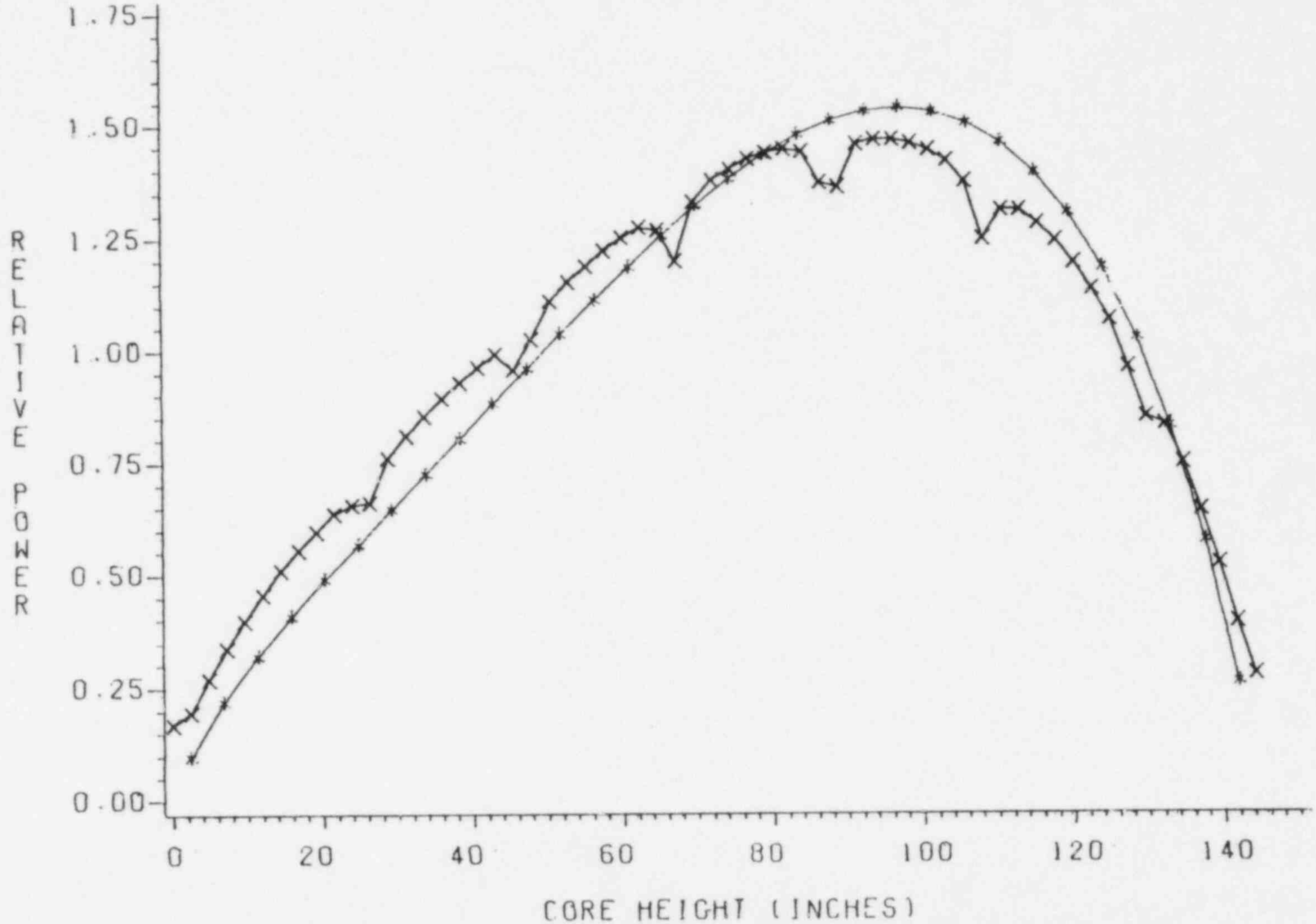


FIGURE 5-8

AXIAL POWER COMPARISON

N1C3 HZP BOC



* NOMAD
X MEASURED

FIGURE 5-9

AXIAL POWER COMPARISON

N1C3 HFP ARO EQ. XE. BOC

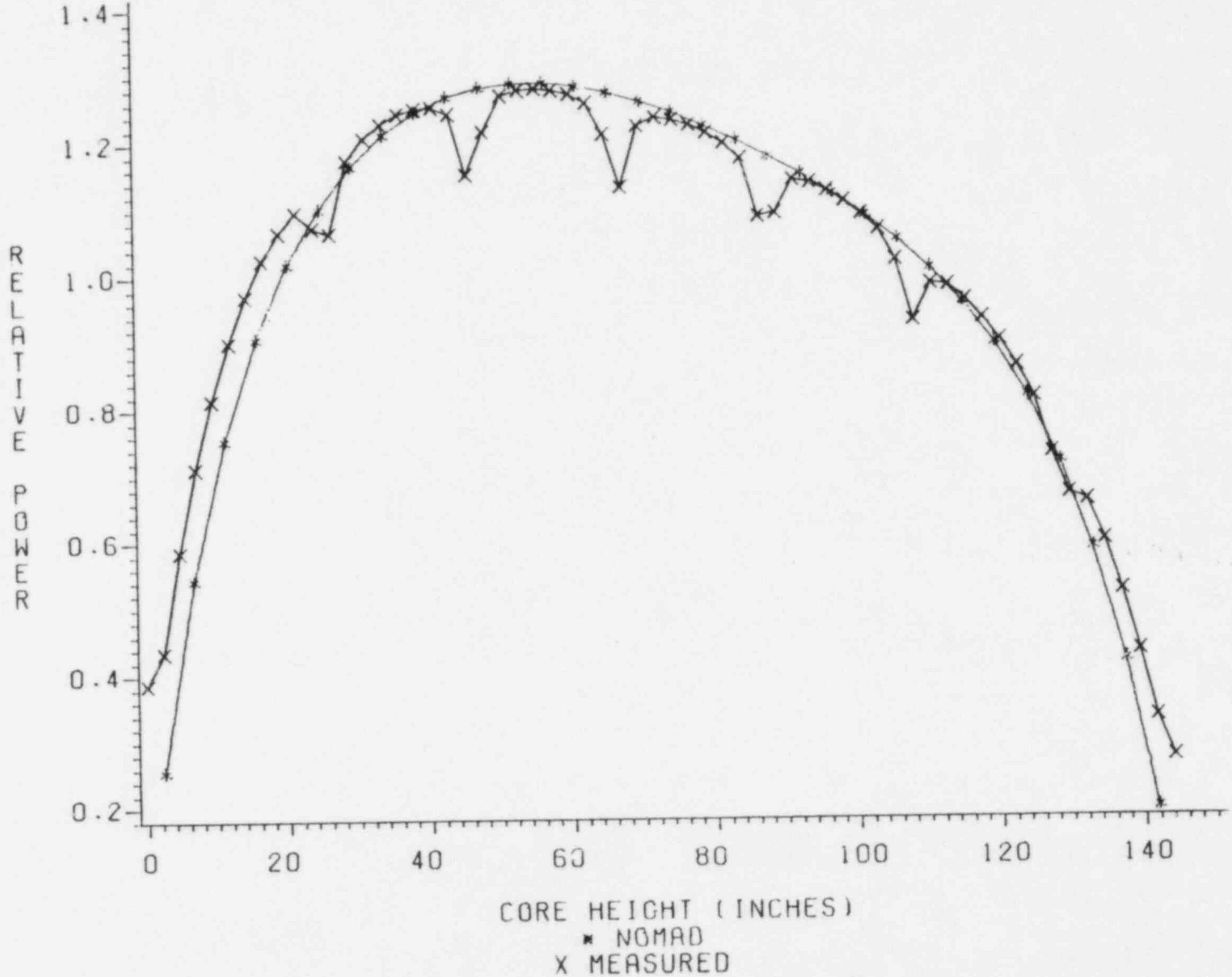


FIGURE 5-10

AXIAL POWER COMPARISON

N1C4 HZP BOC

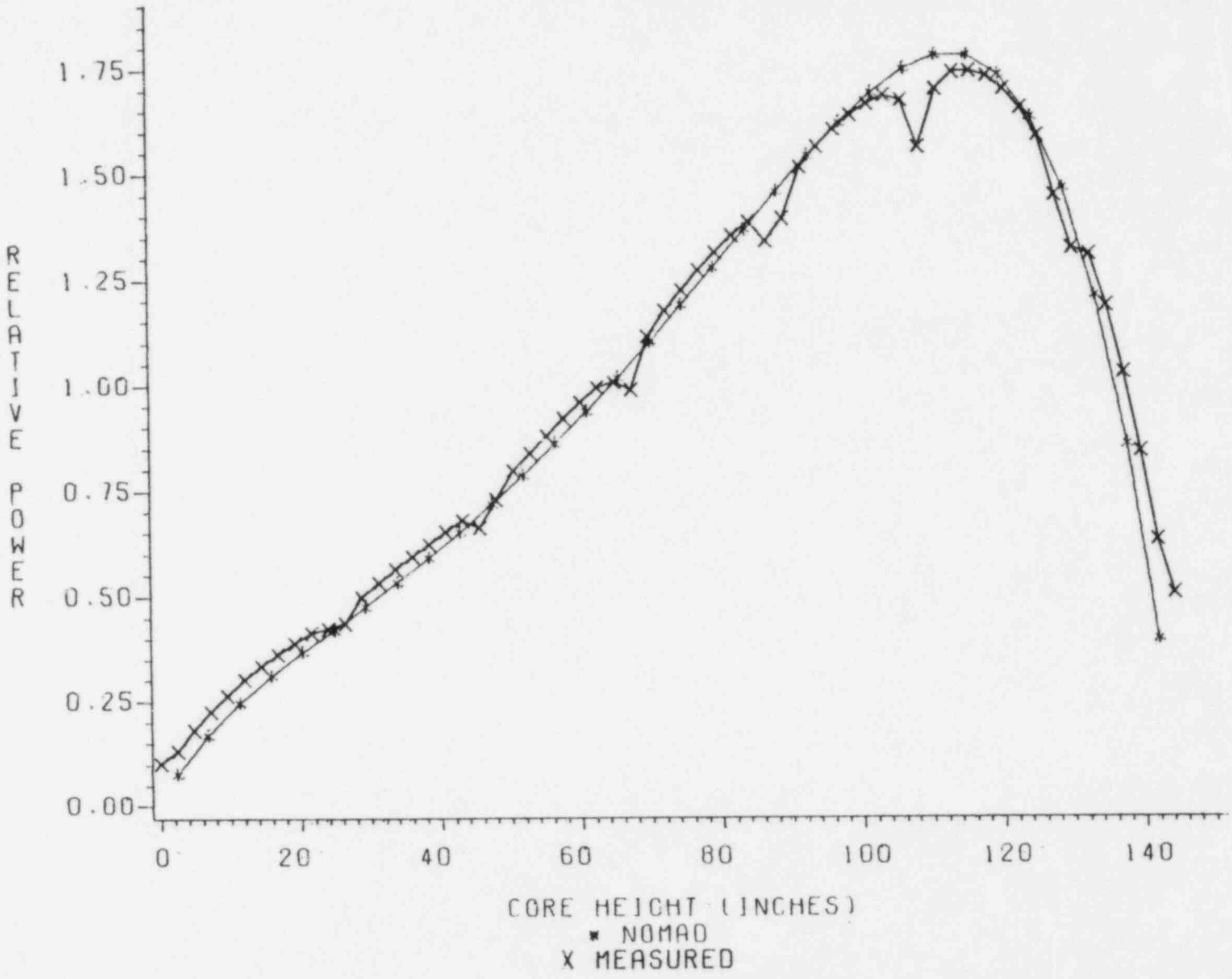


FIGURE 5-11

AXIAL POWER COMPARISON

N1C4 HFP ARO EQ. XE. BOC

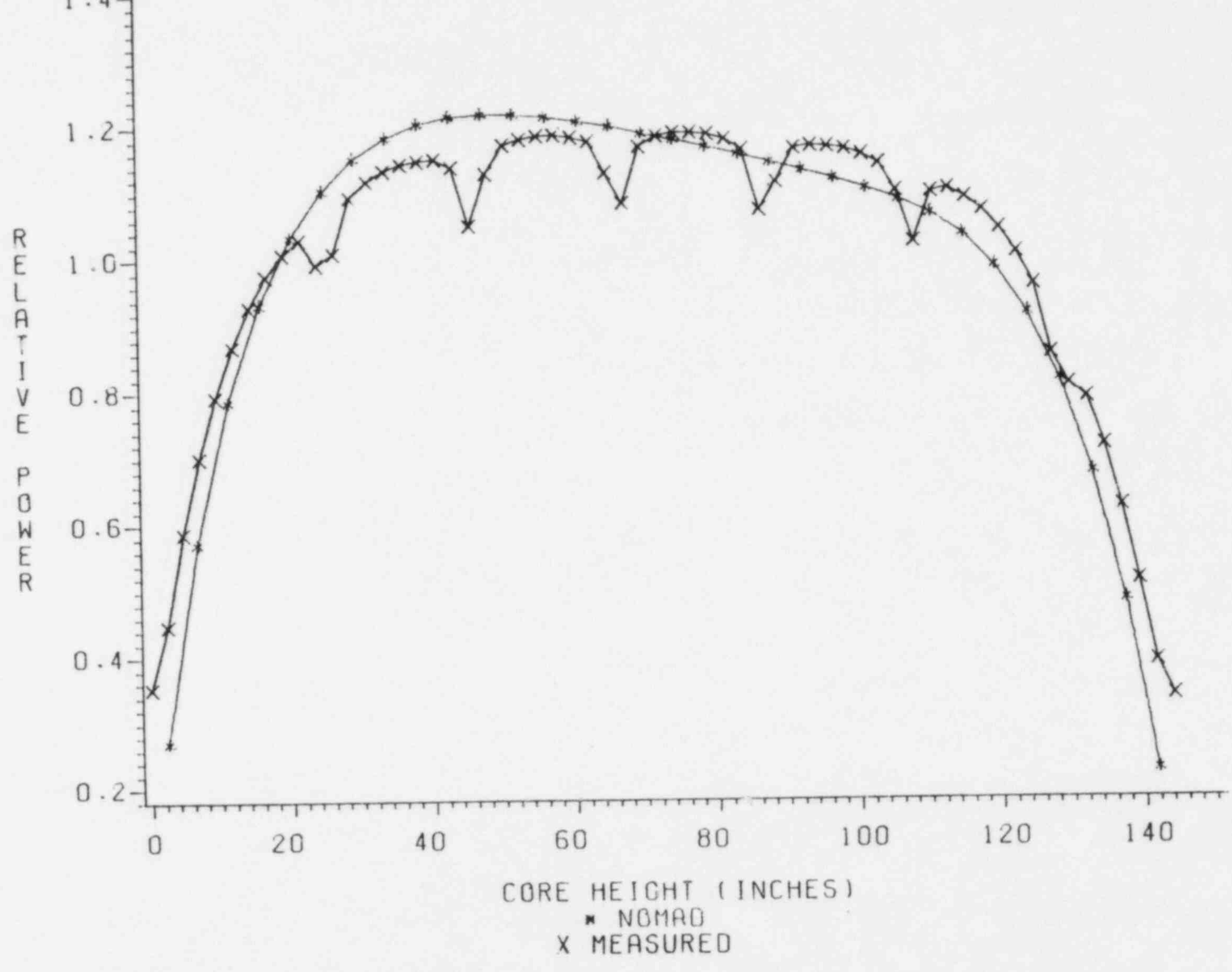


FIGURE 5-12

AXIAL POWER COMPARISON

N2C2 HZP BOC

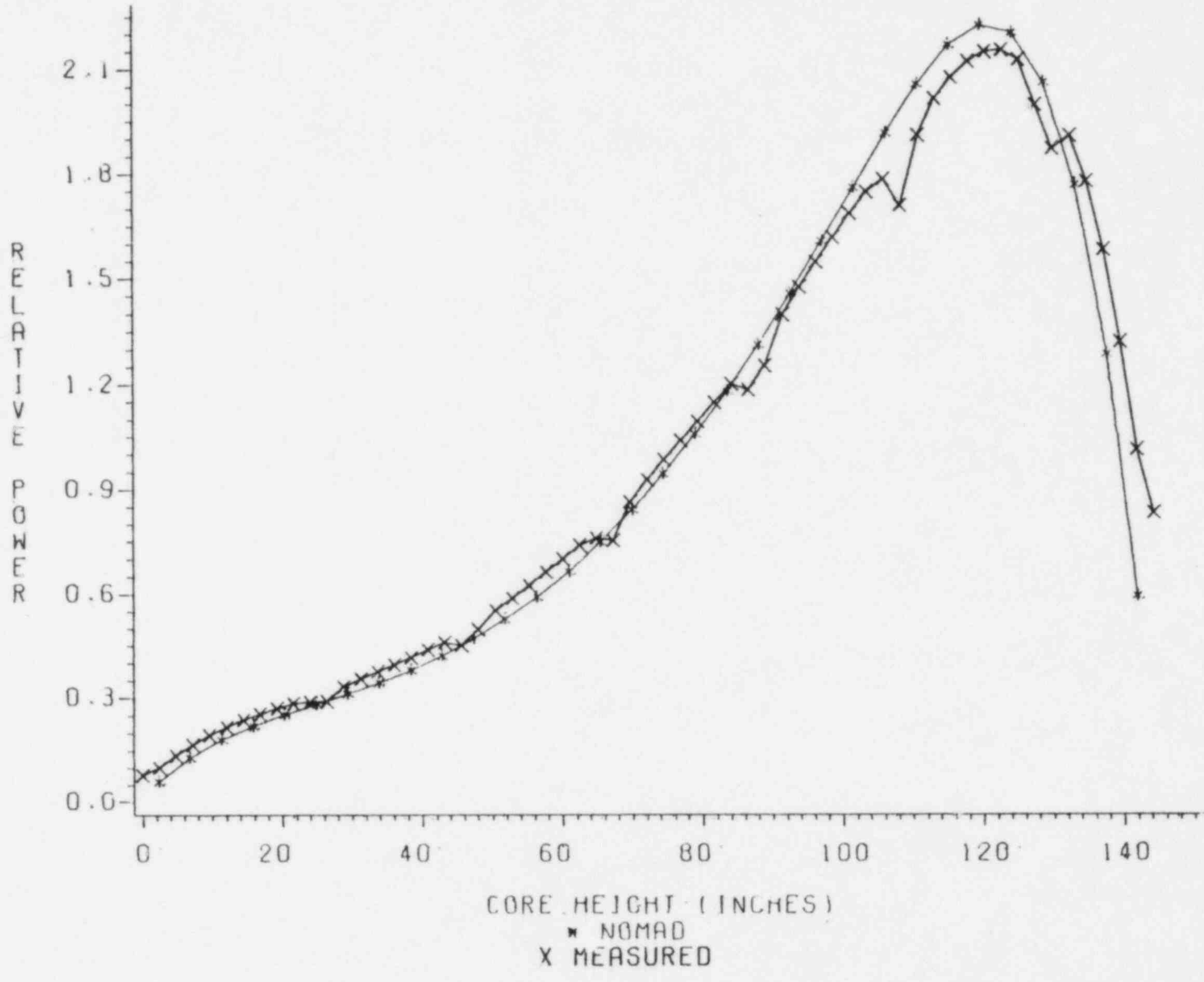


FIGURE 5-13

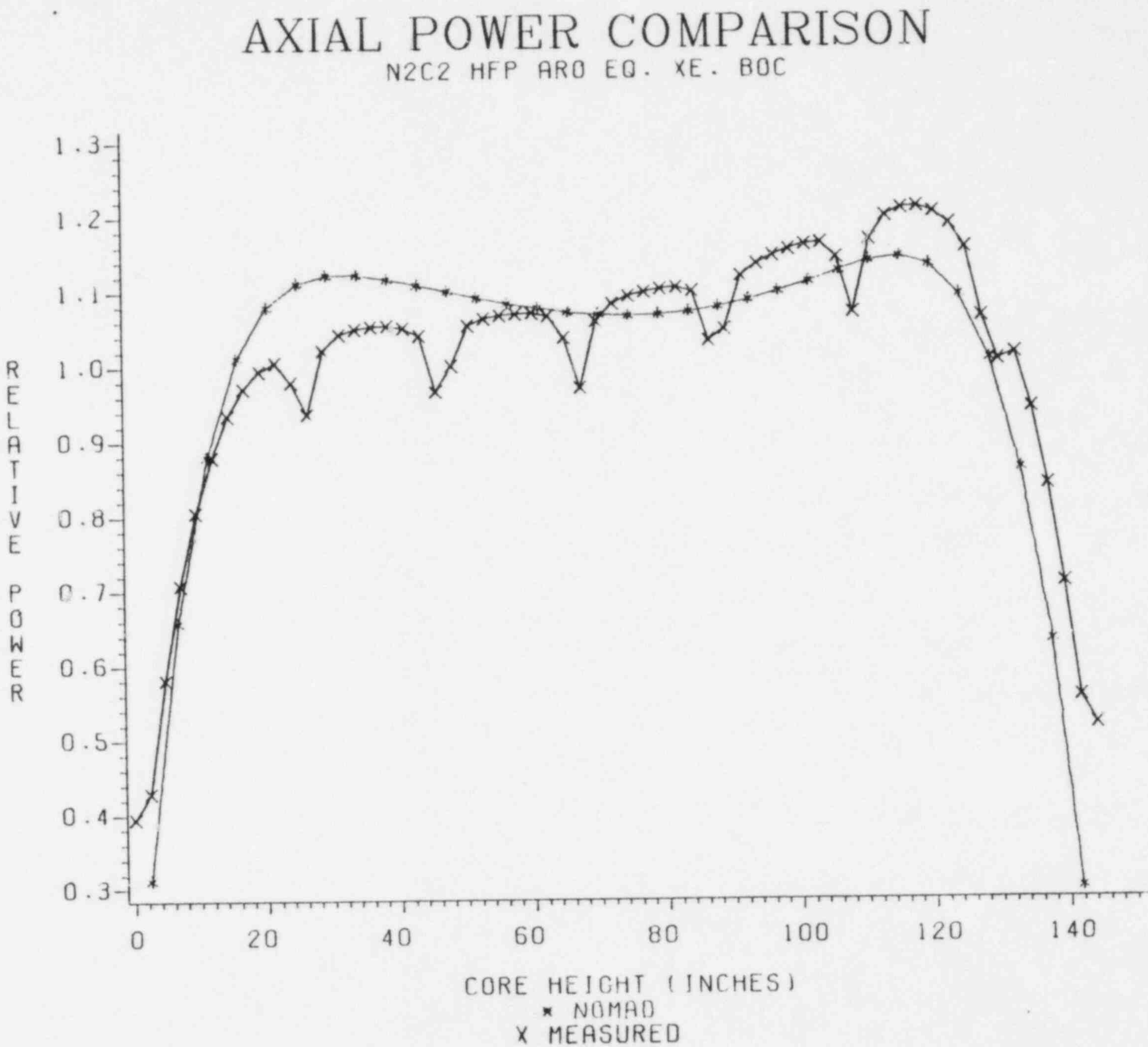

AXIAL POWER COMPARISON
N2C2 HFP ARO EQ. XE. BOC
RELATIVE POWER
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0
20
40
60
80
100
120
140
CORE HEIGHT (INCHES)
* NOMAD
X MEASURED

FIGURE 5-14

AXIAL POWER COMPARISON

S1C6 HZP BOC

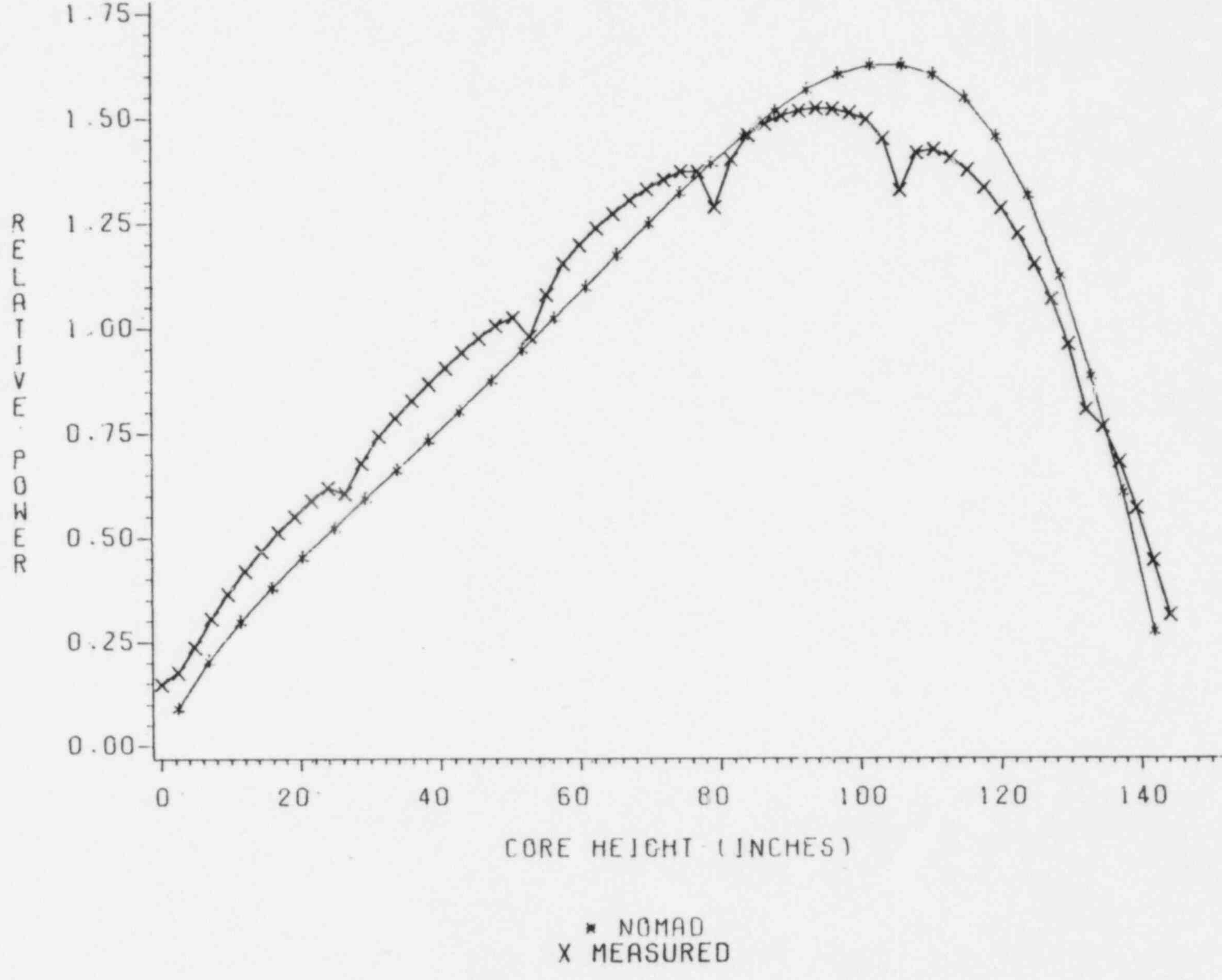


FIGURE 5-15

AXIAL POWER COMPARISON

S1C6 HFP ARO EQ. XE. BOC

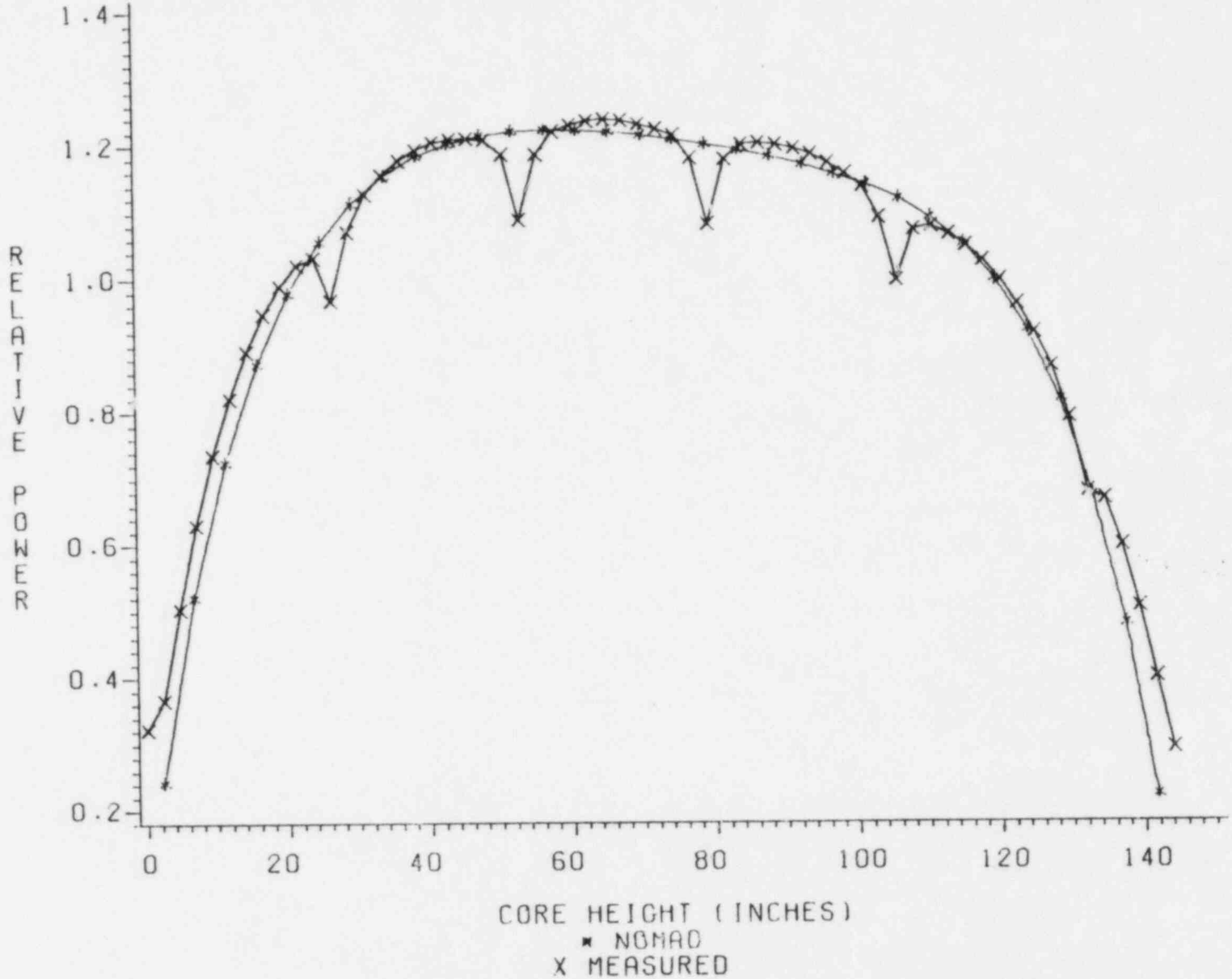


FIGURE 5-16

DIFFERENTIAL ROD WORTH COMPARISON

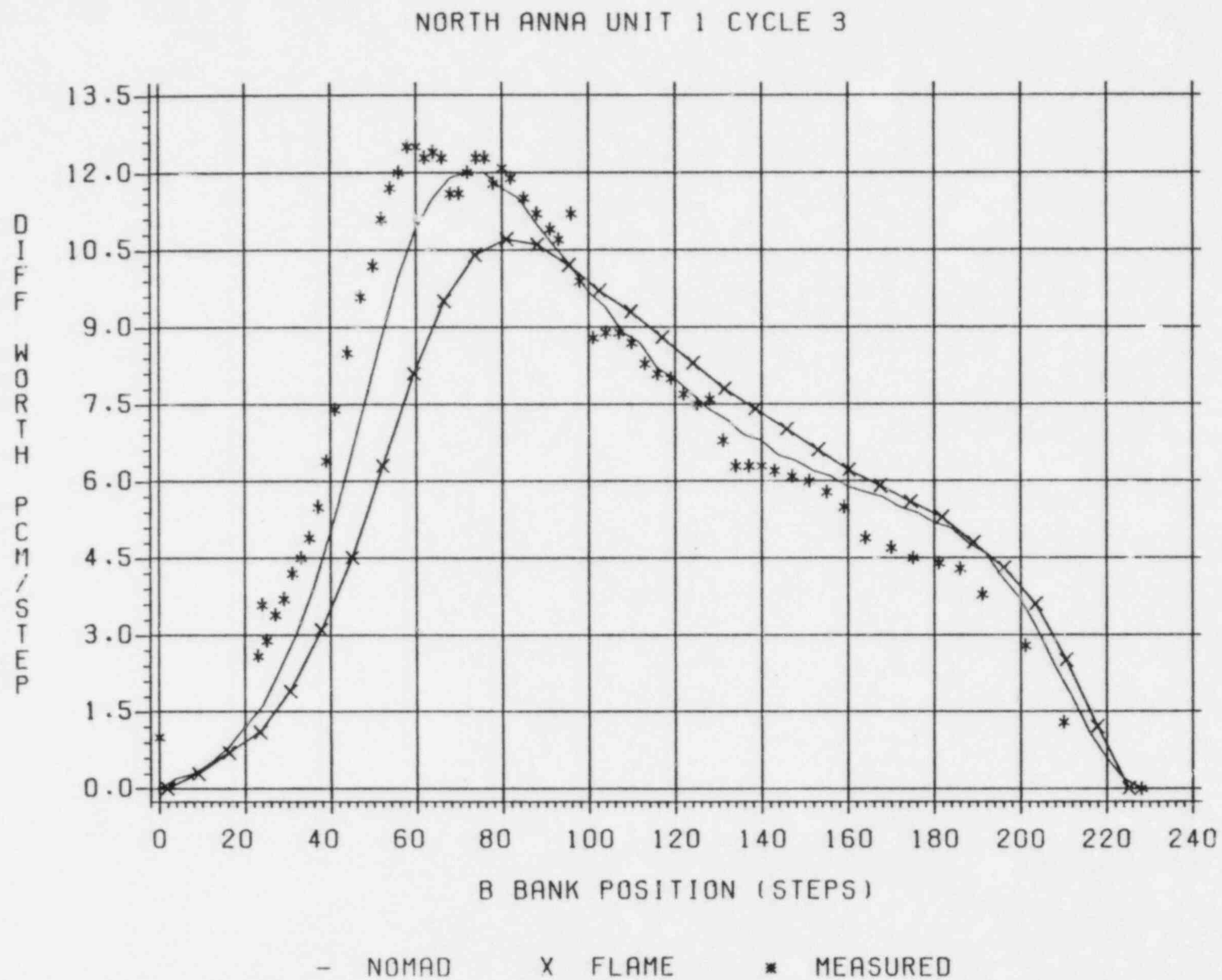
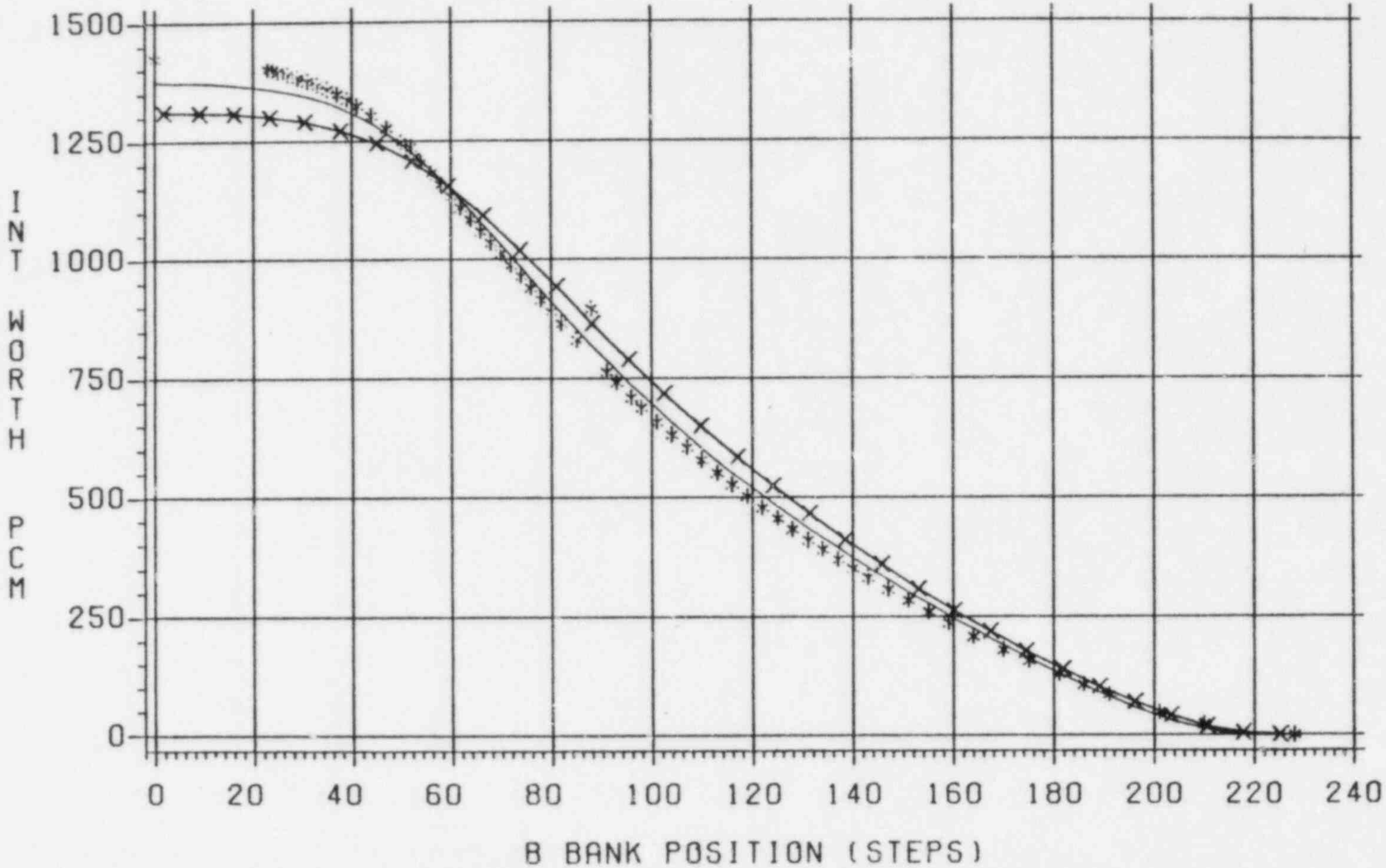


FIGURE 5-17

INTEGRAL ROD WORTH COMPARISON

NORTH ANNA UNIT 1 CYCLE 3



- NOMAD X FLAME * MEASURED

FIGURE 5-18

DIFFERENTIAL ROD WORTH COMPARISON

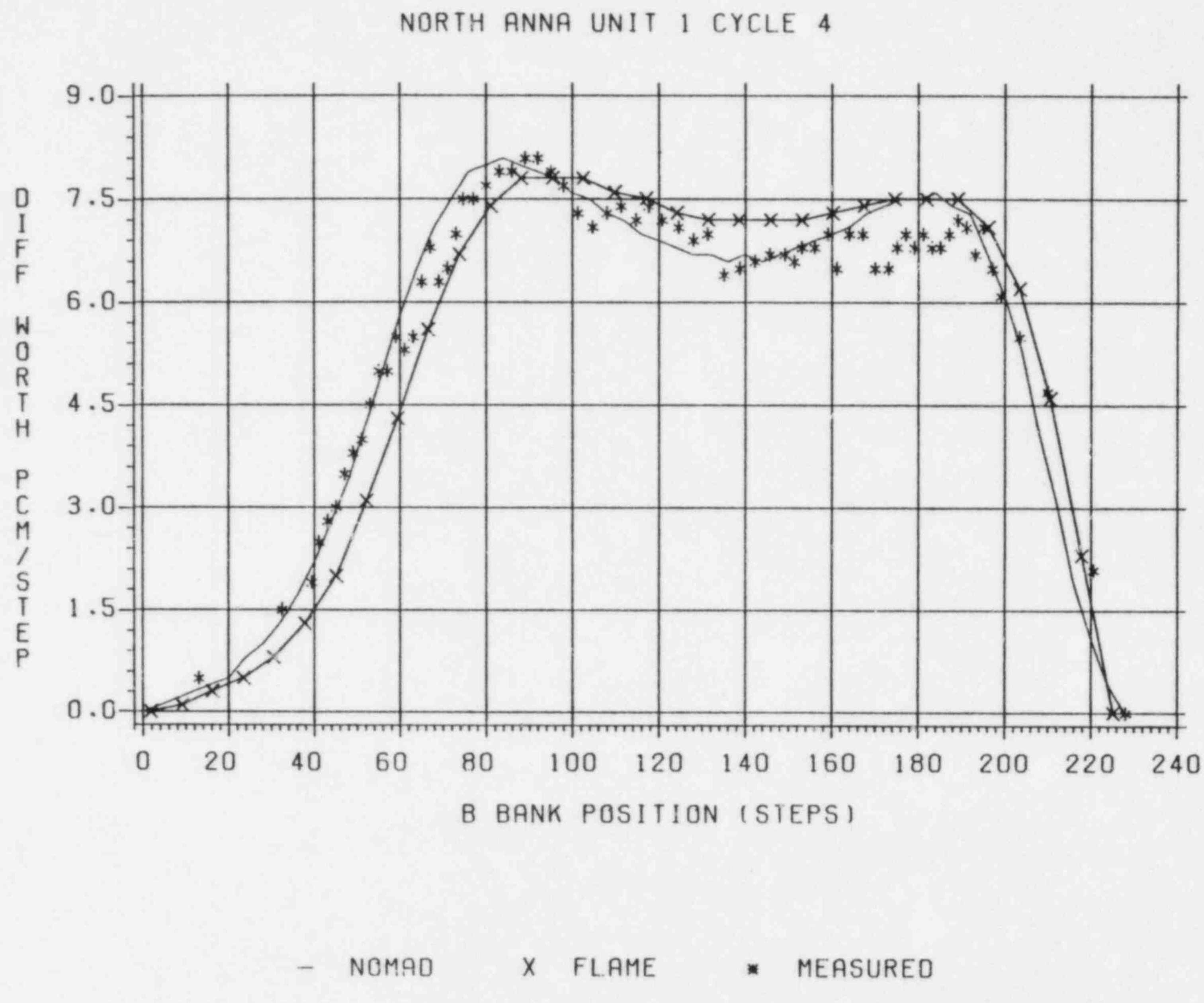


FIGURE 5-19

INTEGRAL ROD WORTH COMPARISON

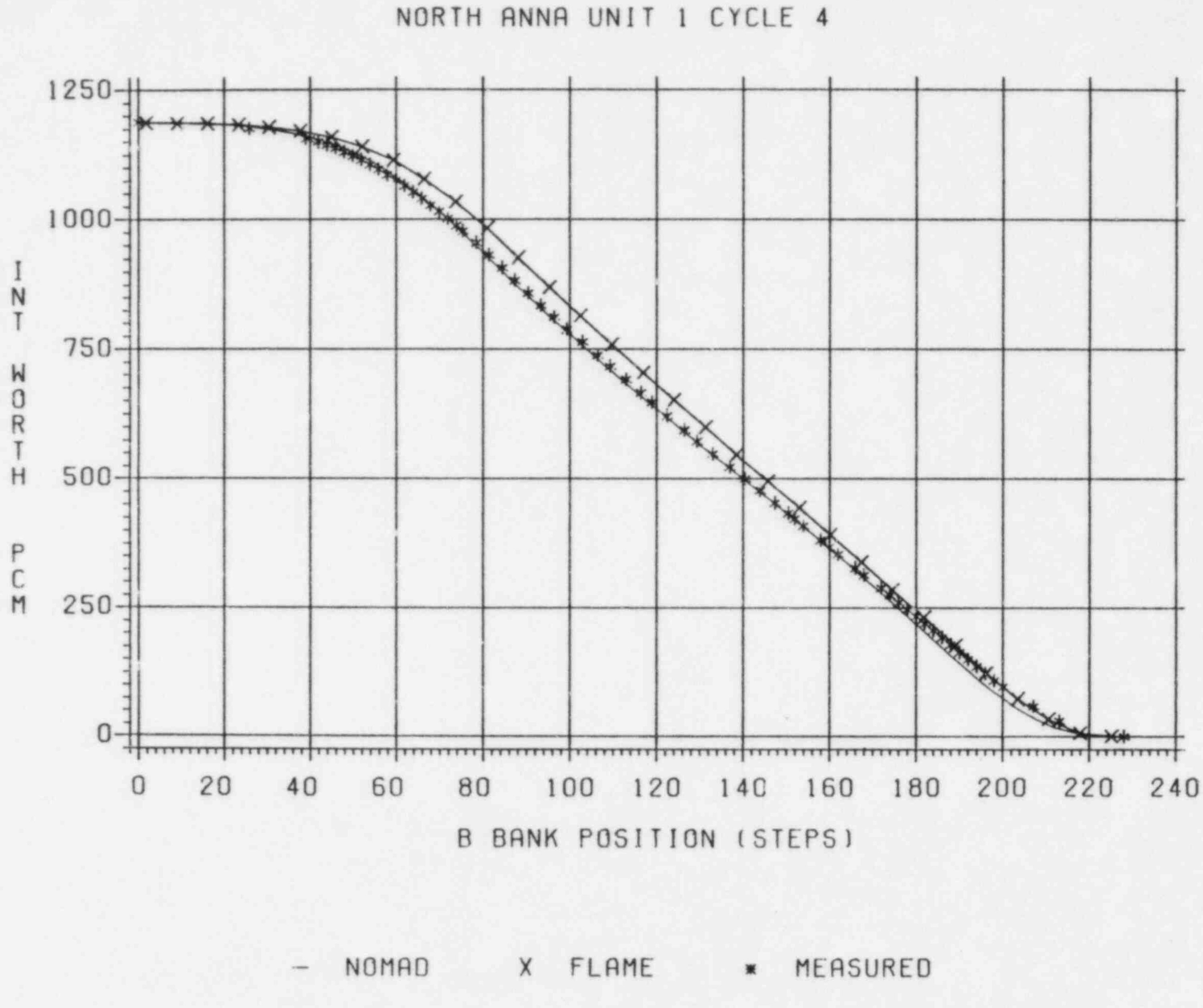


FIGURE 5-20

DIFFERENTIAL ROD WORTH COMPARISON

NORTH ANNA UNIT 2 CYCLE 2

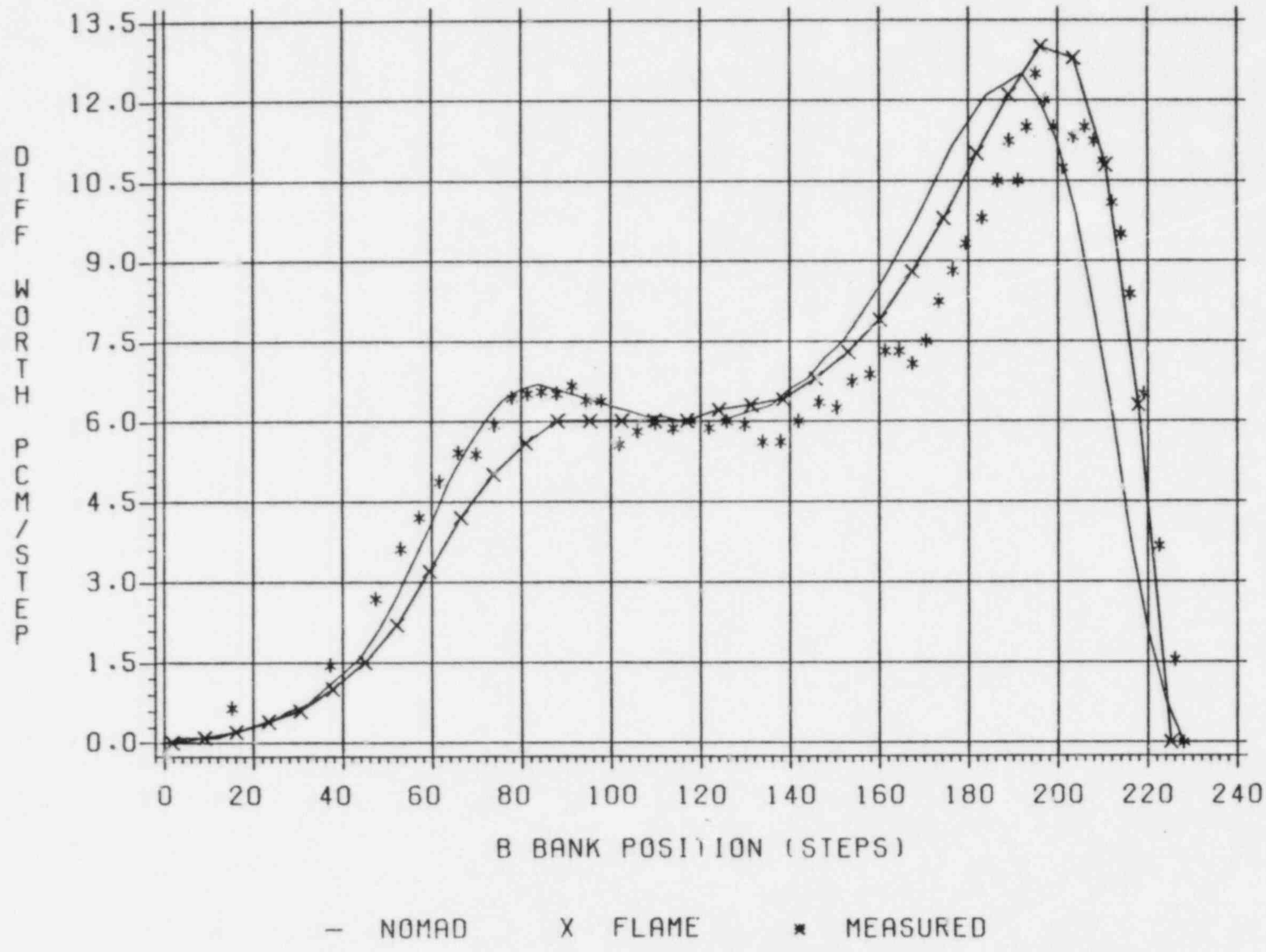


FIGURE 5-21

INTEGRAL ROD WORTH COMPARISON

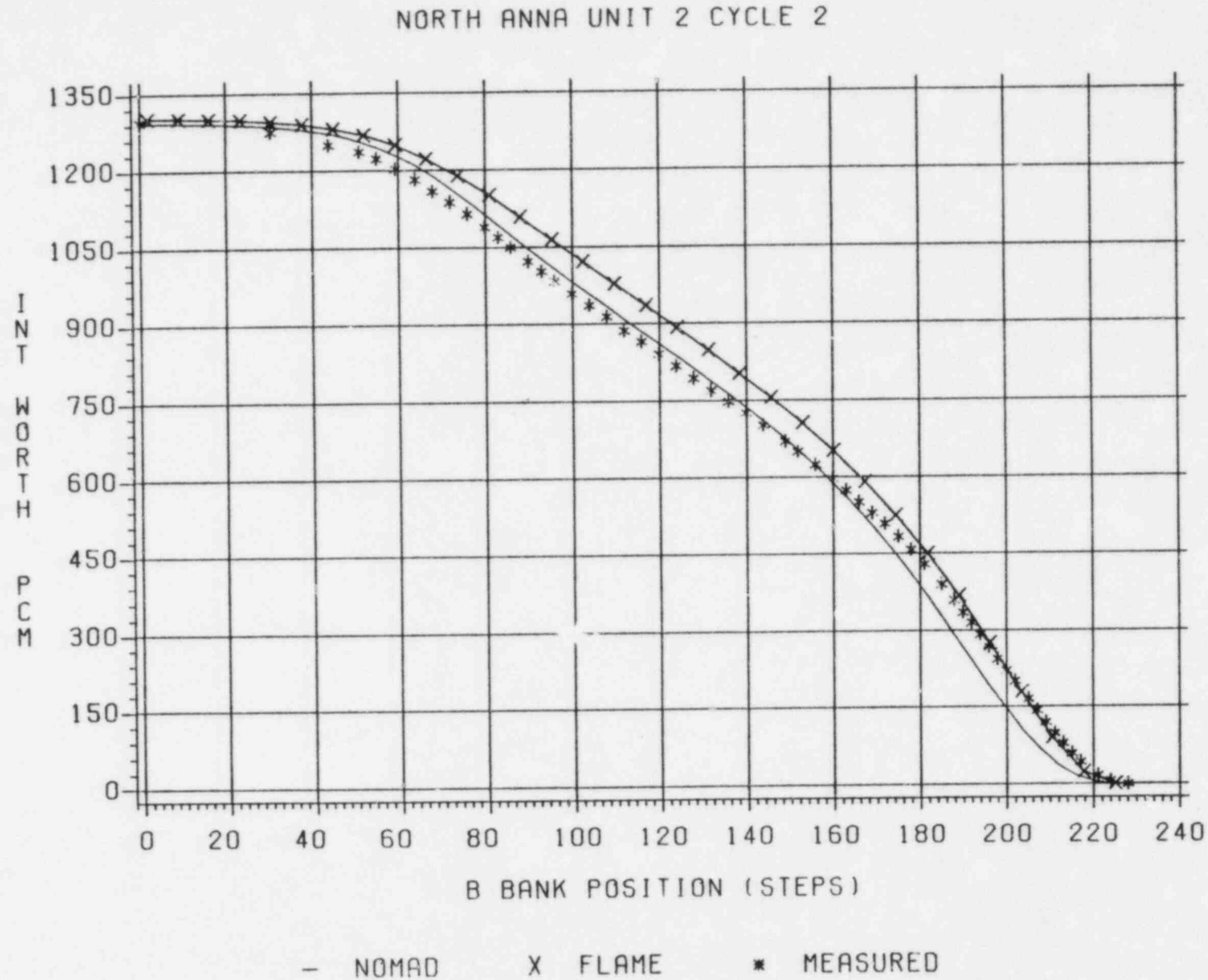


FIGURE 5-22

DIFFERENTIAL ROD WORTH COMPARISON

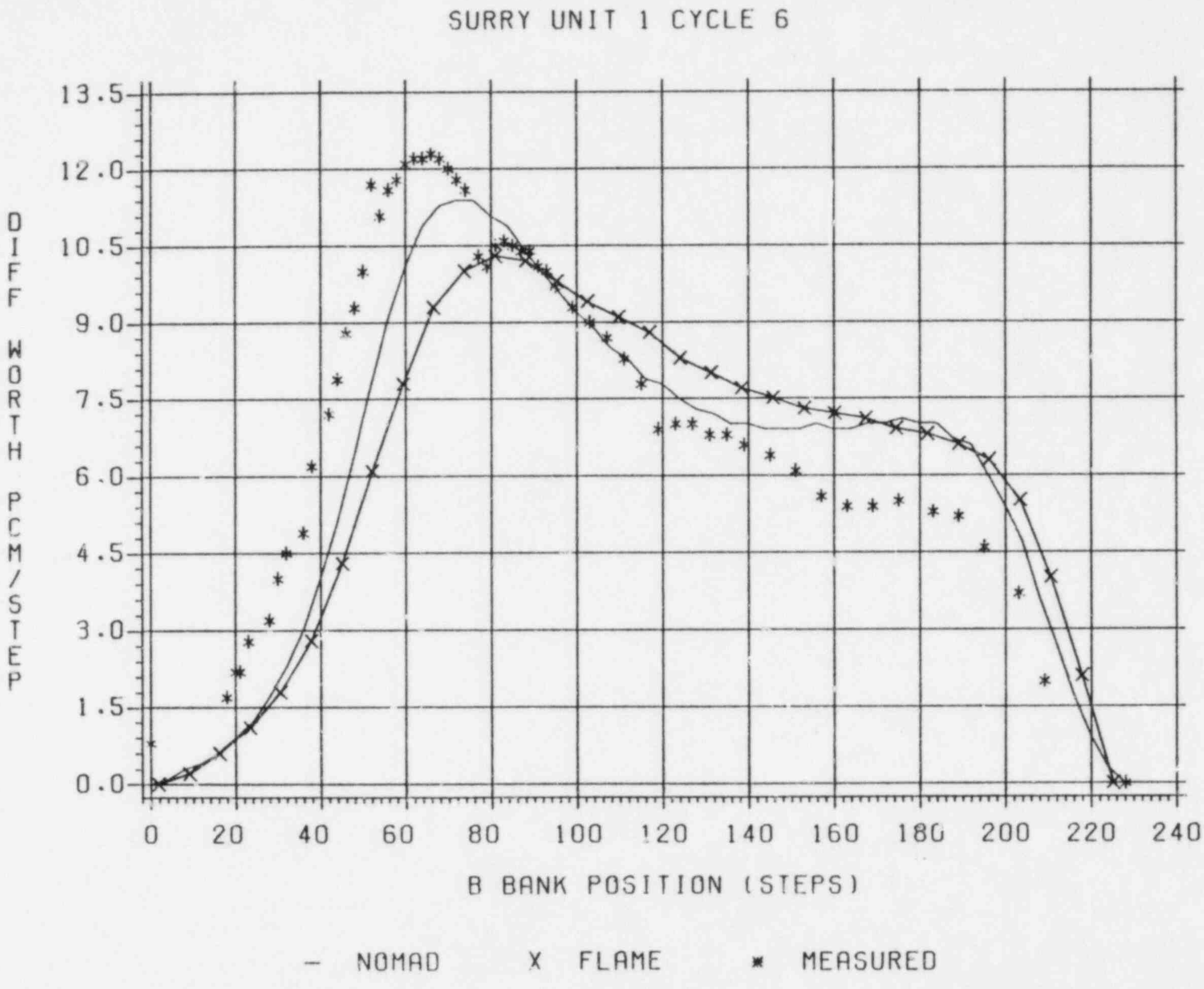


FIGURE 5-23

INTEGRAL ROD WORTH COMPARISON

SURRY UNIT 1 CYCLE 6

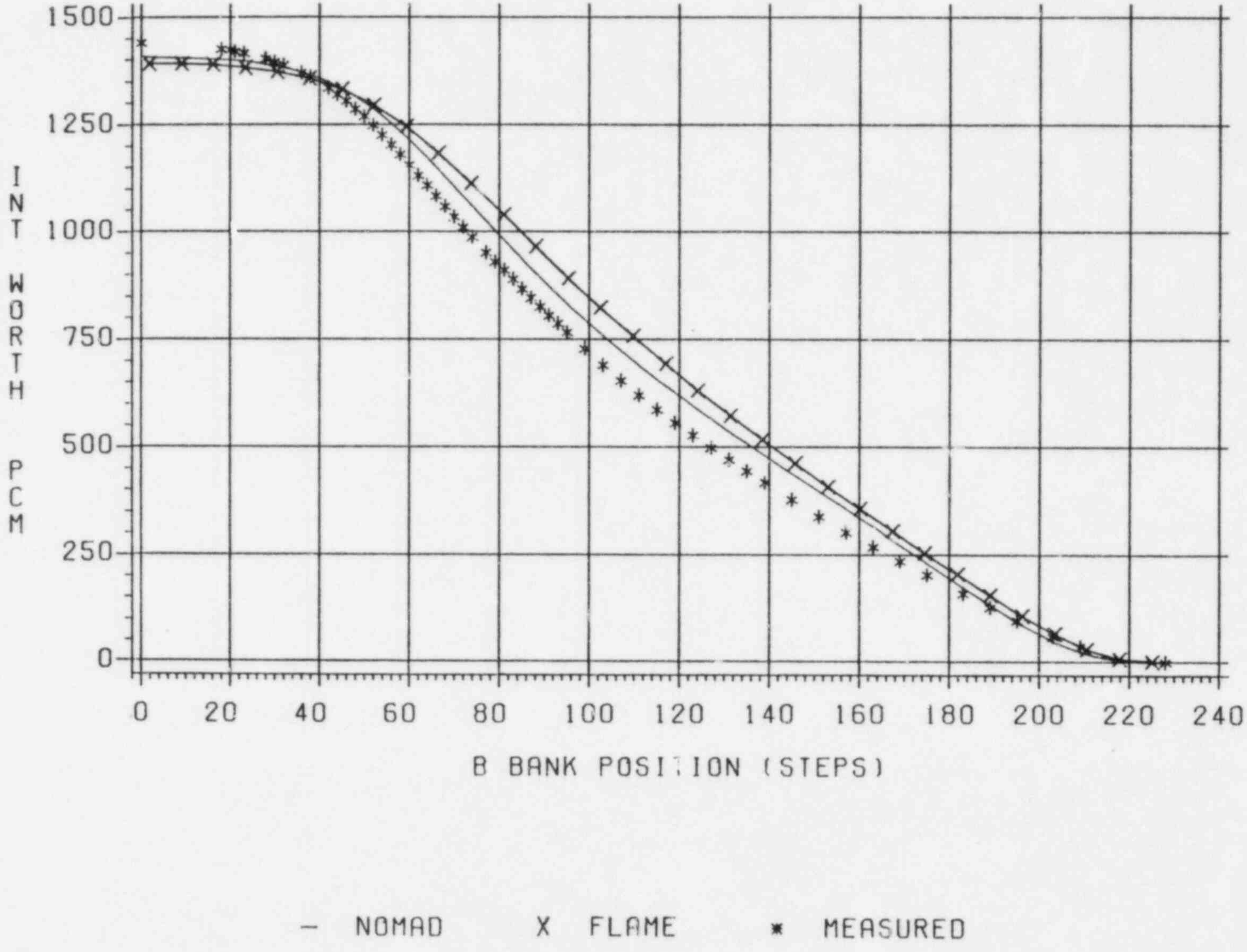


FIGURE 5-24

FIGURE 5-25

DIFFERENTIAL WORTH OF CONTROL BANKS A THROUGH D IN OVERLAP MODE NORTH ANNA UNIT 1, CYCLE 3

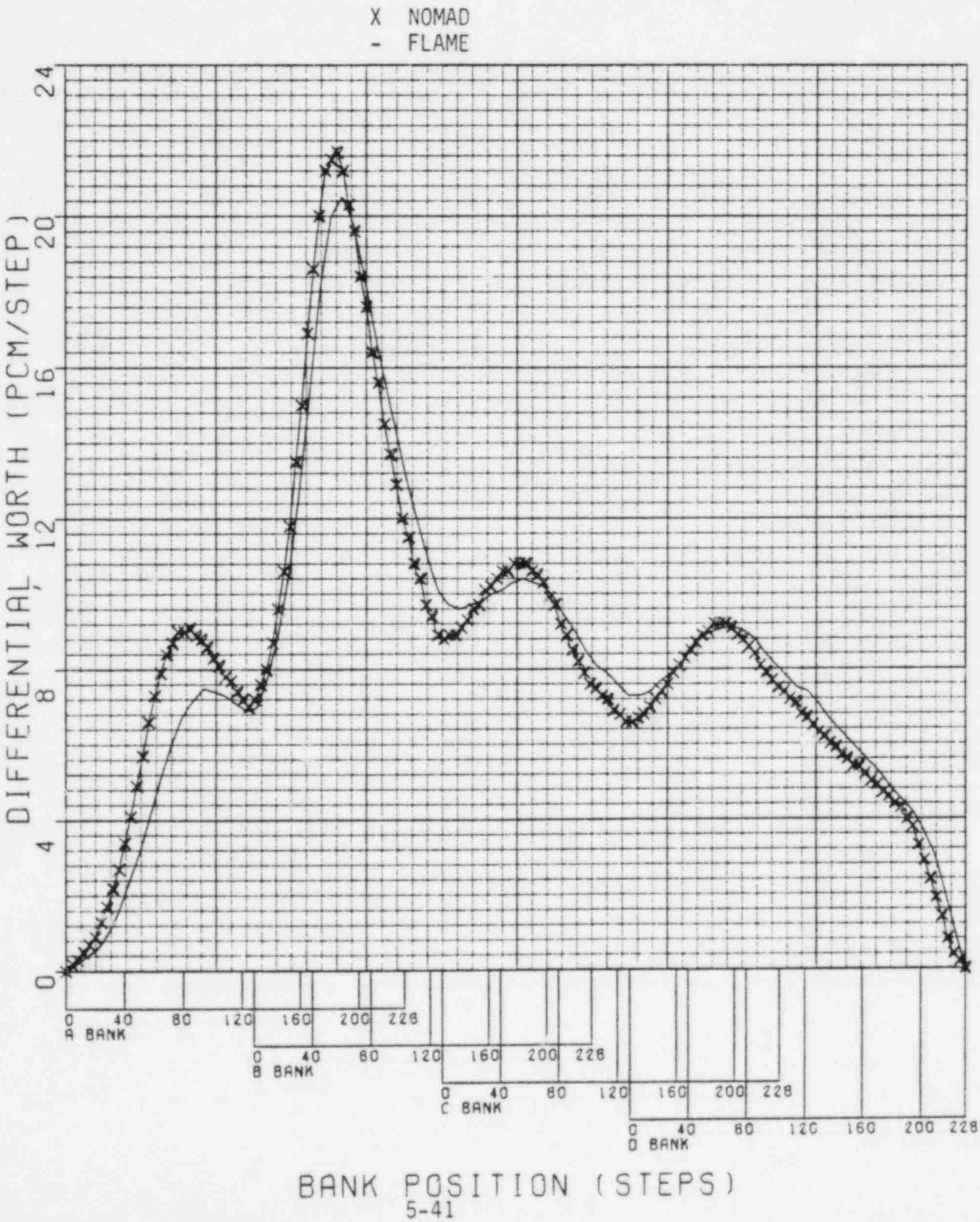


FIGURE 5-26

INTEGRAL WORTH OF CONTROL BANKS
A THROUGH D IN OVERLAP MODE
NORTH ANNA UNIT 1, CYCLE 3

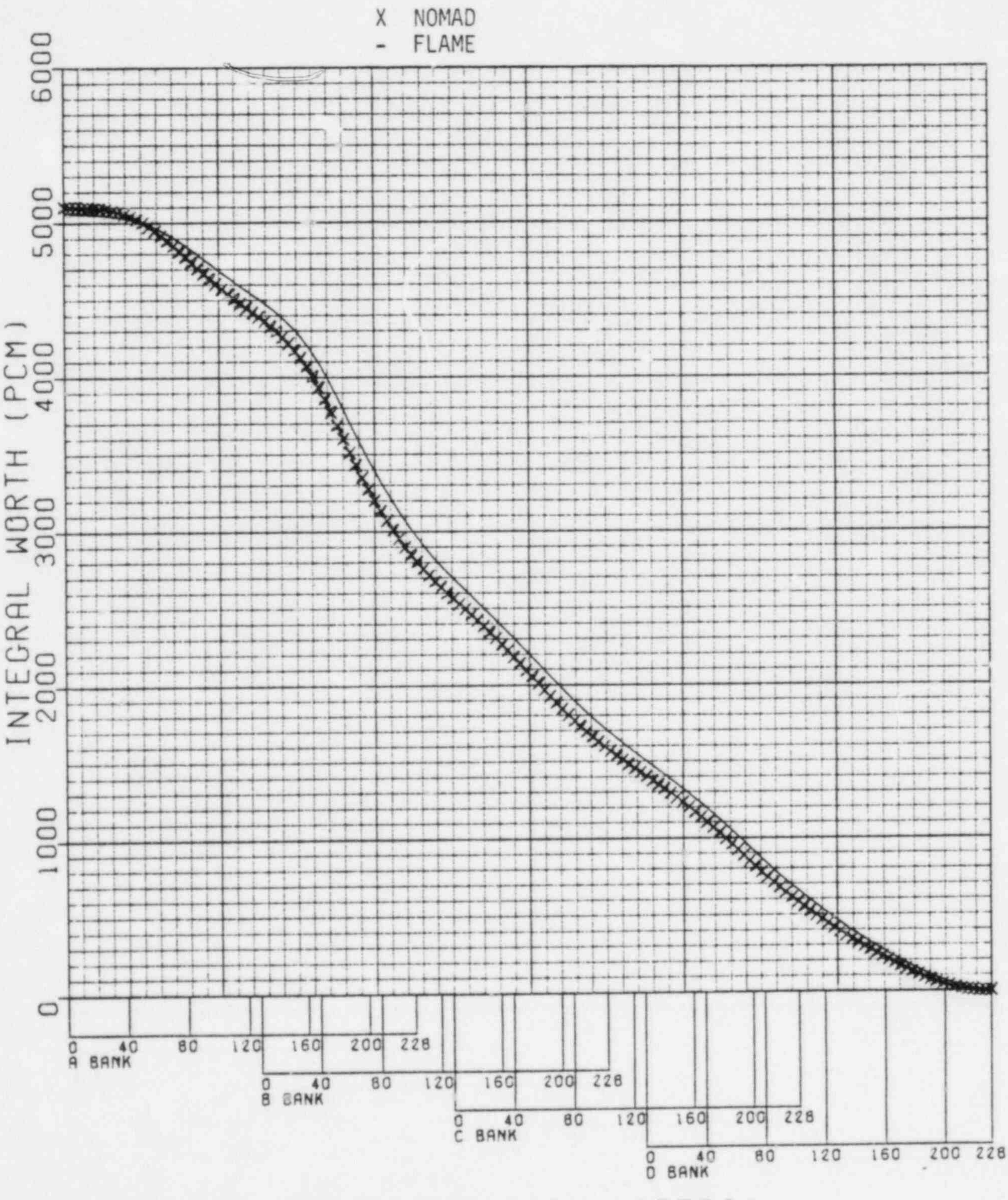


FIGURE 5-27
DIFFERENTIAL WORTH OF CONTROL BANKS
A THROUGH D IN OVERLAP MODE
NORTH ANNA UNIT 1, CYCLE 4

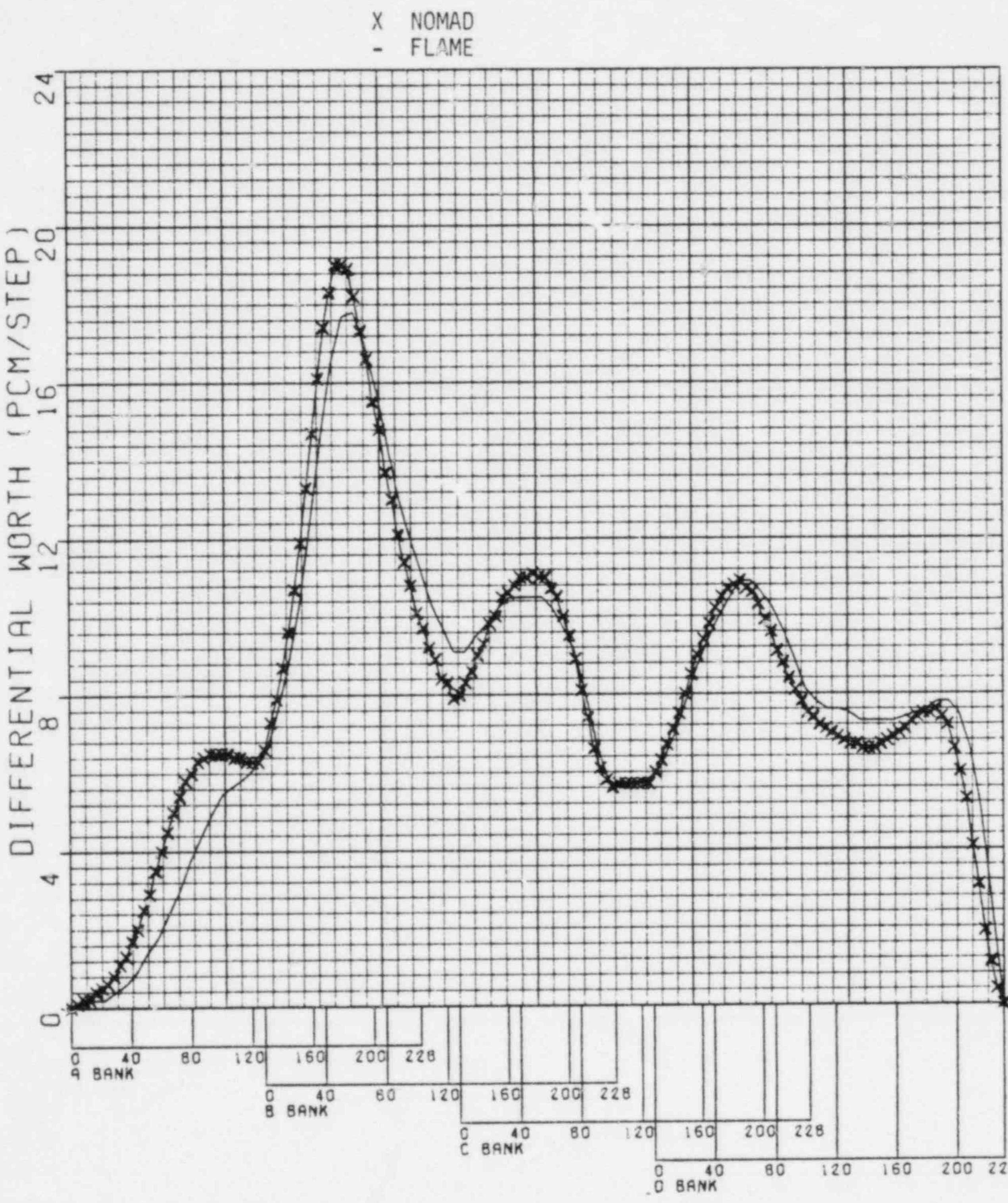


FIGURE 5-28

INTEGRAL WORTH OF CONTROL BANKS
A THROUGH D IN OVERLAP MODE
NORTH ANNA UNIT 1, CYCLE 4

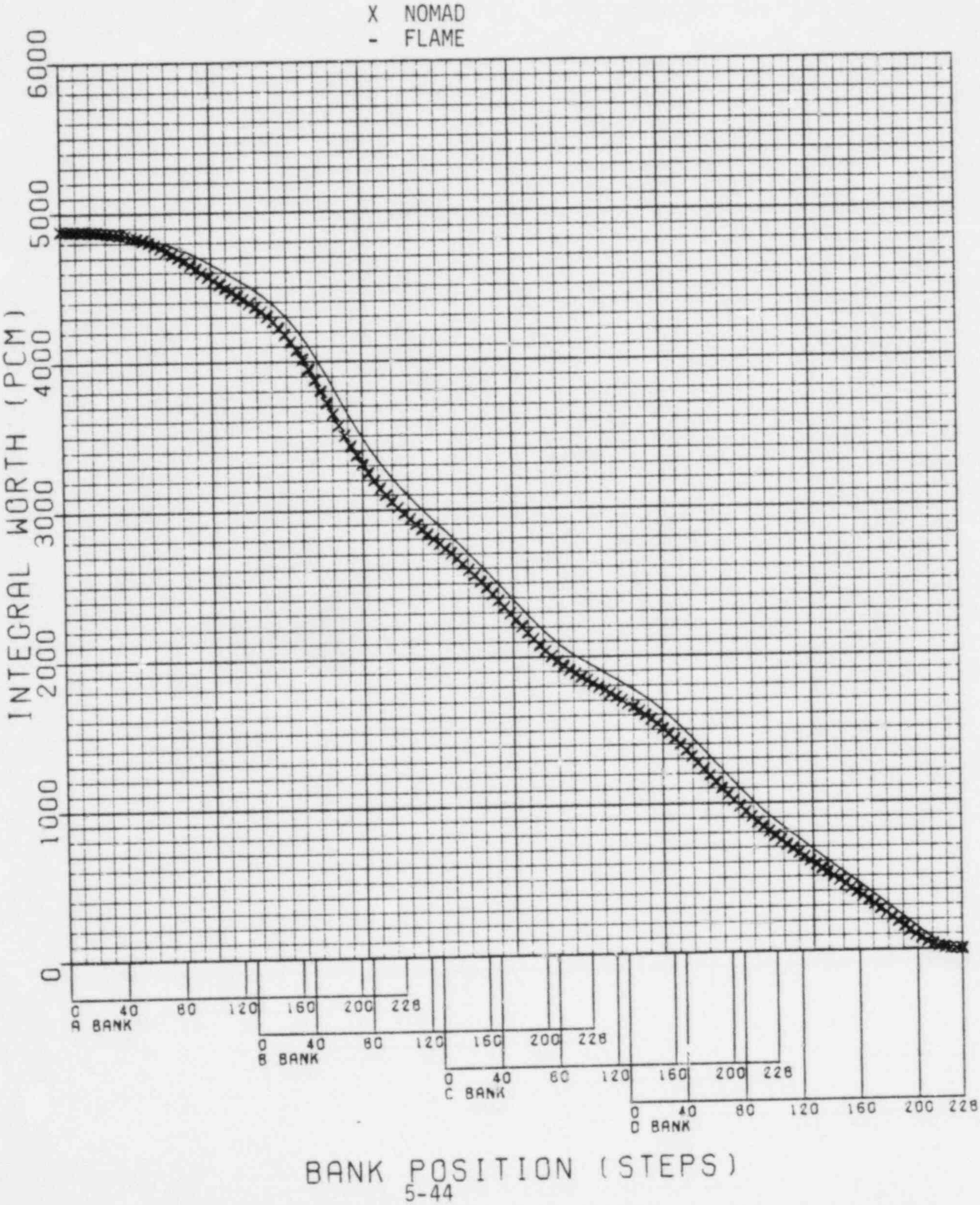


FIGURE 5-29

DIFFERENTIAL WORTH OF CONTROL BANKS A THROUGH D IN OVERLAP MODE NORTH ANNA UNIT 2, CYCLE 2

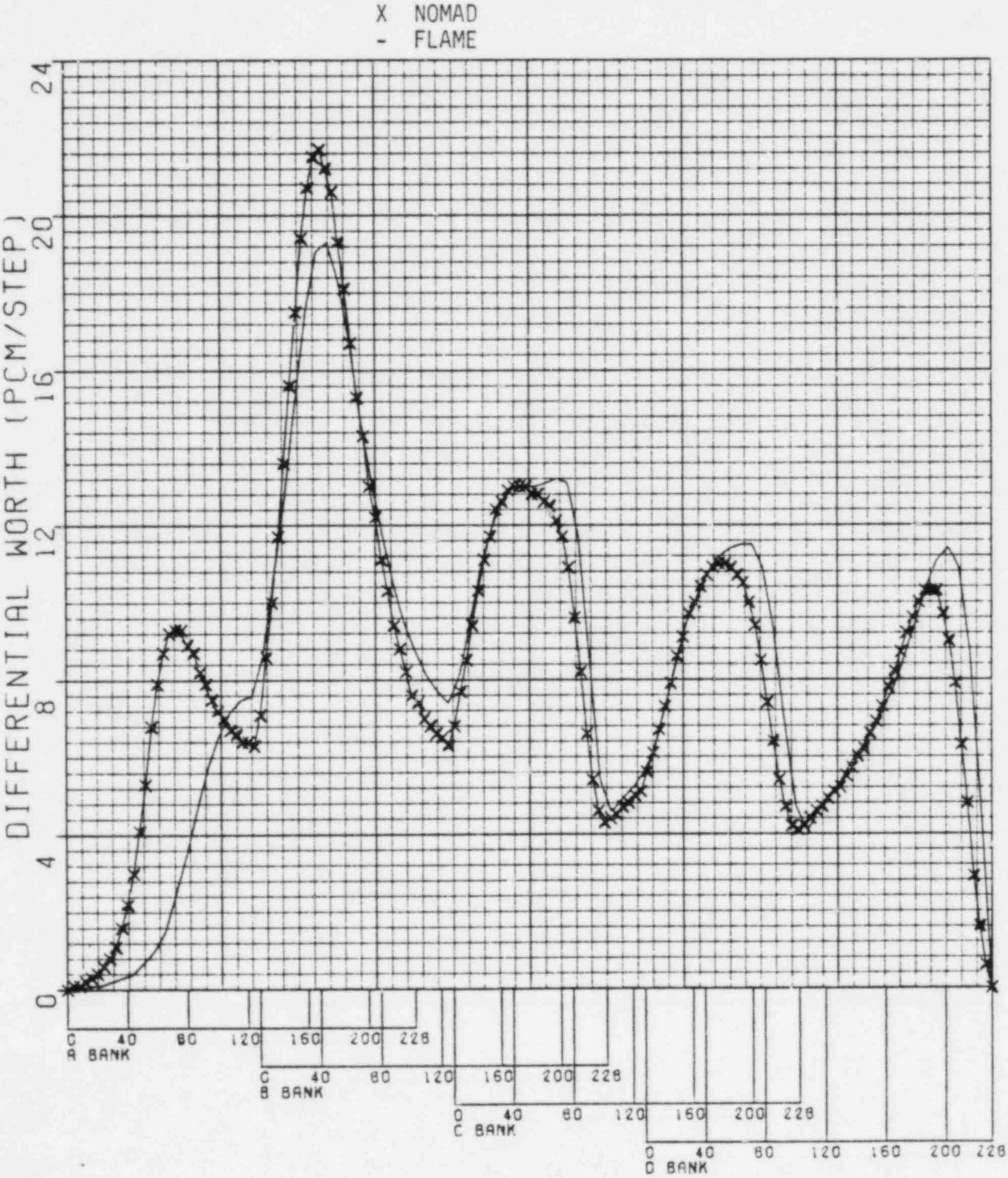


FIGURE 5-30

INTEGRAL WORTH OF CONTROL BANKS
A THROUGH D IN OVERLAP MODE
NORTH ANNA UNIT 2, CYCLE 2

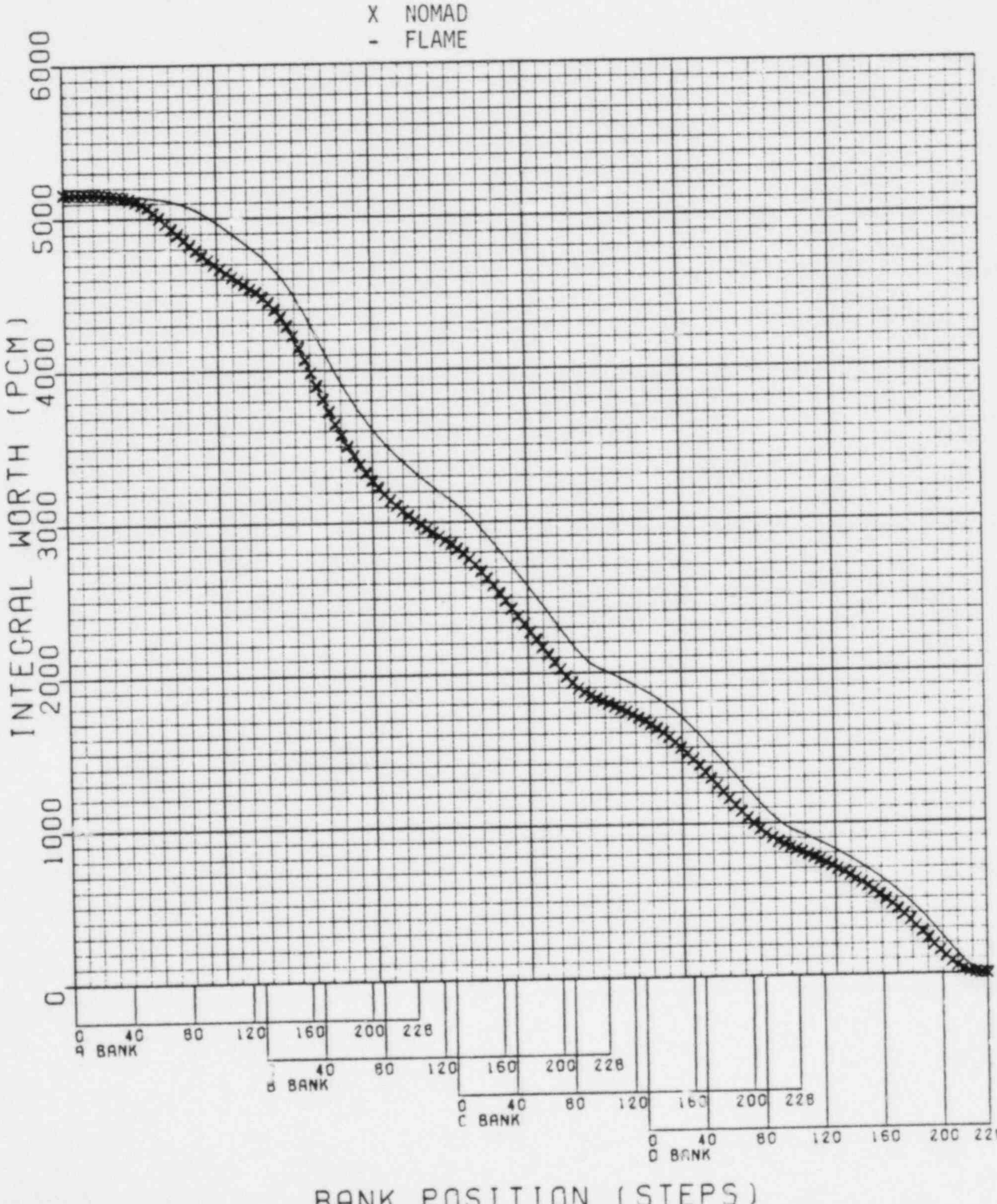


FIGURE 5-31

DIFFERENTIAL WORTH OF CONTROL BANKS
A THROUGH D IN OVERLAP MODE
SURRY UNIT 1, CYCLE 7

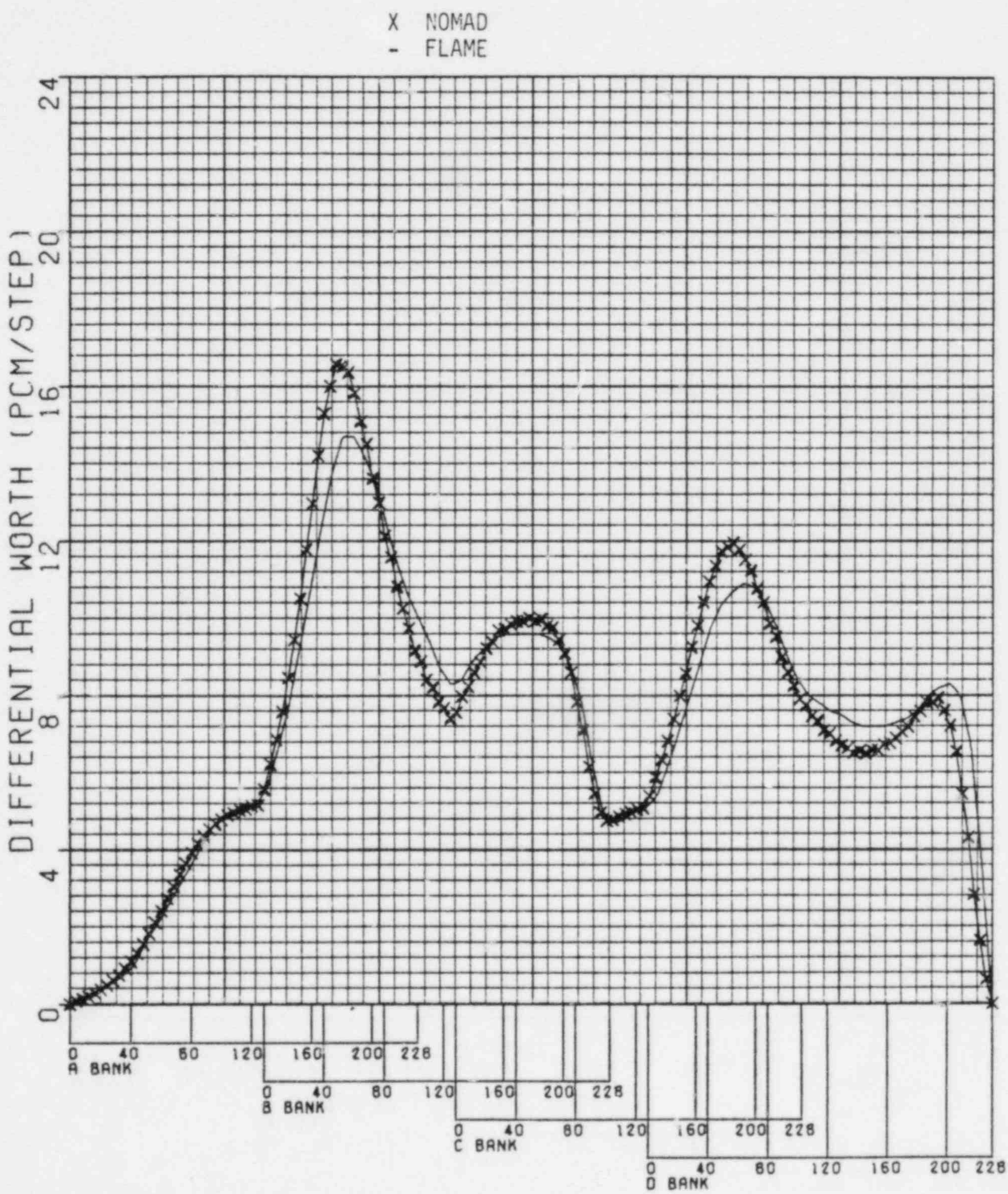


FIGURE 5-32

INTEGRAL WORTH OF CONTROL BANKS
A THROUGH D IN OVERLAP MODE
SURRY UNIT 1, CYCLE 7

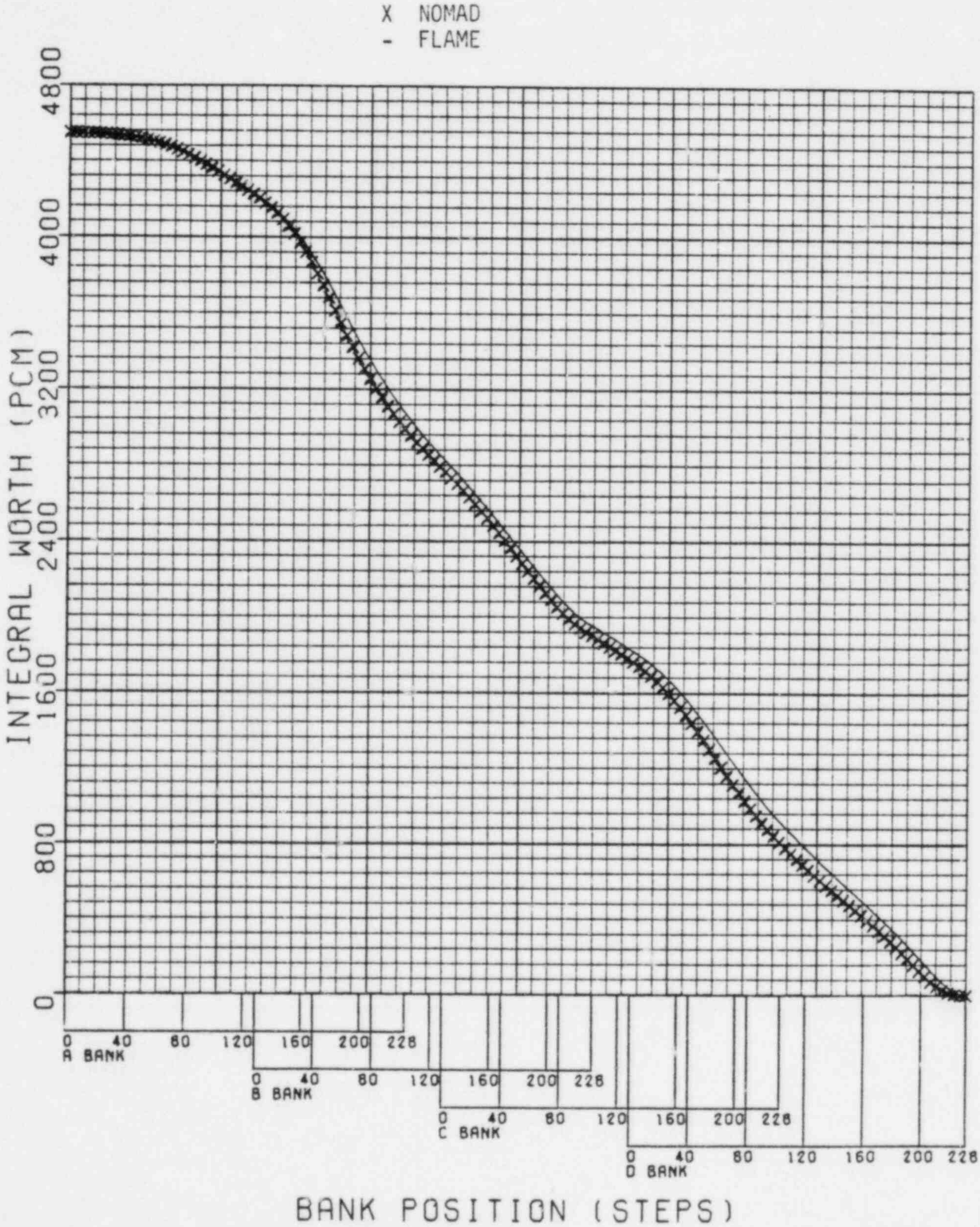
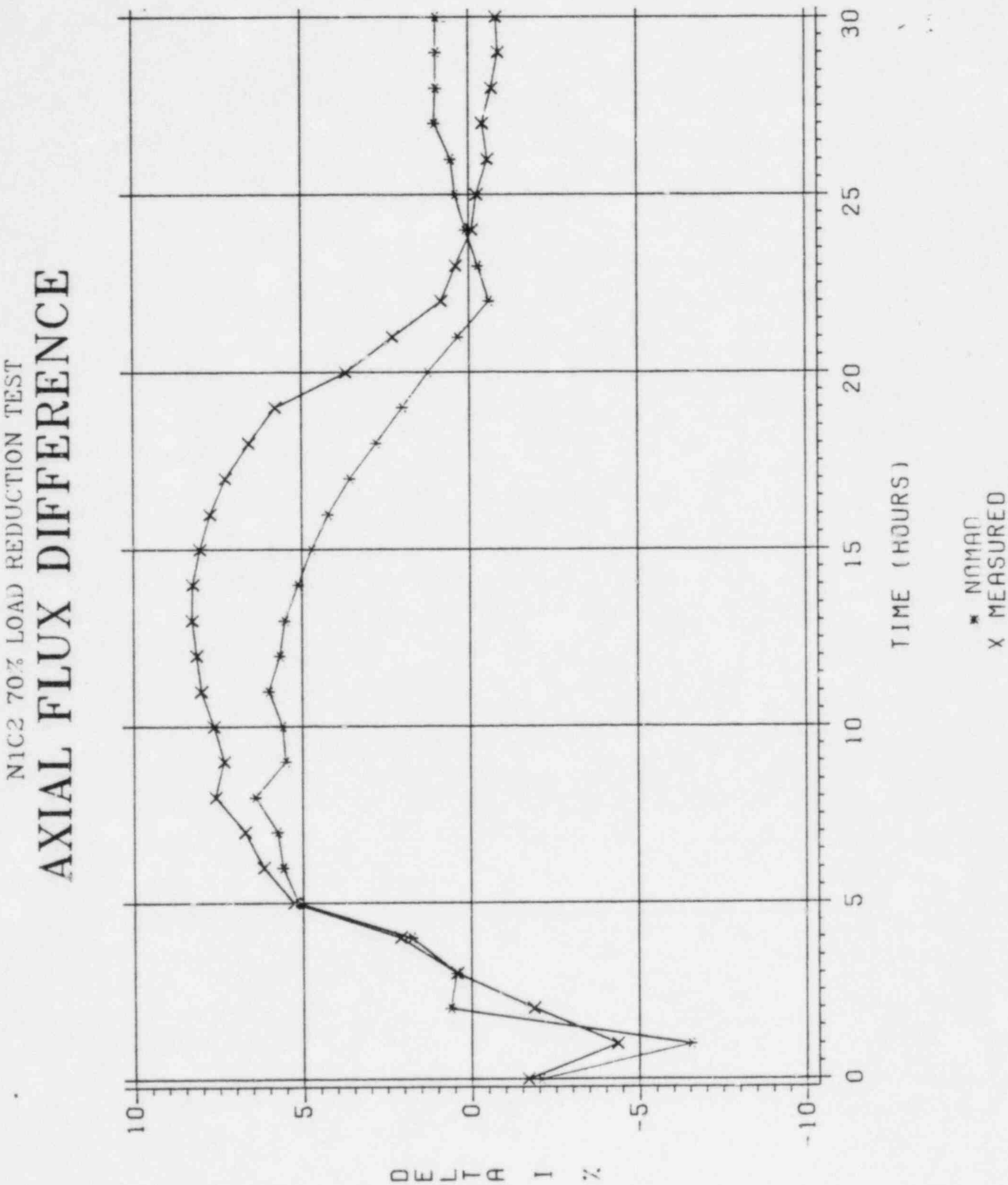


FIGURE 5-33


N1C2 70% LOAD REDUCTION TEST
AXIAL FLUX DIFFERENCE
DELTA I %
10
5
0
-5
-10
0
5
10
15
20
25
30
TIME (HOURS)
* NOMAD
X MEASURED

N1C2 70% LOAD REDUCTION TEST

CRITICAL BORON CONCENTRATION

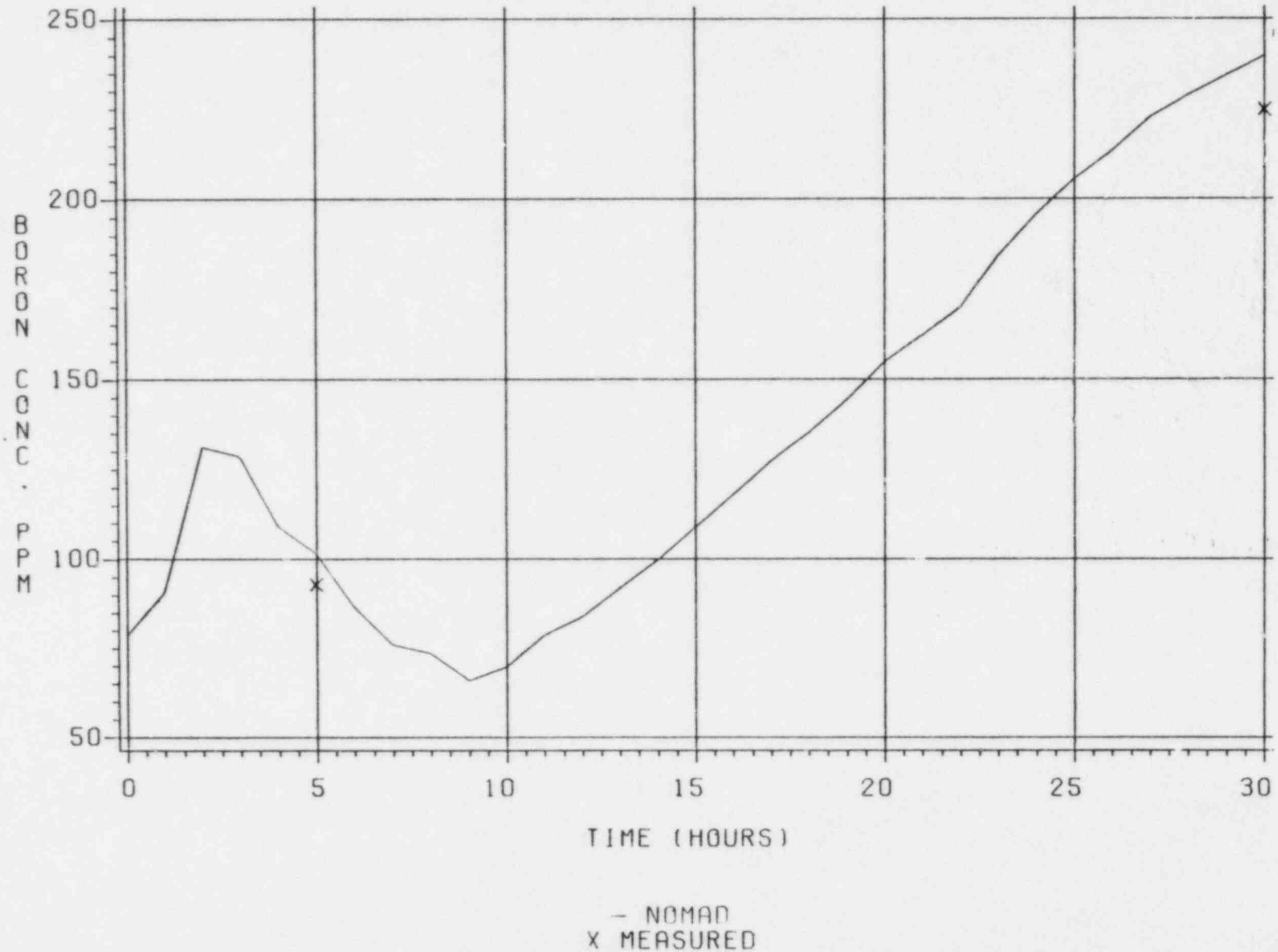
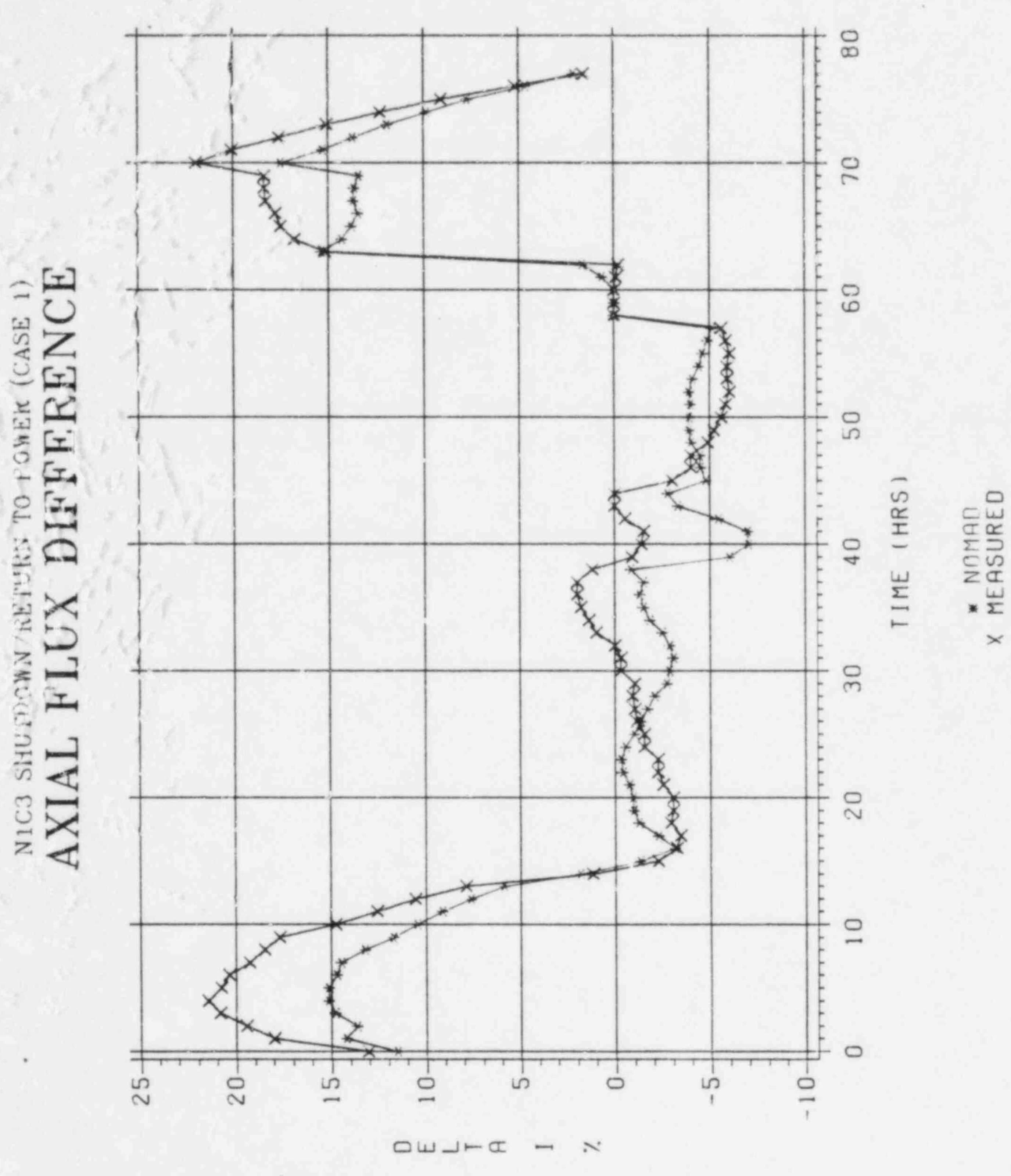


FIGURE 5-34

FIGURE 5-35

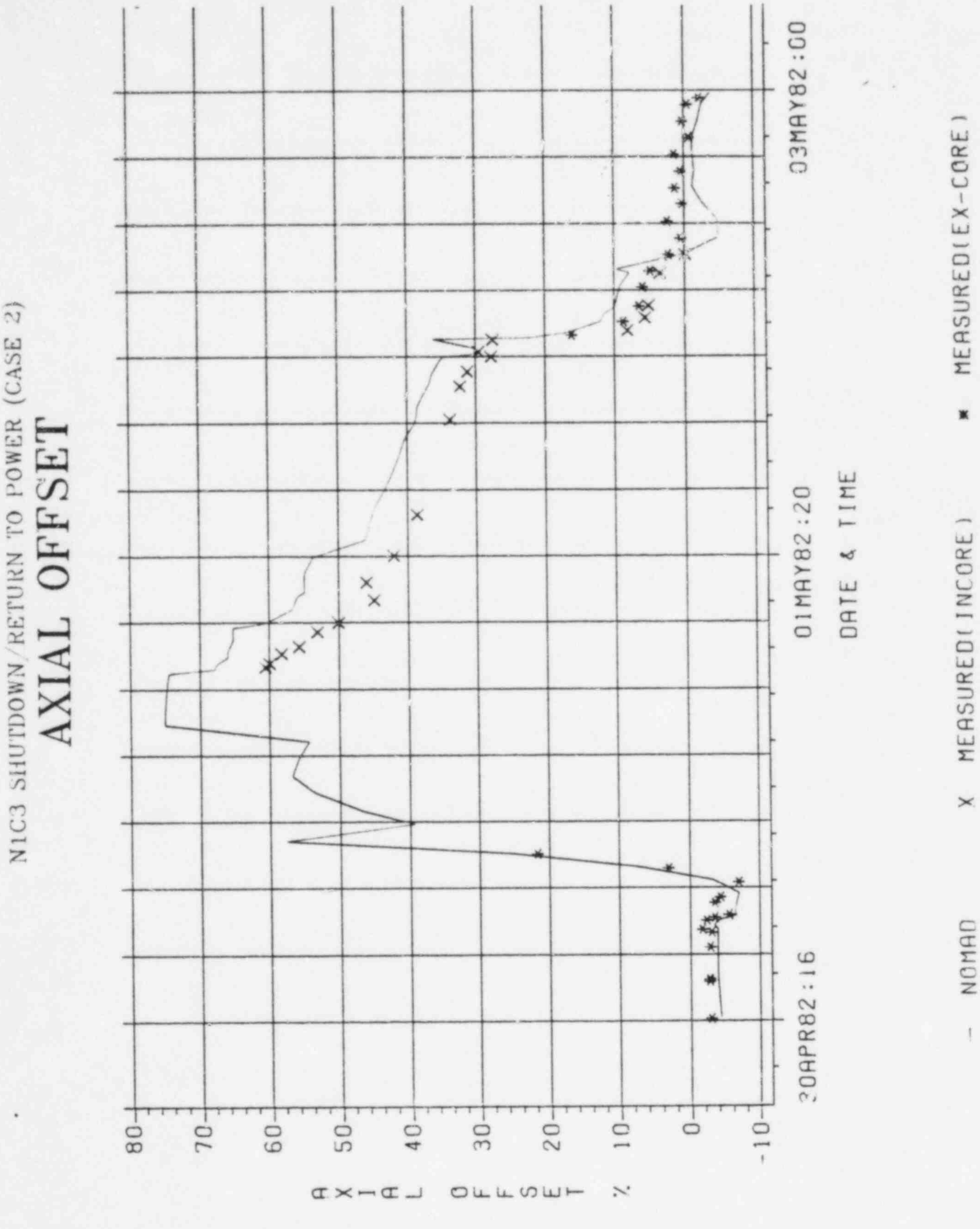

N1C3 SHUTDOWN/RETURN TO POWER (CASE 1)
AXIAL FLUX DIFFERENCE
DELTA I %
TIME (HRS)
* NOMAD
X MEASURED

N1C3 SHUTDOWN/RETURN TO POWER (CASE 1)

CRITICAL BORON CONCENTRATION

FIGURE 5-36

FIGURE 5-37


N1C3 SHUTDOWN/RETURN TO POWER (CASE 2)
AXIAL OFFSET
AXIAL OFFSET %
80
70
60
50
40
30
20
10
0
-10
20APR82:16
01MAY82:20
03MAY82:00
DATE & TIME
— NOMAD
X MEASURED(INCORE)
* MEASURED(EX-CORE)

N1C3 SHUTDOWN/RETURN TO POWER (CASE 2)

CRITICAL BORON CONCENTRATION

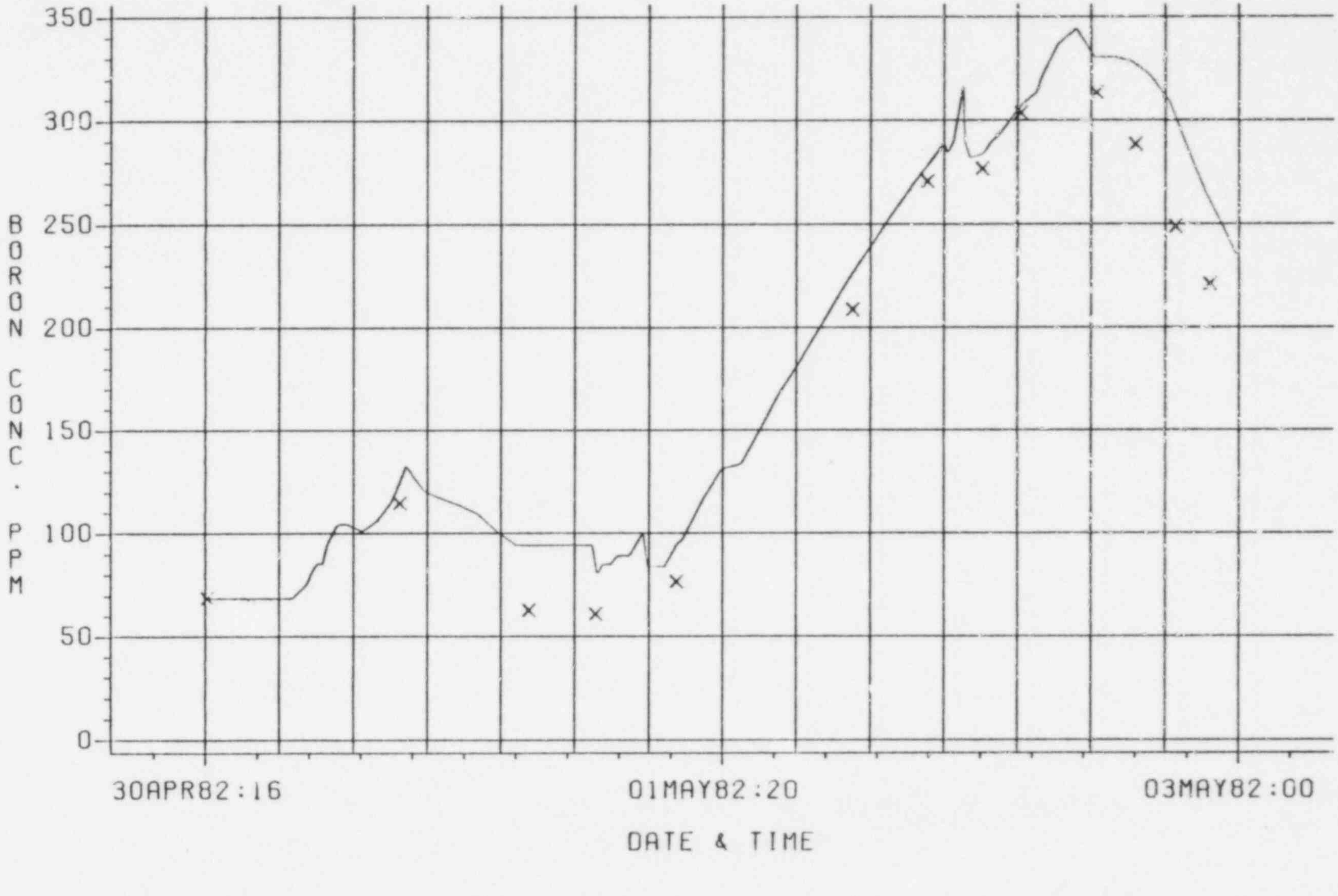
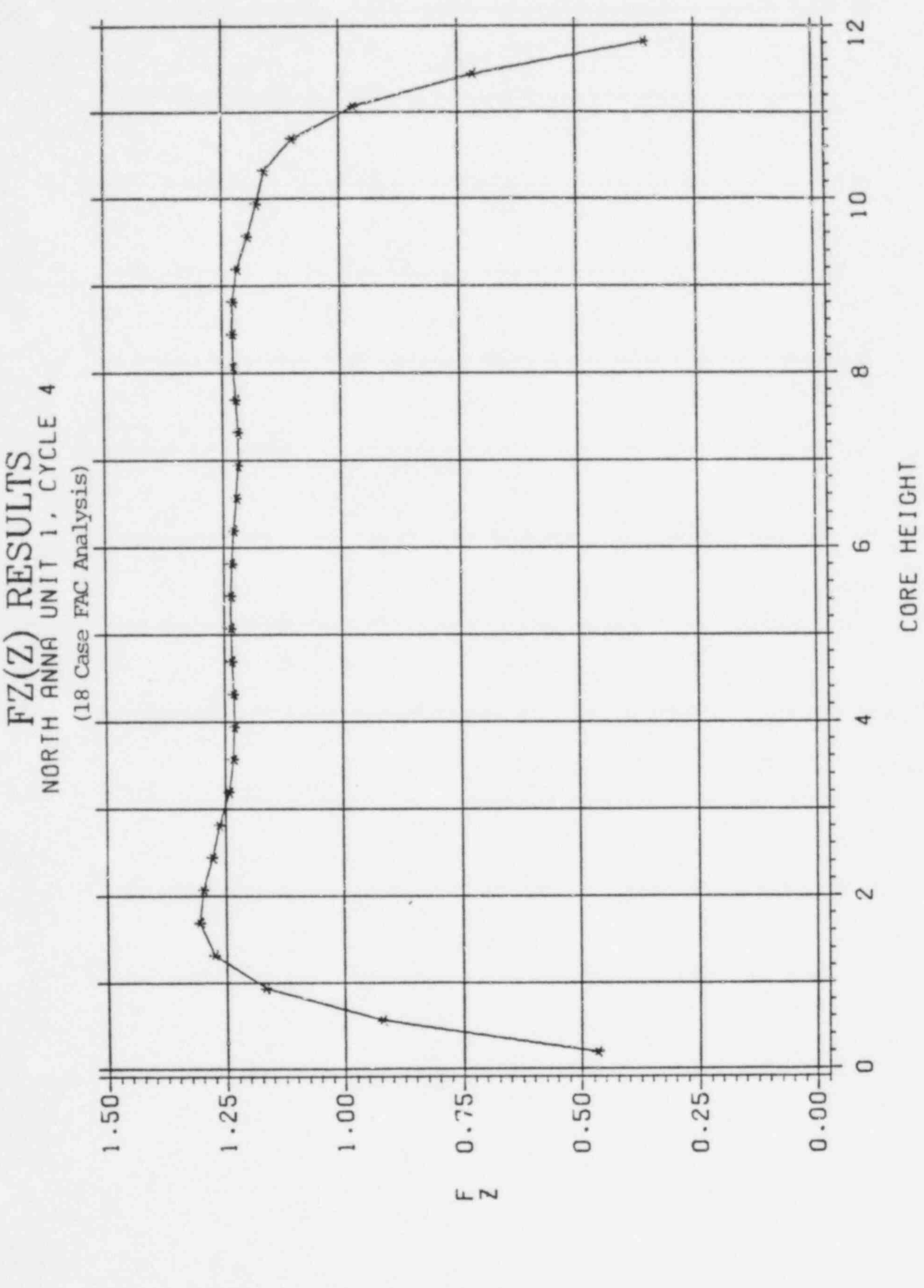


FIGURE 5-38

FIGURE 5-39



FZ(Z) RESULTS
NORTH ANNA UNIT 1, CYCLE 4
(18 Case FAC Analysis)
F
Z
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
2
4
6
8
10
12
CORE HEIGHT

FIGURE 5-40

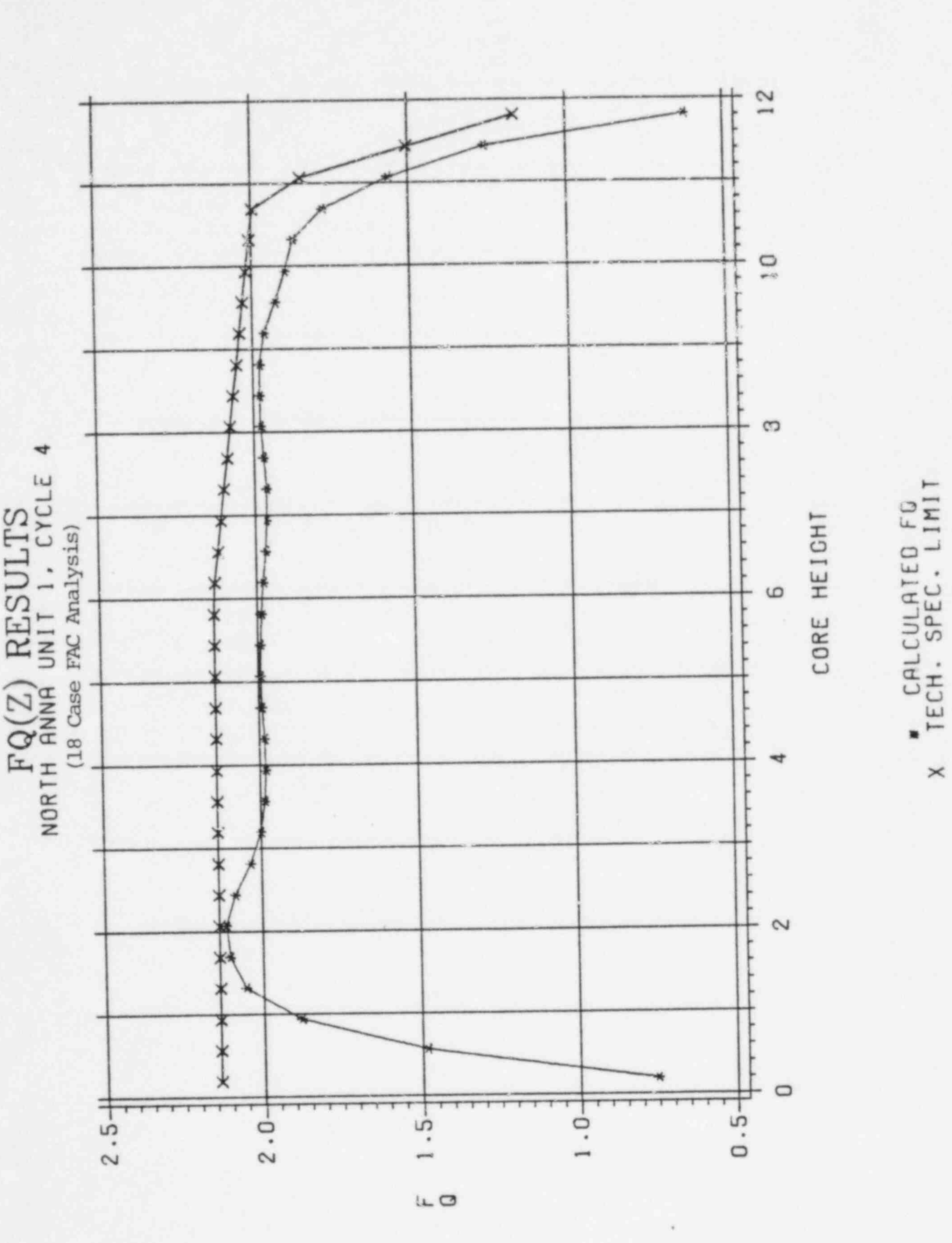

FQ(Z) RESULTS
NORTH ANNA UNIT 1, CYCLE 4
(18 Case FAC Analysis)
F
Q
2.5
2.0
1.5
1.0
0.5
0
2
4
6
8
10
12
CORE HEIGHT
■ CALCULATED FQ
X TECH. SPEC. LIMIT

FIGURE 5-41

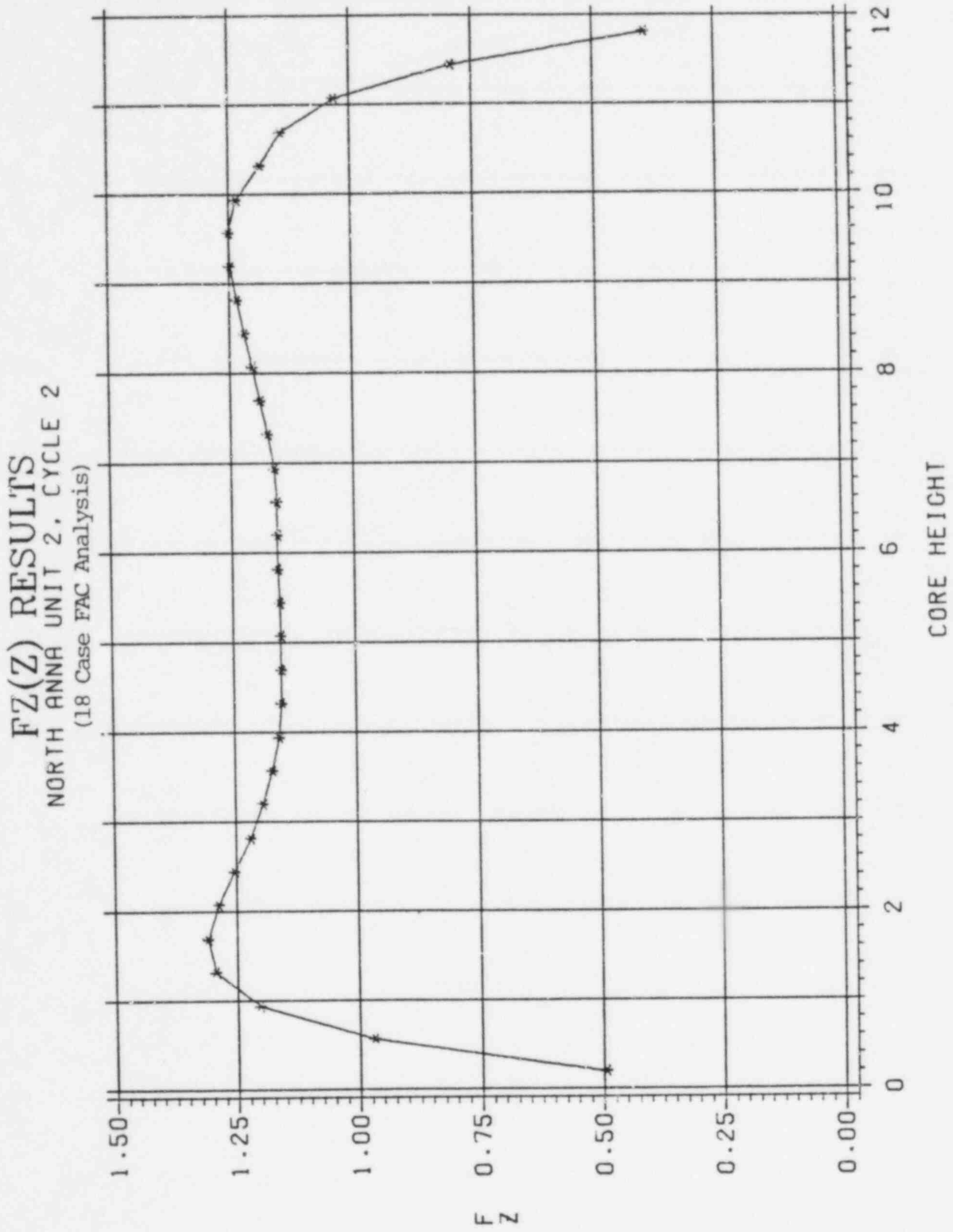
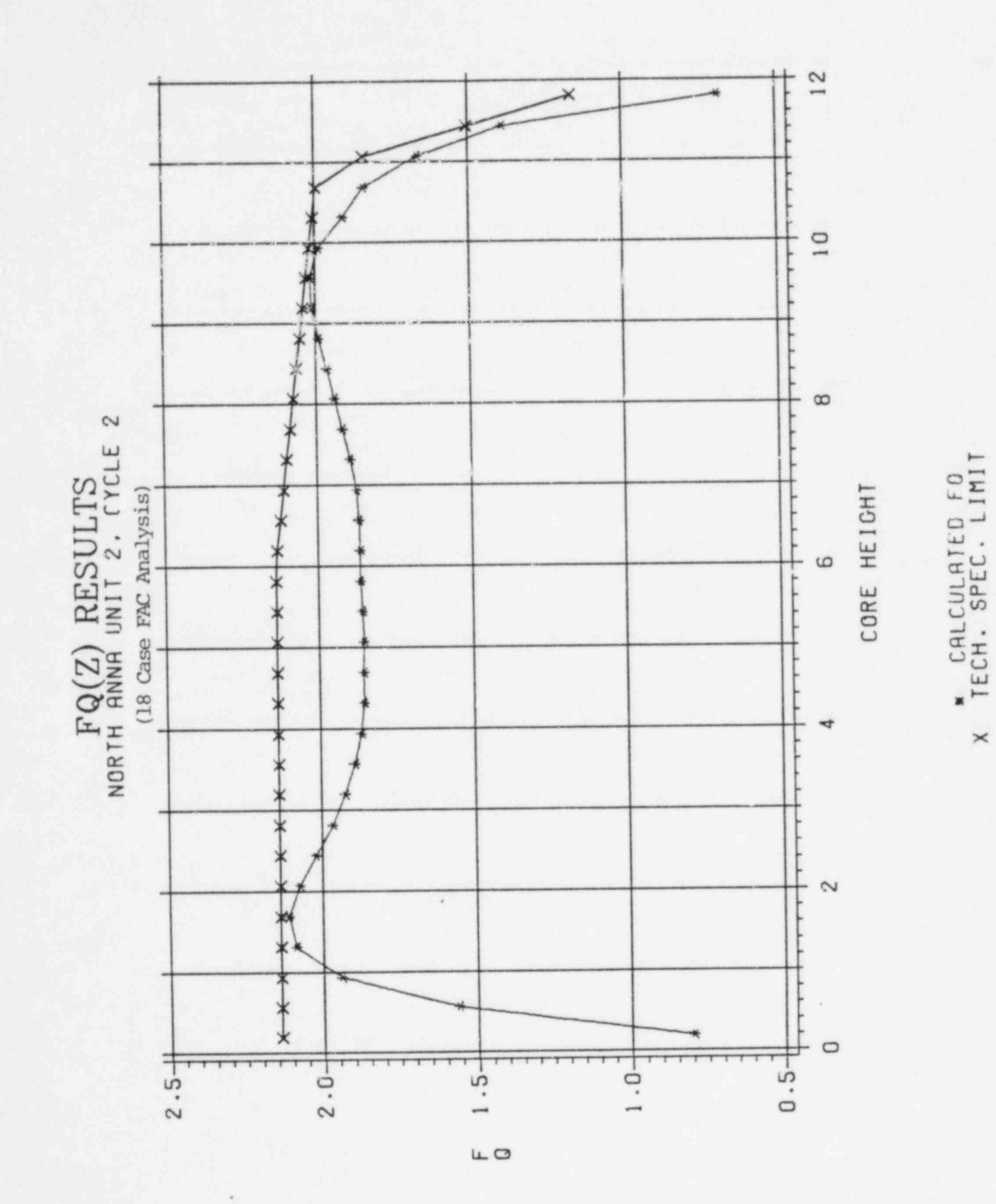

FZ(Z) RESULTS
NORTH ANNA UNIT 2, CYCLE 2
(18 Case FAC Analysis)
CORE HEIGHT
F Z
1.50
1.25
1.00
0.75
0.50
0.25
0.00
0
2
4
6
8
10
12

FIGURE 5-42



FQ(Z) RESULTS
NORTH ANNA UNIT 2. CYCLE 2
(18 Case FAC Analysis)
CORE HEIGHT
F
Q
* CALCULATED FQ
X TECH. SPEC. LIMIT
2.5
2.0
1.5
1.0
0.5
0
2
4
6
8
10
12

SECTION 6 - SUMMARY AND CONCLUSIONS

The Vepco NOMAD code and model are operational at Vepco for the purpose of performing one-dimensional reactor physics analyses and supporting the evaluation of core performance. The model consists of NOMAD with the NULIF, XSEDT, XSFIT, XSEXP, FXYZ, FDELH, and PCEDT codes providing either input or data manipulation. The accuracy of the Vepco NOMAD model has been established through extensive comparison of calculations to measurements from the units at the Surry and North Anna Nuclear Power Stations and to calculations from other Vepco codes. The results of these comparisons indicate that the Vepco NOMAD model (which includes normalization to the Vepco PDQ07 Discrete, PDQ07 One Zone, and FLAME models) provides the capability to predict core peaking factors, axial power distributions, differential and integral rod worths, and load follow maneuvers as well as perform Final Acceptance Criteria (FAC) Analysis.

Verification of and improvements to the Vepco NOMAD code and model will continue to be made as more experience is obtained through the application of the model to the units at the Surry and North Anna Nuclear Power Stations.

SECTION 7 - REFERENCES

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3. J. G. Miller, "The FDELHP01 and PCEDTP01 Codes", NFE Technical Report No. 201, July, 1981 (Virginia Electric and Power Co.).

4. P. D. Breneman, "The NULIFP01 Code", NFE Calculational Note PM-13, March, 1979 (Virginia Electric and Power Co.).

5. M. L. Smith, "The PDQ07 Discrete Model", VEP-FRD-19A, July, 1981 (Virginia Electric and Power Co.).

6. J. R. Rodes, "The PDQ07 One Zone Model", VEP-FRD-20A, July, 1981 (Virginia Electric and Power Co.).

7. W. C. Beck, "The Vepco FLAME Model", VEP-FRD-24A, July, 1981 (Virginia Electric and Power Co.).

8. L. L. Lynn, "A Digital Computer Program for Nuclear Reactor Analysis Design Water Properties", WAPD-TM-680, July, 1967 (Westinghouse Electric Corporation).

9. J. G. Miller, "Thermal-Hydraulic Feedback Input to the Two-Dimensional PDQV2 Code", NFE Calculational Note PM-19, June, 1979 (Virginia and Electric Power Co.).

10. Course Notes "Basic PWR Physics Course", September, 1980 (Westinghouse Electric Corporation).

11. F. W. Sliz, "VEPCO Reactor Core Thermal-Hydraulics Analysis Using the COBRA IIIc/MIT Computer Code", VEP-FRD-33, August, 1979.

12. D. A. Daniels, "XETRN and SMTRN", NFE Technical Report No. 112, February, 1980 (Virginia Electric and Power Co.).