# BIG ROCK POINT PHYSICS METHODOLOGY REPORT

Consumers Power Company

Jackson, Michigan

August 25, 1978

TABLE OF CONTENTS

## ABSTRACT

This report describes the methods used in the Big Rock Point Physics Model developed by Consumers Power Company. Included in this report are descriptions of the techniques used in generation of input and modeling of the core. Comparisons of plant data to results generated by the Physics Model are also included.

## 1.0 INTRODUCTION

The purpose of this report is to describe the methods used in the Big Rock Point Physics Model. The major component of the model is GRØK, a three dimensional boiling water reactor simulator code. GRØK has been used for reactor physics work for the Big Rock Point (BRP) plant since 1972.

Features of GRØK include calculation of reactivity inventory, peak heat flux, minimum critical heat flux ratio, maximum average planar linear heat generation rate, dryout times, critical control rod patterns and assembly exposure inventory.

Consumers Power Company first developed reload physics designs for Cycle 9 (1971) and has used GRØK since Cycle 10. Included in these designs are loading pattern selection, cycle lifetime, cold critical control rod pattern, rod and notch worths, shutdown margin, moderator temperature coefficient, void coefficient, doppler coefficient, liquid poison worth, scram function, misloaded assembly effects, and control rod withdrawal sequence for the entire cycle.

GRØK is also used for core follow work. The fluxwire measured core power shape is compared with GRØK's power shape. Power distribution correction factors are then calculated for various discrete regions of the core and the maximum allowable power level is determined. Plant personnel perform most of the core follow work.

## 2.0 DESCRIPTION OF BIG ROCK POINT

Big Rock Point is a 240 MWt boiling water reactor located on the shore of Lake Michigan just North of Charlevoix, Michigan. Uranium dioxide as well as mixed oxide pellets contained in zircaloy tubing constitute the fuel. There are 84 assemblies of either 9 x 9 or 11 x 11 arrays with an active fuel height of 70 inches. Most assemblies contain removable corner rods. These positions contain either zircaloy experimental rods or cobalt target rods. Figure 2-1 shows the core layout, control rod positions and fluxwire positions.

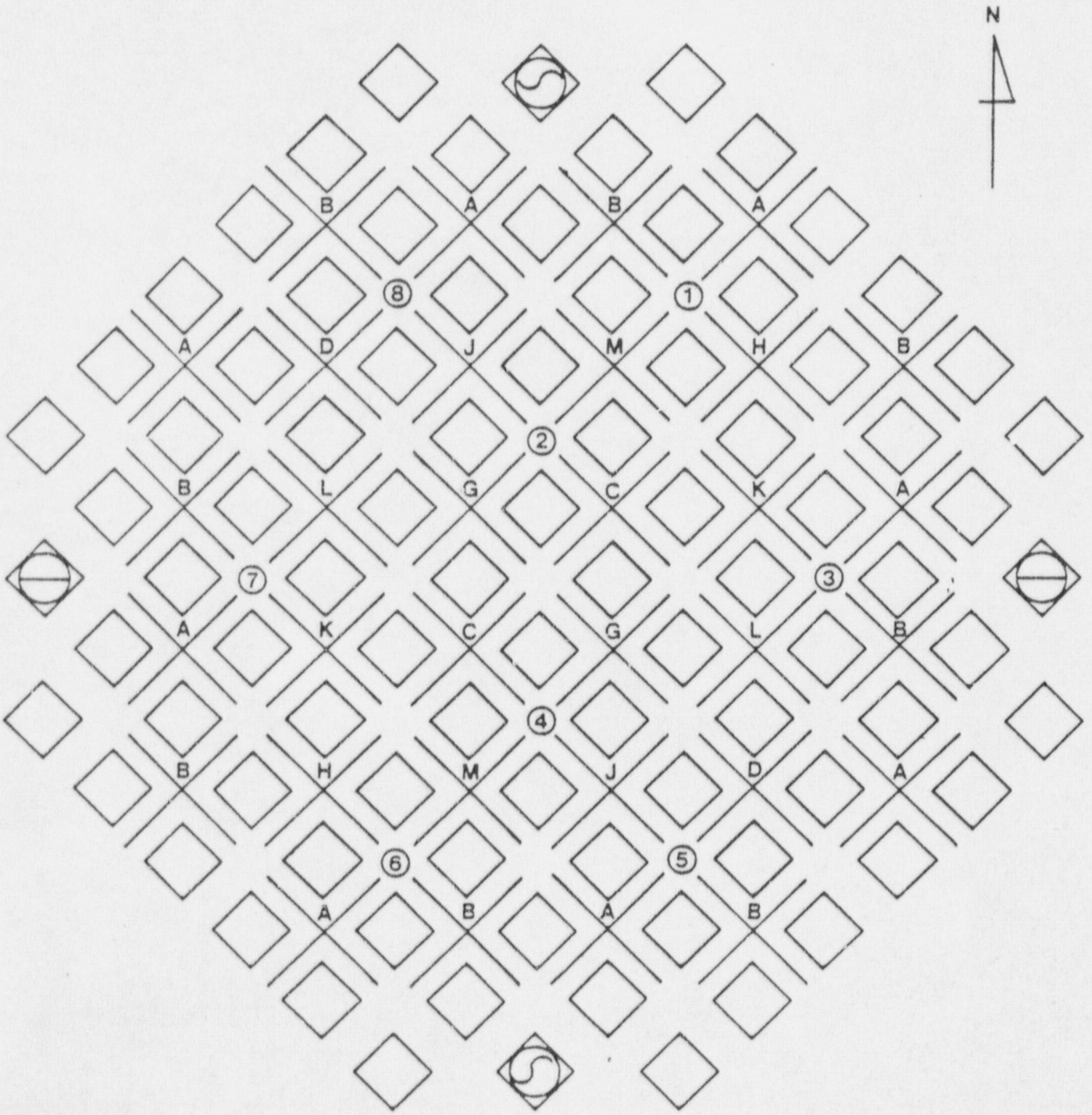
32 $B_4C$ cruciform control rods control the reactor. The 16 outer blades, groups A and B are strong (all poison tubes filled) and are used only for shutdown. The 16 inner blades are weak (40 inner poison tubes are empty). 8 or 10 of these weak blades are typically used to regulate the core at power and they are alternated each cycle. Power maneuvering is done entirely with control rods since Big Rock Point has external circulation and does not use flow control.

Big Rock Point is currently in its 15th refueling cycle.

FIGURE 2-1

BIG ROCK POINT

CORE CONFIGURATION



## 3.0 PHYSICS MODEL

### 3.1 OVERVIEW

The calculational sequence for Big Rock Point physics is diagrammed in Figure 3-1. The primary component of the sequence is the three dimensional reactor simulator program, GRØK. The remainder of the sequence primarily involves the generation of input for GRØK. The input consists of: assembly neutronics parameters as a function of local operating conditions, physics features of each fuel type, hydraulic parameters, technical specification and other limits, and the reactor operating state.

The neutronics parameters are derived from the results of assembly physics calculations performed with such programs as CASMO [1], PDQ7 [2] or XPOSE [3]. These calculations are performed either by Consumers Power or by the fuel vendor.

#### 3.1.1 GRØK Overview

The GRØK program is run using the appropriate option to compute the desired core parameter. Reactivity, power distribution and margin to thermal limits are computed in a straightforward manner by modeling the reactor operating state being analyzed. Control rod worths and notch worths are computed by calculating reactivity at two different states (i.e., rod in sequence and rod inserted). Coefficients (void and doppler) are computed with a special option that allows one feedback mechanism to be varied while the worth of the rest remain constant. Fuel burnup is incremented by inputting a core average exposure and allocating it based on the computed power distribution.

### 3.2 ASSEMBLY PHYSICS CALCULATIONS

The purpose of assembly physics calculations in the calculational sequence is

to determine the relationship between the physical state of the reactor and the neutronics parameters required in the GRØK physics model. These calculations are typically performed for a horizontal slice of a single fuel assembly, for which the region of solution is the area bounded by the centerlines of the water gaps surrounding the bundle. They are two dimensional with reflecting boundary conditions, hence the term "infinite lattice". The method of solution is few group fine mesh diffusion theory (PDQ), or transport theory (CASMO).

The inputs to the assembly physics calculations consist of the location, composition and temperature of all the important materials within the assembly; such as fuel, burnable poison and inert rods; flow channel; control rods and coolant. Dimensions and compositions of assembly components are acquired from the vendor drawings and design reports. Structural materials are assumed to be at the coolant temperature, while the assembly average fuel and cladding temperatures are computed using a one dimensional heat transfer model of a fuel rod as in the program GAPEX.[13] The important outputs from the assembly physics calculations are: the infinite neutron multiplication factor, $K_{oo}$; the migration area, $M^2$; the one-group flux; the ratio of kappa over nu; the fluxwire absorption factor and the local fuel pin power distribution.

The following paragraphs describe the computer programs used for assembly physics calculations.

### 3.2.1 CASMO

CASMO is a multigroup two dimensional transport theory code for static and burnup calculations of BWR and PWR fuel assemblies. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a diagonally symmetric

square pitch array, with allowance for gadolinium and other burnable poisons, in-core instrument channels, water gaps and cruciform control rods. CASMO incorporates libraries of multigroup neutron cross-sections and other nuclear parameters so that the calculation is a one-step process from the users standpoint. The program performs spectrum calculations on the various compositions within the assembly and generates few group cross-sections for each. These are then used in the two-dimensional neutronics calculation, which consists of a variation of transport theory called transmission-probabilities. Using the computed fluxes and few group microscopic cross-sections CASMO depletes the fuel isotopes stepwise out to the projected lifetime of the assembly.

3.2.2 PDQ7

PDQ7 solves the few-group neutron diffusion problem with burnup in one to three dimensions. Cross-section input for PDQ7 is generated using programs such as HAMMER [11], EPRI-CELL [12] or XPOSE. In the development of the cross sections a standard "pin cell" approach is used for interior fuel pins, while an extra region of water is added to the cell to account for the presence of water gaps for fuel pins on the outside row of the assembly. Diffusion theory "equivalent" cross sections are found for control rods and lumped absorbers by adjusting absorption cross sections to match reaction rates computed with a transport theory model such as CASMO. Cross sections for water gaps and structure are derived from an exterior fuel pin spectrum. PDQ7 is used for special studies that require a code with more flexability than CASMO such as 3D calculations [5], asymmetric bundles and multi-assembly configurations.

## 3.3 GRØK INPUT GENERATION

In order to compute a three dimensional reactor power distribution, the physics algorithm in GRØK requires three dimensional distributions of the basic physics parameters $K_{oo}$, $M^2$ and the ratio $\kappa/\upsilon$. These parameters in turn are dependent on the local operating state: power, steam voids, control rods and the burnup history of the fuel. These dependencies are computed separately from GRØK using assembly cell physics calculations simulating various local conditions, and the results are input to GRØK in the form of coefficients to polynomial functions.

For instance, to compute the dependence of $K_{oo}$ on local steam void fractions, assembly physics calculations are performed at three discrete void fractions, usually 0%, 25%, and 50% steam voids. The three values of $K_{oo}$ resulting from these calculations are then fit to a simple polynomial in moderator density through the solution of 3 simultaneous equations resulting in:

$$K_{oo} = B_4(1 + B_7U + B_{10}U^2) \qquad (1)$$

Where U is the moderator density relative to saturation, and $B_4$, $B_7$ and $B_{10}$ are the solved coefficients which are then input to GRØK.

Since $K_{oo}$ is also a function of control rods and the dependency on control rods and steam voids are interrelated (non-separable), six more assembly physics calculations are performed resulting in two more equations similar to (1), one for one adjacent control rod, and another for two adjacent control rods. For partially roded nodes, GRØK interpolates between the two appropriate evaluated equations.

The reactivity worth of other reactor operating parameters such as fuel burnup, burnable poison concentration, doppler and equilibrium xenon concentration is also computed with assembly physics calculations. These worths are expressed in terms of $\Delta K/K$ vs nodal power, exposure or void weighted exposure, where

$$\Delta K/K = \frac{K_1 - K_2}{K_1} \qquad (2)$$

$K_1$ is the "reference" value of $K_{oo}$ and $K_2$ is the "perturbed" value where the perturbation is due to changes in the operating parameter, i.e., doppler, xenon etc. The calculated value of $K_{oo}$ is then modified by a series of multipliers of the form $(1 - \Delta K/K)$ for each factor affecting reactivity. This procedure is repeated for each individual node in the calculation resulting in the three dimensional array of physics parameters required for the neutronics calculation.

Table I is a compilation of the physics parameters evaluated for each fuel type in the generation of GRØK input. Column 1 is the desired physics parameter, columns 2 and 3 indicate what physical parameters are considered in the calculation of the physics parameter, columns 4 and 5 indicate the degree of polynomial used in GRØK to describe the relationship, and column 6 shows the number of assembly physics calculations required to generate the coefficients. Many of the assembly calculations are used for more than one purpose, so the total number of calculations performed is less than the total indicated by adding column 6.

TABLE I

Input for Power Operation

| Neutronics Parameter | Functional A | Dependence B | Degree of Fit A | Degree of Fit B | Minimum Number of Assembly Calculations |
|---|---|---|---|---|---|
| $M^2$ | Steam Voids | -------- | 2 | - | 3 |
| $K_{oo}$ | Control Rods | Steam Voids | 1 | 2 | 9 |
| $\Delta K/K$) Doppler | Power | Steam Voids | 1 | 1 | 3 |
| $\Delta K/K$) Equilibrium Sm | Power | -------- | 0 | - | 2 |
| Xenon Cross Sections | Burnup | -------- | 2 | - | 3 |
| $\Delta K/K$) Xenon | Xenon Concentration | -------- | 2 | - | 3 |
| $\Delta K/K$) Initial Burnable Poison | Control Rods | Steam Voids | 1 | 2 | 9 |
| $\Delta K/K$) Burnup | Burnup | Void-Weighted Burnup | 2 | 2 | 30 |
| Fraction of B.P. Remaining | Burnup | -------- | 2 | - | 6 |
| $\kappa/\nu$ | Burnup | -------- | 2 | - | 10 |

Input for Zero Power

| | | | | | |
|---|---|---|---|---|---|
| $M^2$ | Temperature | -------- | 2 | - | 3 |
| $K_{oo}$ | Control Rods | Temperature | 1 | 2 | 18 |
| $\Delta K/K$) Peak Samarium | Burnup | -------- | 0 | - | 2 |
| $\Delta K/K$) Initial Burnable Poison | Control Rods | Temperature | 1 | 2 | 18 |

Xenon, Burnup, Fraction of Burnable Poison remaining and $\kappa/\nu$ are the same as above.

## 3.4 DESCRIPTION OF THE GRØK COMPUTER PROGRAM

The following is a description of the calculations performed in the GRØK computer program. GRØK incorporates the FLARE [4] physics model along with a thermal-hydraulic feedback model, a local peaking factor calculation, and routines which evaluate margin to thermal limits.

### 3.4.1 GRØK Physics Model

GRØK calculates a nodal power density for a three dimensional core geometry. The code is based on modified one group diffusion theory using the infinite multiplication factor ($K_{oo}$) and migration area ($M^2$) as the basic physics inputs. An albedo at the core surface simulates the reflector so that only mesh points within the active fuel region are considered. Each assembly is represented by one horizontal and nine vertical mesh points (nodes).

The complete iterative calculation consists of four levels:

1. Source or Power Distribution,
2. Void Distribution,
3. Control Rod Positioning and Power Variation, and
4. Fuel Burnup or Thermal Limits Evaluation.

A description of each level follows. An explanation of the equations solved also appear.

#### 3.4.1.a Source or Power Iteration

The neutron source at each node is calculated as a function of:

1. $K_{oo}$ at that point,
2. the neutron source at the six neighboring points, and
3. a transport kernel.

The transport kernel is a measure of the probability that a neutron born at node m is absorbed at node ℓ and is a function of migration area and node spacing. $K_{oo}$ is calculated at each node and includes the following effects:

1. Presence or absence of one or two adjacent control rods. These control rods can be strong or weak in any combination.
2. Local moderator density or coolant temperature.
3. Power dependent xenon and Doppler reactivity.
4. Local fuel exposure.
5. Presence of burnable poisons.
6. Equilibrium or peak samarium.

$M^2$ is also calculated at each node as a function of local moderator density or temperature. The initial values of $K_{oo}$ and $M^2$ are based on a flat power distribution and are updated thereafter following each void iteration.

3.4.1.a.1 Source Equations

Source at node ℓ is defined by:

$$S_\ell = K_{oo_\ell} A_\ell \qquad (3\text{-}3)$$

where $A_\ell$ = absorption rate at ℓ *

or $$A_\ell = \sum_m{}' S_m W_{m_\ell} + S_\ell W_{\ell\ell} \qquad (3\text{-}4)$$

---

* note: a single subscript ℓ or m will be used interchangeably with the subscript $_{ijk}$.

where $W_{m_\ell}$ = probability that a neutron born in node m will be absorbed in node $\ell$ and the prime indicates summation over the six nearest neighbors. Combining these equations and dividing $K_{oo}$ by $\lambda$ (the eigenvalue) yields:

$$S_\ell = \frac{K_{oo_\ell}\left[\sum' S_m W_{m_\ell}\right]}{\lambda + K_{oo_\ell}\left[(6 - \alpha_\ell) W_{\ell_m} - 1\right]} \qquad (3\text{-}5)$$

where $\lambda$ = eigenvalue

$\alpha_\ell$ = albedo

$\alpha_\ell$ is non-zero at the boundaries of the core (<6 neighbors) only and is handled separately for the top reflector, bottom reflector, and for the sides of each peripheral fuel assembly. $\lambda$ is recalculated after each iteration based on a solution to the neutron balance summed over the entire core:

$$\lambda = \frac{\text{Source + Inleakage - Outleakage}}{\text{Absorption}} \qquad \text{or,}$$

$$\lambda = \frac{\sum_\ell S_\ell + \sum_m' S_m W_{m_\ell} - \sum_\ell S_\ell W_{\ell m}(n_\ell - \alpha_\ell)}{\sum_\ell \frac{S_\ell}{K_{oo_\ell}}} \qquad (3\text{-}6)$$

where $n_\ell$ = number of external nodes adjacent to node $\ell$, both $n_\ell$ and $\alpha_\ell$ are zero for all internal nodes, and the sums are over all nodes in the reactor.

Equations 3-5 and 3-6 are the basic equations solved during the source iteration. The transport kernel is defined as:

$$W_{\ell m} = \frac{\sqrt{M^2_\ell}}{2r_{\ell m}} \qquad (3\text{-}7)$$

where $M^2_\ell$ = migration area of node $\ell$

$r_{\ell m}$ = mesh spacing.

A more detailed analysis of these equations are presented in Section 3 of reference 4.

Several source iterations are performed per void iteration. After each series of source iterations the nodal power distribution is computed from the neutron source by multiplying by the ratio of kappa (the effective energy released per fission) over nu (the average neutron yield per fission). A polynomial function of $\kappa/\upsilon$ versus burnup is included in GRØK for each fuel type.

3.4.1.a.2 Physics Parameters - $K_{oo}$ and $M^2$

Two basic neutronics parameters are used to calculate the reactor fission source distribution: $K_{oo}$, the infinite multiplication factor, and $M^2$, the neutron migration area. A three dimensional array of these quantities is generated by the program using algebraic functions that describe the dependence of $K_{oo}$ and $M^2$ on local reactor conditions. The coefficients of these functions are input for each fuel type and are derived from the results of assembly physics calculations performed over the range of expected local operating conditions. The $K_{oo}$ and $M^2$ arrays are reevaluated after each void iteration and used as input for the next series of source iterations.

There are two reactor conditions normally simulated separately using GRØK: zero power and power operation. For zero power calculations, $M^2$ is calculated as a quadratic function of isothermal reactor temperature ranging from ambient conditions up to the operating saturation temperature. For power operation calculations, $M^2$ is computed as a function of in-channel steam voids ranging from zero to core exit void fraction at full power. $K_{oo}$ varies with void fraction, control

rod configuration, burnable poison concentration, fuel exposure, fuel temperature (nodal power) and xenon and samarium concentrations.

For zero power calculations $K_{oo}$ is required as a function of reactor temperature for the following combinations of control rods inserted into a node:

1. no control rods,
2. one weak control rod,
3. one strong control rod,
4. two weak control rods,
5. two strong control rods, and
6. one weak and one strong control rod.

The core locations of the strong versus weak rods do not change and therefore have been built into the program. Since the strong control rods are withdrawn before significant power levels are achieved, $K_{oo}$'s for power operation are computed as a function of steam voids for the following fractions of control only:

1. no control rods,
2. one weak control rod, and
3. two weak control rods.

If the fuel type in question contains burnable poisons, coefficients to polynomials are input that describe $\Delta K_{oo}/K_{oo}$ due to the burnable poison as a function of control density and moderator condition in the same manner as $K_{oo}$. Since the burnable poison typically does not cover the entire active length of the assembly, parameters are input that describe the axial position of the poison.

Functions are also provided that describe the reactivity defect due to xenon as a function of nodal power. The equilibrium xenon-135 and iodine-135 concentrations are calculated for each node according to:

$$X_{eq\ell} = \frac{(\sigma_I + \sigma_X)\ \phi\Sigma_f\ P \cdot S_\ell}{\lambda_X + \phi\sigma_a^X} \qquad (3\text{-}8)$$

$$I_{eq\ell} = \frac{\sigma_I\ \phi\Sigma_f\ P \cdot S_\ell}{\lambda_I} \qquad (3\text{-}9)$$

Where P is the reactor power and $\lambda$ is a decay constant. The quantities $\phi\Sigma_f\gamma_I$, the iodine yield at rated power in an average node; $\phi\Sigma_f\sigma_X$, the xenon yield; and $\phi\sigma_a^X$, the xenon absorption rate, are calculated as a function of fuel burnup based on input coefficients for each fuel type. Transient xenon after shutdown is calculated from the following:

$$X_{\ell t} = X_{eq\ell}e^{-\lambda Xt} + \frac{\lambda I\ I_{eq\ell}\,(e^{-\lambda Xt} - e^{-\lambda It})}{\lambda_I - \lambda_X} \qquad (3\text{-}10)$$

Where t is the time after shutdown. The reactivity worth in $\Delta K/K$ of xenon is then computed as a function of the calculated xenon concentration using input coefficients.

The program also accounts for the reactivity defect due to samarium. For power operation calculations an equilibrium value is used, while for zero power calculations either the peak samarium defect is used or zero defect depending on whether the assembly has been previously burned or is fresh fuel.

The reactivity defect due to doppler is computed as a function of local power and steam voids. The two basic relationships used to calculate this function are:

1. the variation of effective fuel temperature with LHGR, and
2. the reactivity defect from doppler as a function of fuel temperature at different void conditions.

GRØK calculates the doppler defect as a linear function of power multiplied by a linear function of steam voids as specified by input coefficients for each fuel type.

The $K_{oo}$ vs exposure and void-weighted exposure equation is of the following form:

$$\left.\Delta \frac{K}{K}\right|_{\text{Burnup}} = \left[B_{22}E_{\ell}\ (1.0 - B_{23}E_{\ell}\ ) - B_{24}V_{\ell}\ (1.0 - B_{25}V_{\ell}\ )\right] \frac{\sqrt{E_{\ell}}}{B_{26} + \sqrt{E_{\ell}}} \qquad (3\text{-}11)$$

where

$E_{\ell}$ = exposure at node $\ell$

$V_{\ell}$ = void history at node $\ell$ (product of exposure and voids)

The coefficients for equation 3-11 are calculated by a two dimensional least squares fit of computed $K_{oo}$ versus burnup out to at least 1.5 times the expected average discharge exposure for three different void conditions (usually 0, 25 and 50%). For burnable poison fuel, a polynomial function is provided that describes the burnup-dependent behavior of the reactivity worth of the poison.

3.4.2 Void Iteration

The void model in FLARE has been entirely replaced by a more sophisticated thermal hydraulics model. It requires the core active coolant flow, coolant inlet enthalpy, reactor vessel outlet pressure, total reactor power, and the nodal relative power distribution calculated in the source loop. Spacer loss coefficients, core inlet friction factors, hydraulic diameters, wetted perimeters and thermal hydraulic model selections are required user input data. These parameters are obtained from analyses and test data from both the vendor and Consumers Power. The reactor core is hydraulically modeled with closed flow channels each containing one fuel assembly.

An iterative solution technique is used where the total core active coolant flow is uniformly apportioned to each flow channel in the reactor core. The total pressure gradient, made up of components due to friction, momentum, elevation, and local losses, through each flow channel is then calculated using this estimate.

The average pressure gradient for the reactor core is then calculated and each channel flow rate is modified as follows:

$$Wnew\ (I,J) = Wold\ (I,J)\sqrt{\frac{Pave}{P(I,J)}} \qquad (3\text{-}12)$$

where: Wnew (I,J) = New estimate of flow rate in flow channel (I,J).

Wold (I,J) = Previous estimate of flow rate in flow channel (I,J).

Pave = Average pressure gradient for the reactor core.

P(I,J) = Pressure gradient of flow channel (I,J).

After the channel flow rates are modified they are summed and normalized to the known total core active coolant flow. This new estimate of channel flow rates is then used to recompute the flow channel pressure gradients. The procedure is repeated for each power/flow iteration.

Nodal steam void fraction is calculated from nodal quality using the correlation presented as equations 3-17 of reference 4 and is converted to relative moderator density. This is needed to calculate $K_{oo}$ and $M^2$ for the next series of source iterations. The void and source iterations are continued until a converged power and void distribution is attained.

### 3.4.3 Control Rod Positioning and Power Variation

GRØK has a control loop which will automatically adjust control rod groups one notch at a time in any given input sequence until the core reactivity equals a pre-set value. An automatic search for the reactor power required for criticality at any given control rod position is also available.

### 3.4.4 Fuel Burnup

GRØK updates nodal fuel burnups using the computed power distribution, the assembly weights as input by fuel type, and the core average exposure increment for the burnup step.

## 3.5 PEAKING FACTOR ALGORITHM

The peaking factor correlation's function in GRØK is to find peak pin powers for each assembly, which are necessary inputs to the MCHFR, MCPR and heat flux calculations. The code first takes the coarse global power shape and finds a

polynomial function describing the axial power shape in each node via a SPLINE curve fit. These functions are evaluated at four points per node. This method describes in finer detail the axial power variations that occur over the length of the assembly and results in a computed power for every two inches of core height. These will be referred to as expanded or sub-nodal powers.

The local fuel pin power is then obtained by multiplying the expanded nodal powers by a local peaking factor. These local peaking factors are calculated by combining the gross horizontal power tilt across an assembly at each axial sub-node with the beginning of life (BOL) assembly infinite lattice local power distribution. The infinite lattice local power distributions are derived by means of two dimensional assembly physics calculations for each fuel type, and are input for three assembly control conditions:

1. uncontrolled,
2. singly weak controlled, and
3. doubly weak controlled.

The program will orient the local power distributions for controlled assemblies based on which corners of the assembly are adjacent to control rods. If a sub-node contains a control rod tip, the controlled power distribution is chosen. For the sub-node immediately above the control rod tip, the following equation is used to calculate the pin powers:

$$PP = 0.47 \times PP_u + 0.53 \times PP_c \qquad (3\text{-}13)$$

where PP = pin power

u = uncontrolled

c = controlled

This equation is conservatively based on a three dimensional physics calculation of the power peaking at the tip of a control rod. [5]

The gross horizontal power tilt across each assembly at each sub-node is calculated by fitting the one group neutron flux in the assembly with the flux in the assembly on either side to a quadratic equation. This equation is solved at the center line of each row of fuel pins in both the X and Y directions and the result is multiplied by the infinite lattice locals to yield total local peaking factors. For assemblies on the edge of the core a straight line interpolation is performed with the one adjacent assembly.

The flux is calculated from the sub-nodal power and polynomial equations that describe the variation of the flux to power ratio ( $\frac{1}{K\Sigma_f}$ ) as a function of steam voids, control and fuel exposure. The coefficients for these equations are derived from the assembly physics calculations and are input to GRØK.

After the nodal local pin powers are calculated, the program selects the highest one. The highest value between the maximum infinite lattice local power and the maximum total local power including tilt is then chosen as the local peaking factor for that sub-node and is later used in the calculation of peak heat flux.

## 3.6 FLUXWIRE CALCULATIONS

GRØK includes a fluxwire activation computation for comparison of the calculated core power shape with fluxwire measurements. Coefficients to functions that describe the ratio of the neutron activation in a copper fluxwire located in the corner of an assembly to the assembly power as a function of steam voids,

control rod position, and assembly exposure are input for each fuel type. This data is obtained from two dimensional assembly physics calculations. The results from the four assemblies adjacent to the fluxwire position are averaged and the axial shape is printed out. The total activations for the eight fluxwires are normalized and printed out also. The fluxwire positions can be seen in Figure 2-1.

## 3.7 THERMAL HYDRAULIC LIMITS

A thermal limit calculation to determine the peak heat flux and the minimum critical heat flux ratio (MCHFR) in the core is performed by GRØK. Local peak heat fluxes for each sub-node are calculated based on total core power, normalized power shape, local peaking factors and the heat transfer area for each fuel type. Hot channel flow reduction and enthalpy rise factors are included in this calculation. The synthesized Hench-Levy correlation [6] is then applied to yield critical heat flux and MCHFR. Another thermal limit found is the minimum critical power ratio (MCPR) as calculated by the Exxon Nuclear XN2 correlation. [7] MCPR is expected to replace MCHFR as a technical specification limit in the near future, and at that time MCHFR will no longer be calculated.

The ratio of calculated to allowable maximum average planar linear heat generation rate (MAPLHGR) for each sub-node of every assembly is found based on total core power, normalized power shape and the limits for MAPLHGR input by fuel type as a function of exposure. The maximum fraction of MAPLHGR limit for each assembly and fuel type is then determined as shown by equations 3-14 and 3-15.

$$Z = \frac{(X)(F)(B)}{137.1441\ (PTH)} \qquad (3\text{-}14)$$

where $Z$ = maximum allowable radial times axial for each quarter node

$F$ = active fuel length

B = number of fuel rods per assembly

X = MAPLHGR limit at each node based on exposure of that node and a linear interpolation between input MAPLHGR limits

PTH = core thermal power

$$137.1441 = \frac{.96 * 10^3 \frac{kw}{MW} * 12 \frac{in}{ft}}{84 \text{ assemblies}}$$

where .96 = ECCS gamma smearing factor.

$$\text{Fraction of MAPLHGR limit} = \frac{P}{Z} \qquad (3\text{-}15)$$

where P = actual radial times axial.

GRØK computes the duration of time it takes each assembly to uncover during a loss-of-coolant accident and compares this to the time calculated in the LOCA analysis. Actual assembly dryout times are calculated by using the core thermal power, the radial power distribution and the Modified Armand [8] steady state void fraction. These times are compared to the dryout times assumed in the LOCA analysis corresponding to the MAPLHGR limit which is most closely approached in each assembly. The following set of equations are used to calculate dryout times. [9]

$$\Delta t = \frac{0.31 \, \rho_f \, h_{fg}}{Q'''} \qquad (3\text{-}16)$$

where $\Delta t$ = time to boiling transition

$\rho_f$ = saturated liquid density at initial operating pressure

$h_{fg}$ = enthalpy of vaporization at initial operating pressure

$$Q''' = \frac{\dot{q}}{V(1 - \alpha_i)}$$

where $\dot{q}$ = initial steady state power generation in the bundle

V = active coolant volume

$\alpha_i$ = initial steady state void fraction averaged over entire fuel rod length (as calculated by the Modified Armand model)

Equation 3-12 reduces to:

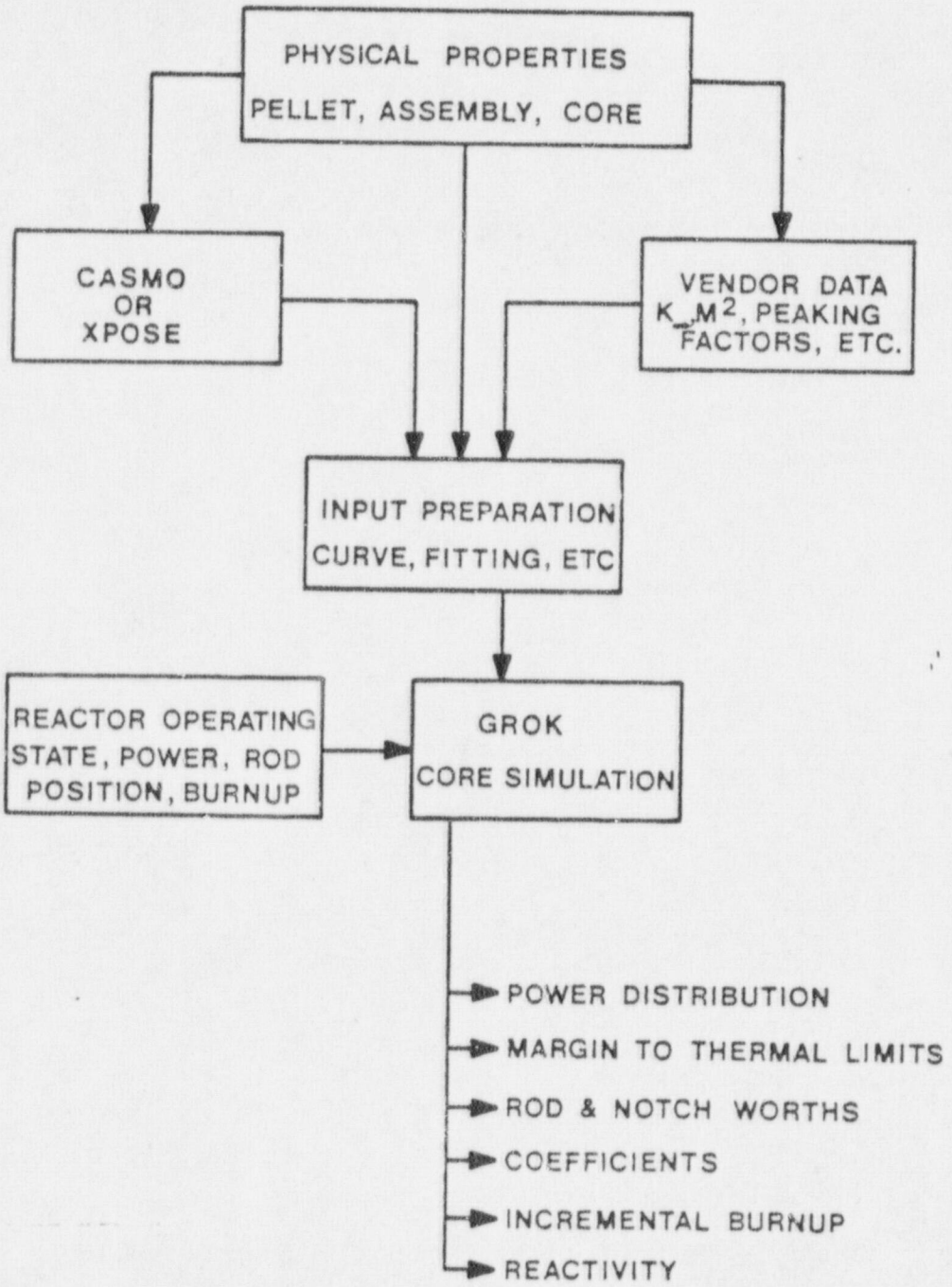
$$\Delta t = \frac{7920.7168}{Q'''} \qquad (3\text{-}17)$$

MAPLHGR limits are from the Technical Specifications as derived from LOCA analyses.

FIGURE 3-1

BRP PHYSICS

CALCULATIONAL SEQUENCE



## 4.0 VERIFICATION OF THE BIG ROCK POINT PHYSICS MODEL

The results achieved from the BRP Physics Model are examined in this section. Comparisons with reactor operating data are made for the BOC cold critical control rod patterns and fluxwire measurements. Also, the calculated core multiplication factor versus core burnup is presented. Data is presented from the last two operating cycles as the physics models used in the analysis of those cycles most closely represent the current procedures as outlined in this report.

### 4.1 BOC COLD CRITICAL CONTROL ROD PATTERN

BOC 14 and 15 actual and predicted cold critical control rod patterns are shown in Figures 4-1 and 4-2. Comparisons of actual to predicted cold critical rod patterns show that GRØK predicted correctly the total notches withdrawn at the beginning of cycle 15 and was within three notches for cycle 14.

### 4.2 AXIAL PROFILES

The ability of GRØK to predict axile power profiles is verified by comparing actual fluxwire results to profiles generated by GRØK. An explanation of the techniques used by GRØK to generate power profiles at the various fluxwire locations is given in Section 3.6. Comparisons of fluxwire profiles to those of GRØK are shown in Figures 4-3 to 4-34. Included in these figures are comparisons for BOC 14, EOC 14, BOC 15 and MOC 15. These Figures show consistantly good agreement between the calculated and measured shapes.

## 4.3 CORE MULTIPLICATION FACTOR VERSUS EXPOSURE

Figure 4-35 shows the calculated core multiplication factor as a function of exposure for cycles 14 and 15. This figure indicates that the physics model for Big Rock Point is consistant in its representation over the wide range of core conditions existing during the course of a cycle, including variations in control rod density, burnable poison concentrations, void density, reactor power, and fuel exposure. The low calculated reactivity at the beginning of each cycle is attributed to samarium and xenon nonequilibrium conditions.

FIGURE 4-1

ACTUAL VS PREDICTED BOC 14 COLD CRITICAL

ROD POSITIONS (23=OUT)

| | A | B | C | D | E | F |
|---|---|---|---|---|---|---|
| 1 | | 5 | 7 | 5 | 6 | |
| 2 | 6 | 0 | 0 | 0 | 0 | 5 |
| 3 | 5 | 0 | 0 | 0 | 0 | 7 |
| 4 | 7 | 0 | 0 | 0 | 0 | 5 |
| 5 | 5 | 0 | 0 | 0 | 0 | 6 |
| 6 | | 6 | 5 | 7 | 5 | |

PREDICTED CRITICAL

TOTAL NOTCHES WITHDRAWN = 92

| | A | B | C | D | E | F |
|---|---|---|---|---|---|---|
| 1 | | 5 | 7 | 5 | 7 | |
| 2 | 7 | 0 | 0 | 0 | 0 | 5 |
| 3 | 5 | 0 | 0 | 0 | 0 | 7 |
| 4 | 7 | 0 | 0 | 0 | 0 | 5 |
| 5 | 5 | 0 | 0 | 0 | 0 | 7 |
| 6 | | 6 | 5 | 7 | 5 | |

ACTUAL CRITICAL

TOTAL NOTCHES WITHDRAWN = 95

FIGURE 4-2

ACTUAL VS PREDICTED BOC 15 COLD CRITICAL ROD POSITIONS (23=OUT)

| | A | B | C | D | E | F |
|---|---|---|---|---|---|---|
| 1 | | 5 | 5 | 4 | 5 | |
| 2 | 5 | 0 | 0 | 0 | 0 | 5 |
| 3 | 4 | 0 | 0 | 0 | 0 | 5 |
| 4 | 5 | 0 | 0 | 0 | 0 | 4 |
| 5 | 5 | 0 | 0 | 0 | 0 | 5 |
| 6 | | 5 | 4 | 5 | 4 | |

PREDICTED CRITICAL
TOTAL NOTCHES WITHDRAWN = 81

| | A | B | C | D | E | F |
|---|---|---|---|---|---|---|
| 1 | | 5 | 5 | 4 | 5 | |
| 2 | 5 | 0 | 0 | 0 | 0 | 5 |
| 3 | 4 | 0 | 0 | 0 | 0 | 5 |
| 4 | 5 | 0 | 0 | 0 | 0 | 4 |
| 5 | 4 | 0 | 0 | 0 | 0 | 5 |
| 6 | | 5 | 4 | 5 | 5 | |

ACTUAL CRITICAL
TOTAL NOTCHES WITHDRAWN = 81

FIGURE 4-3

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 14

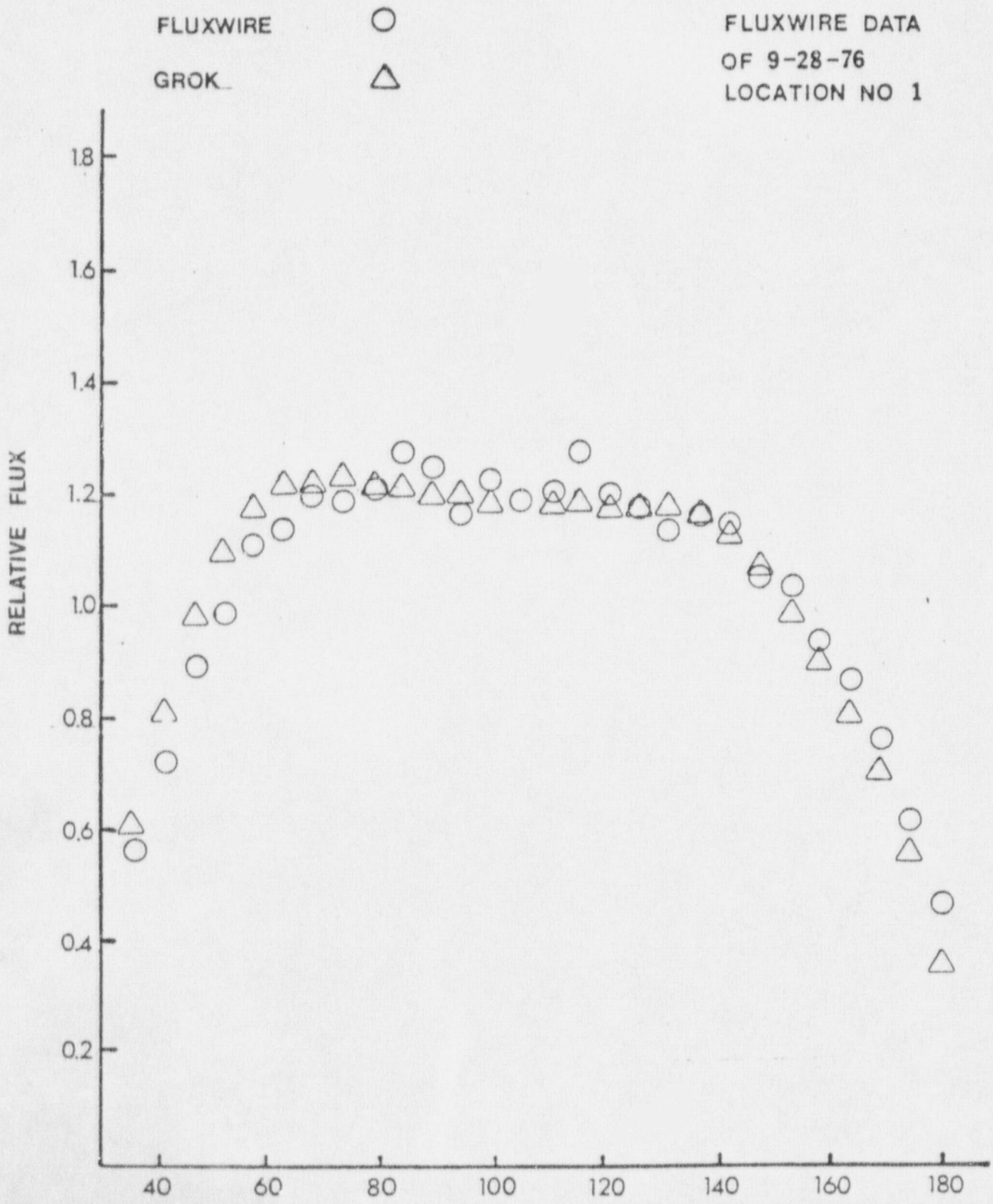


FIGURE 4-4

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

BOC 14

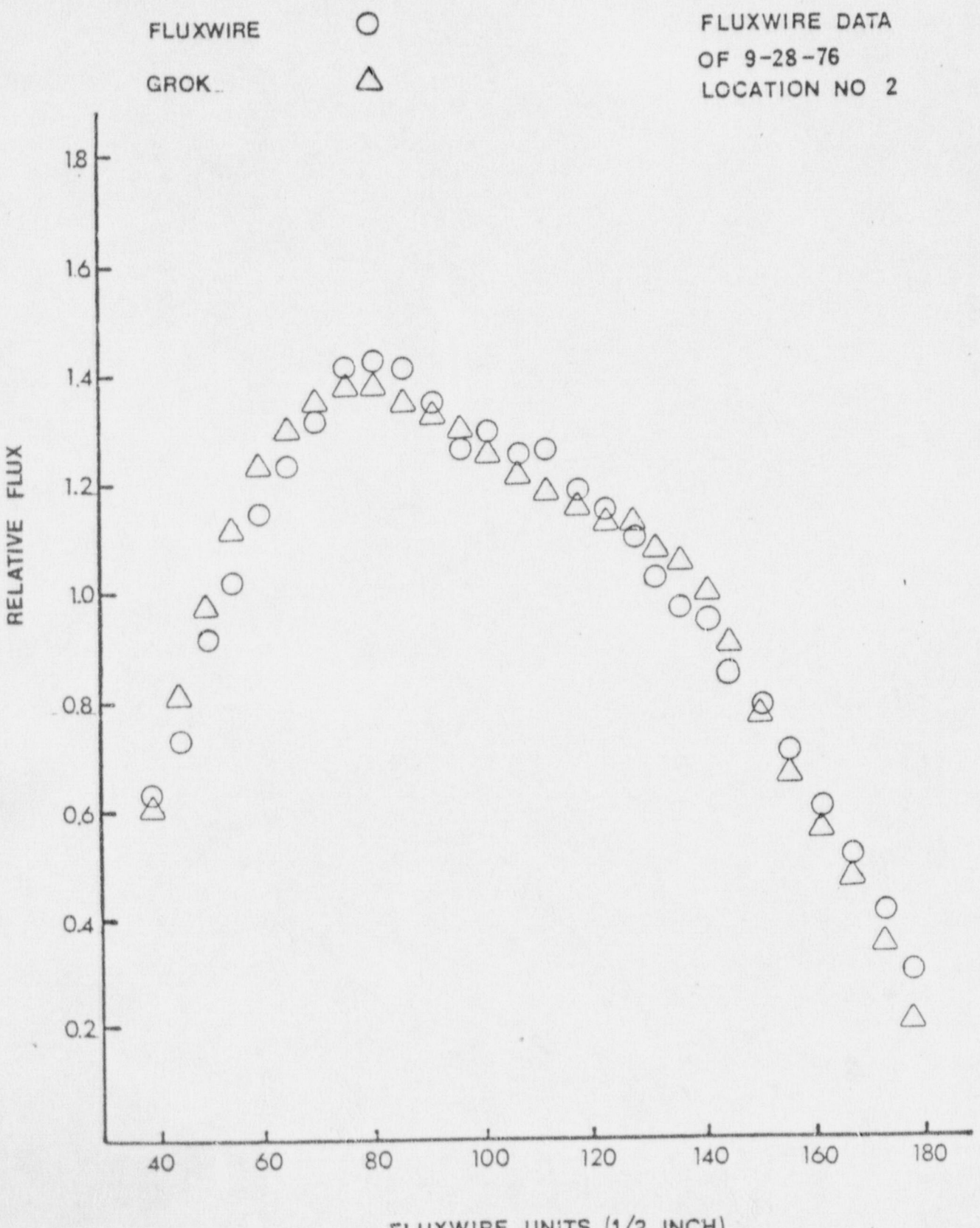


FIGURE 4-5

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 14

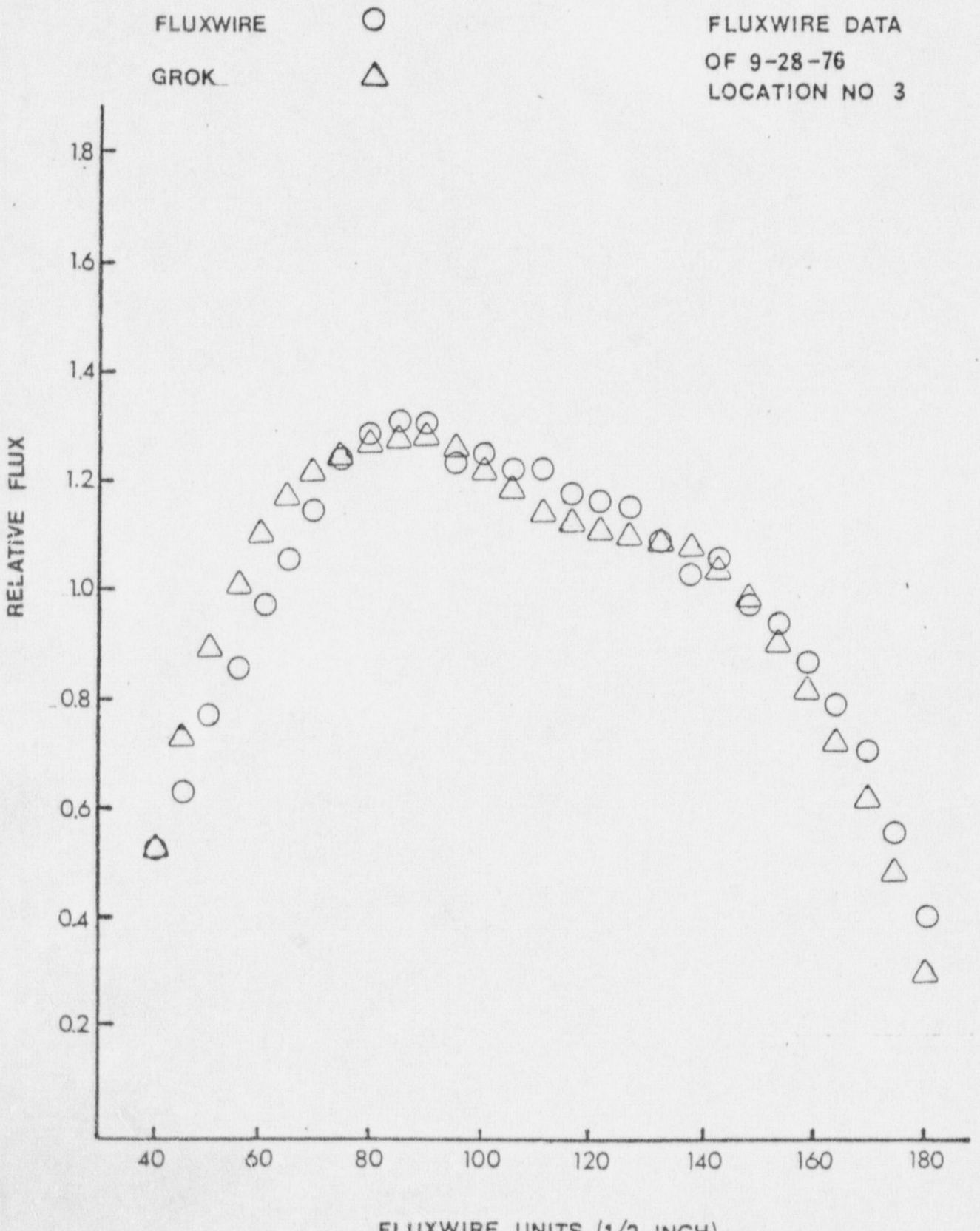


FIGURE 4-6

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

BOC 14

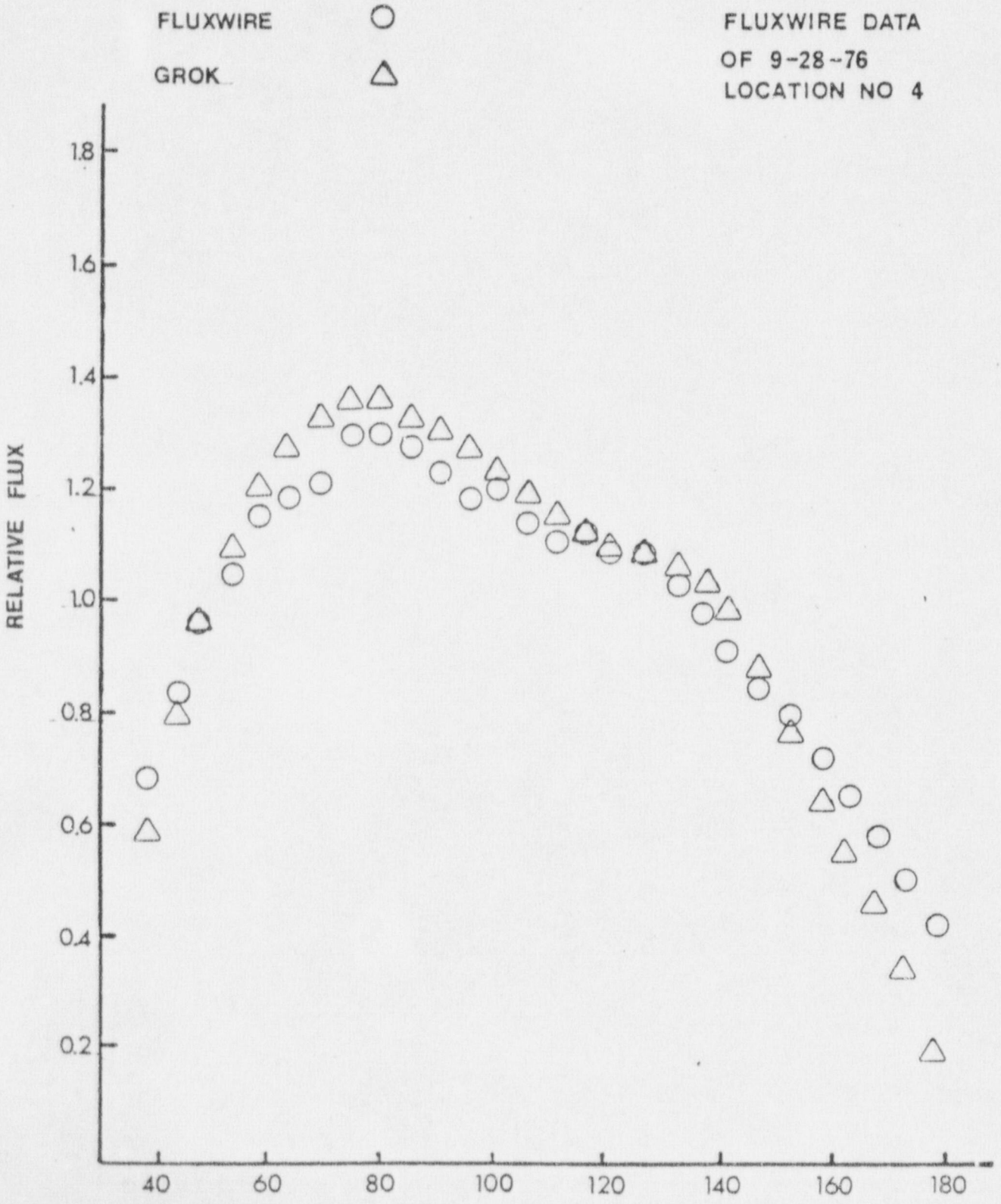


FIGURE 4-7

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 14

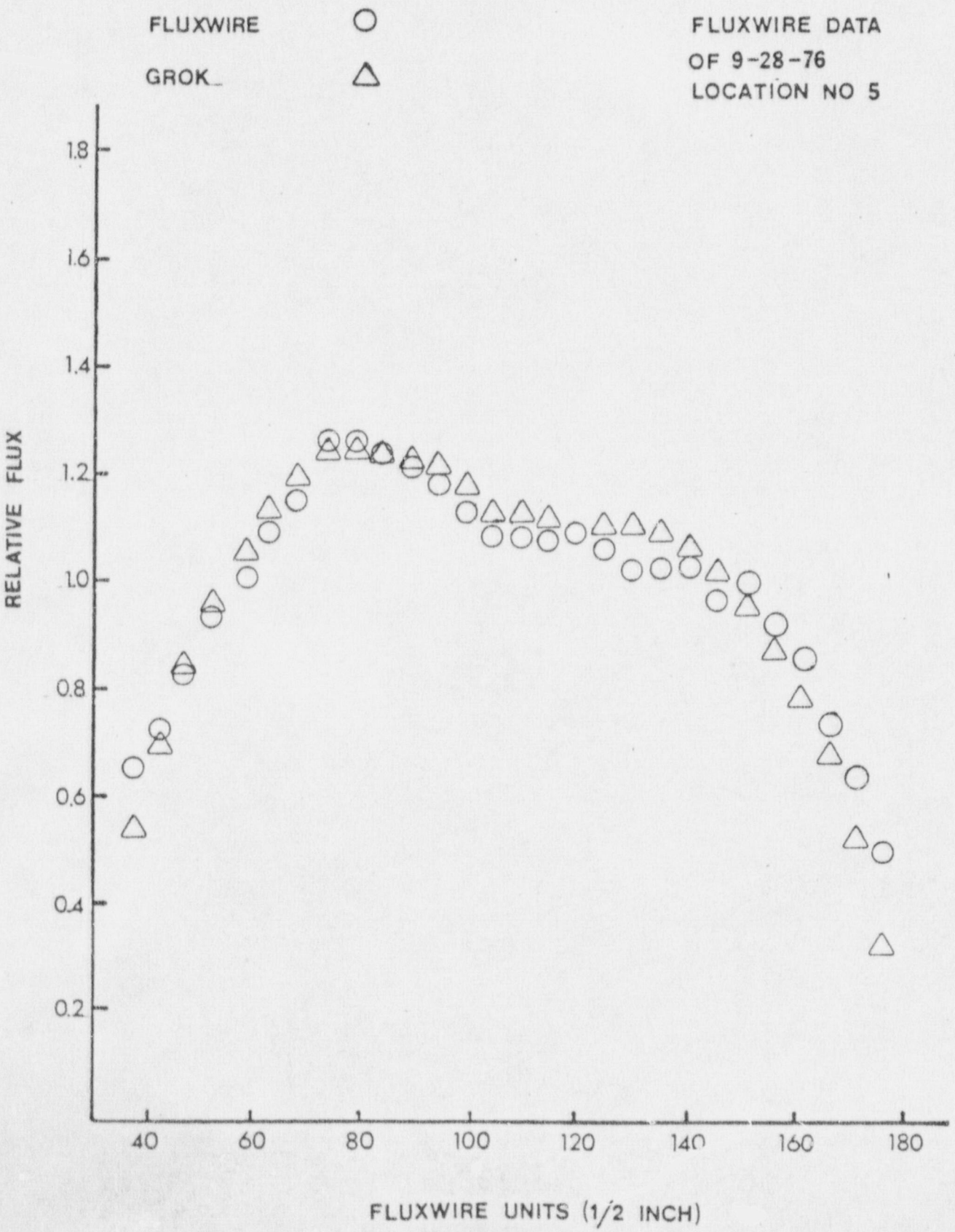


FIGURE 4-8

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 14

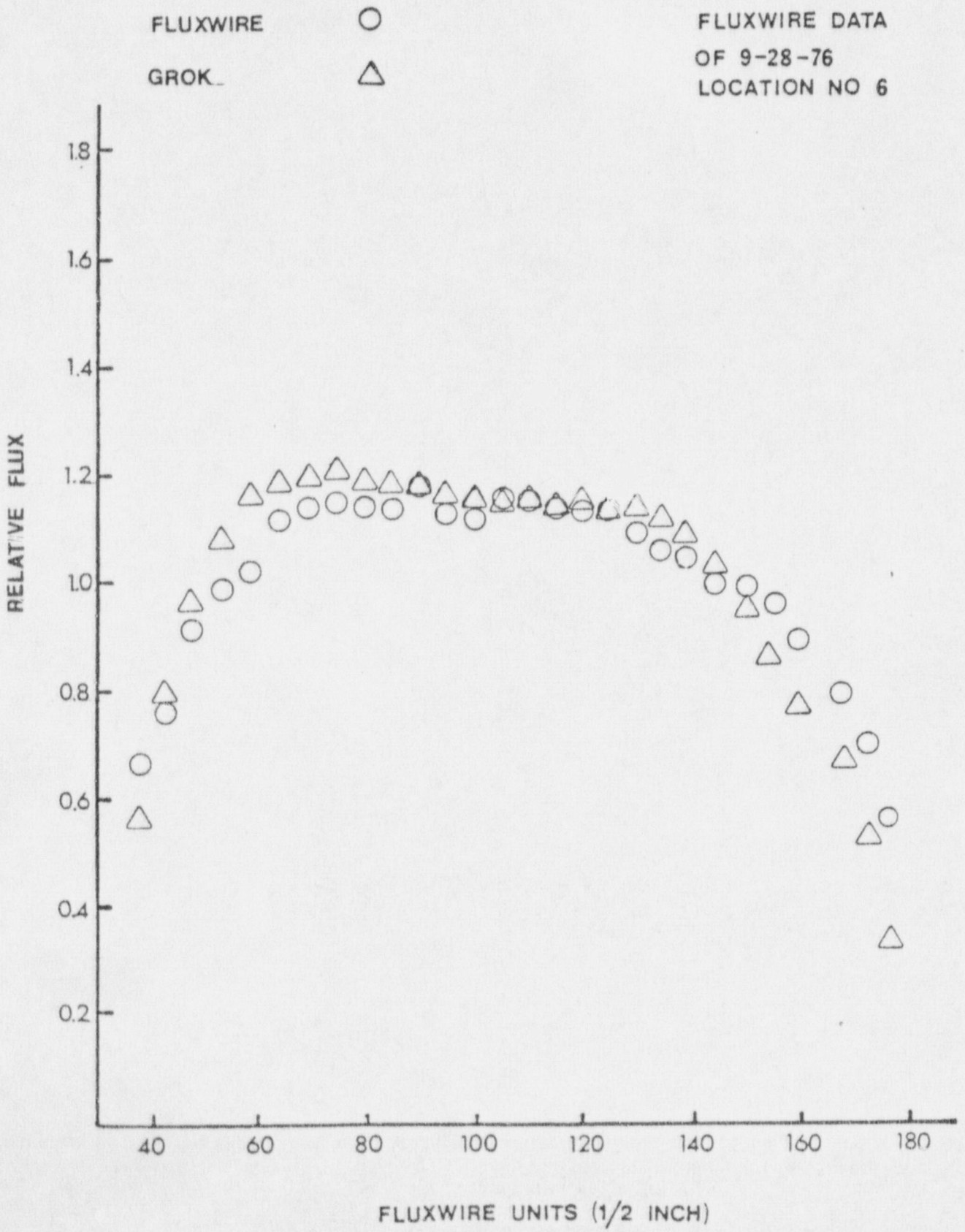


FIGURE 4-9

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 14

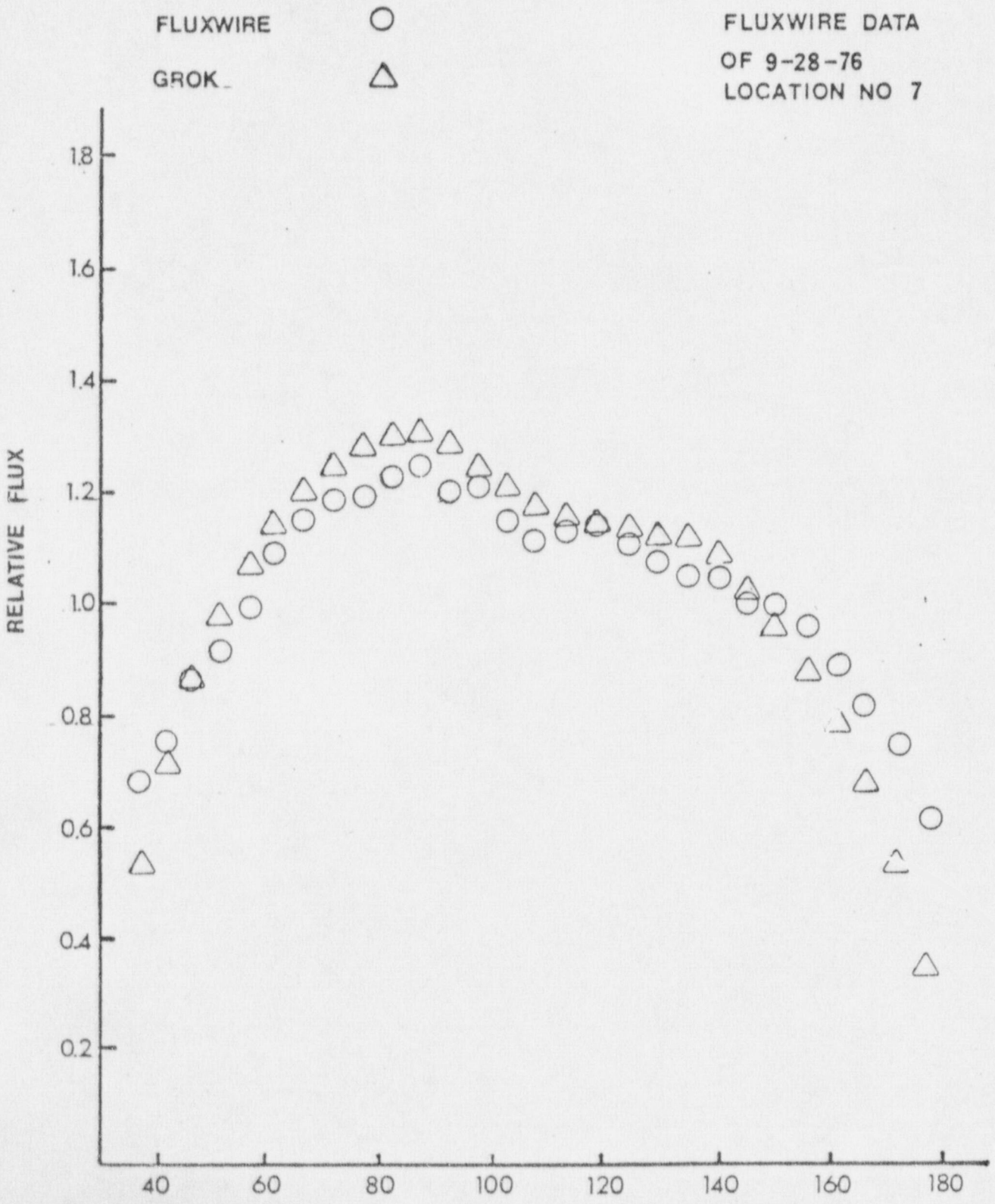


FIGURE 4 - 10

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 14

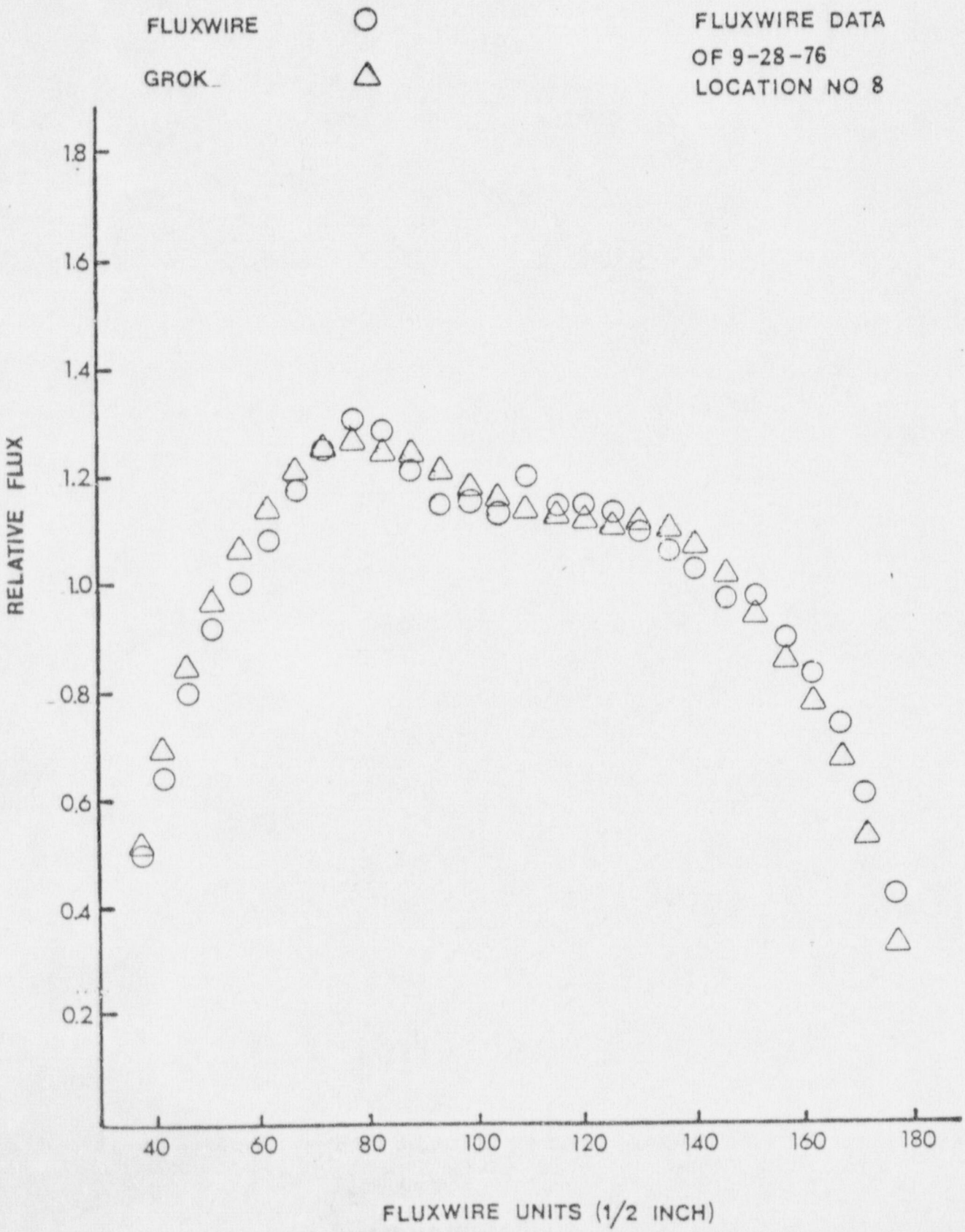


FIGURE 4-11

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
EOC 14

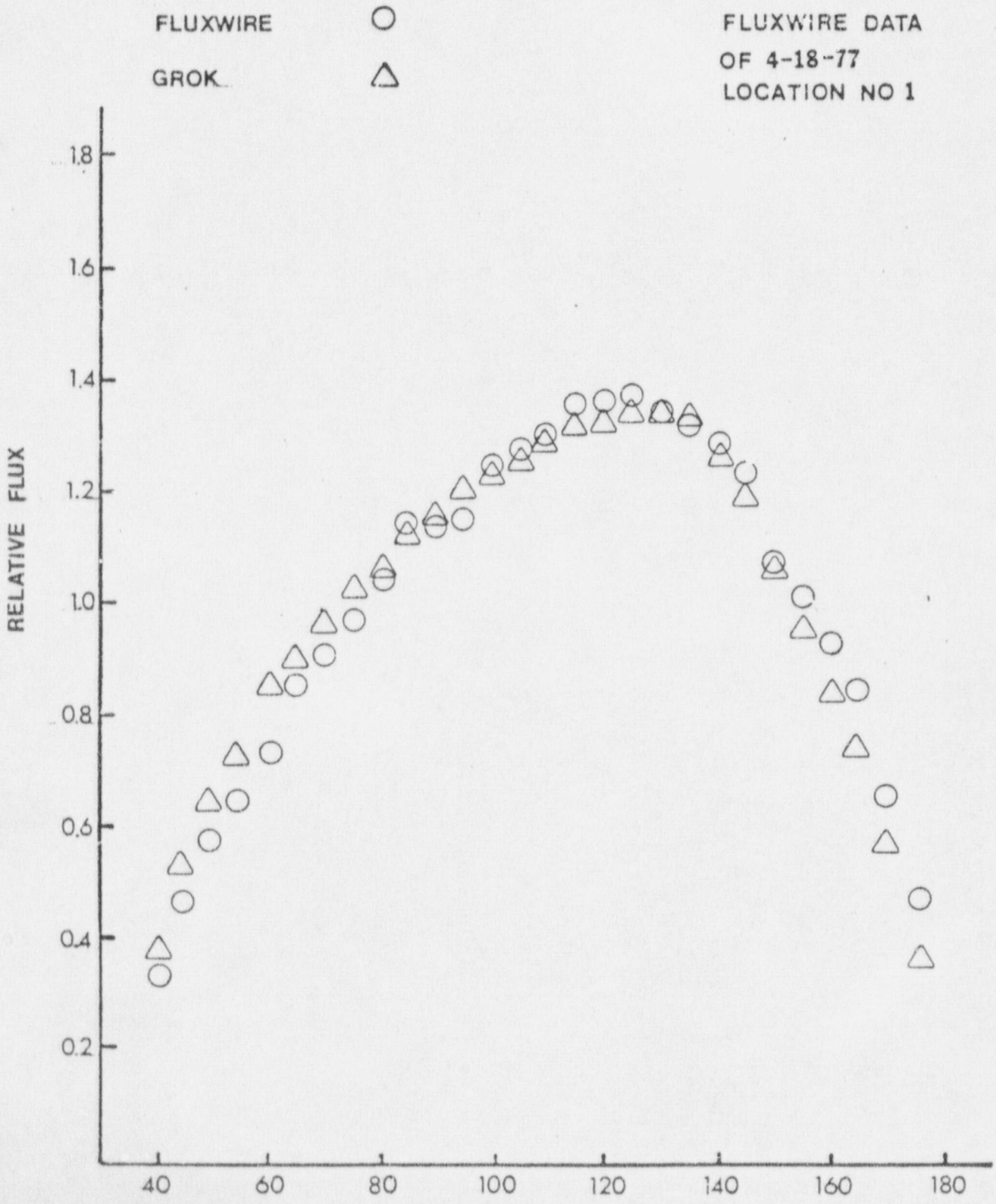


FIGURE 4-12

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

EOC 14

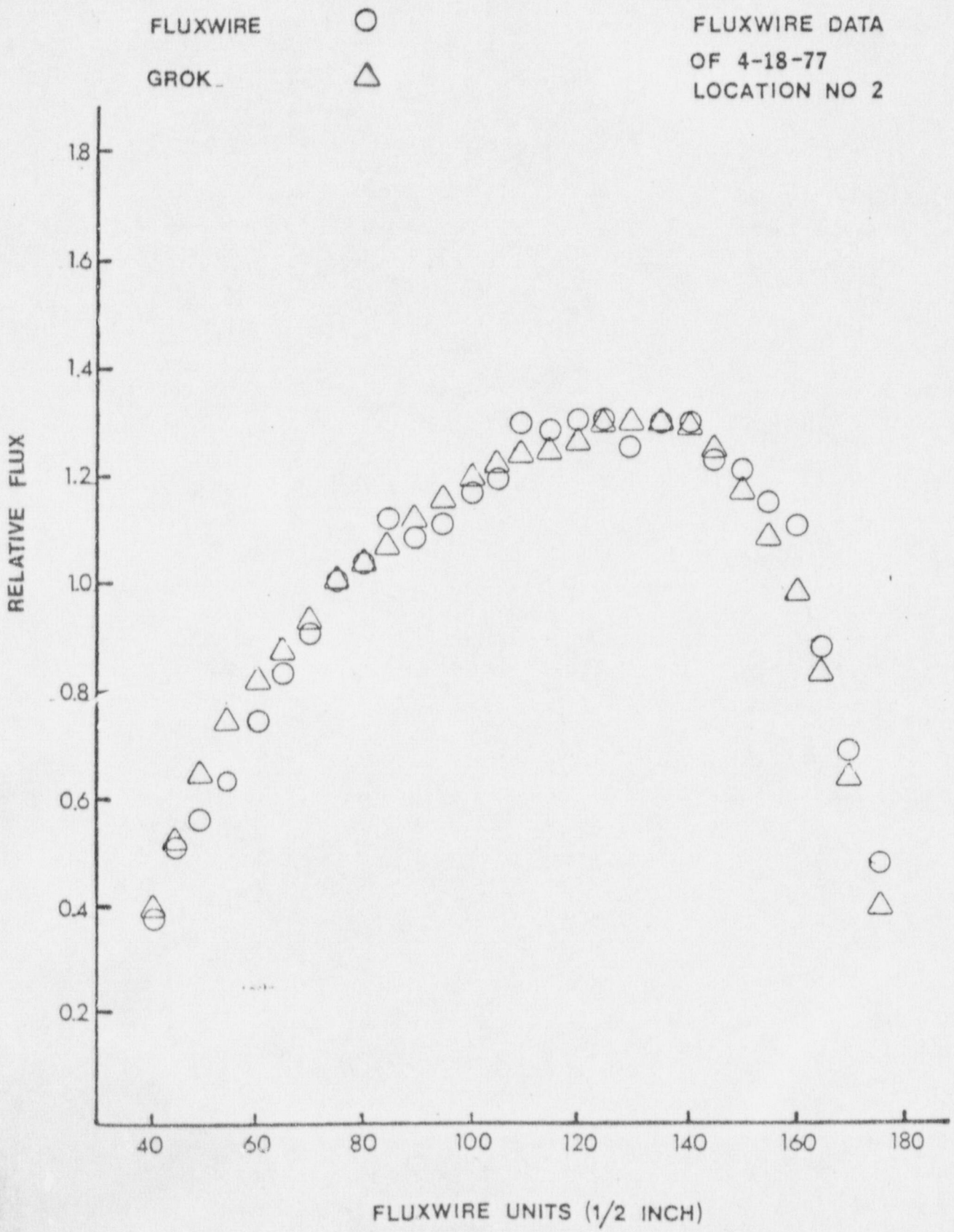


FIGURE 4-13

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
EOC 14

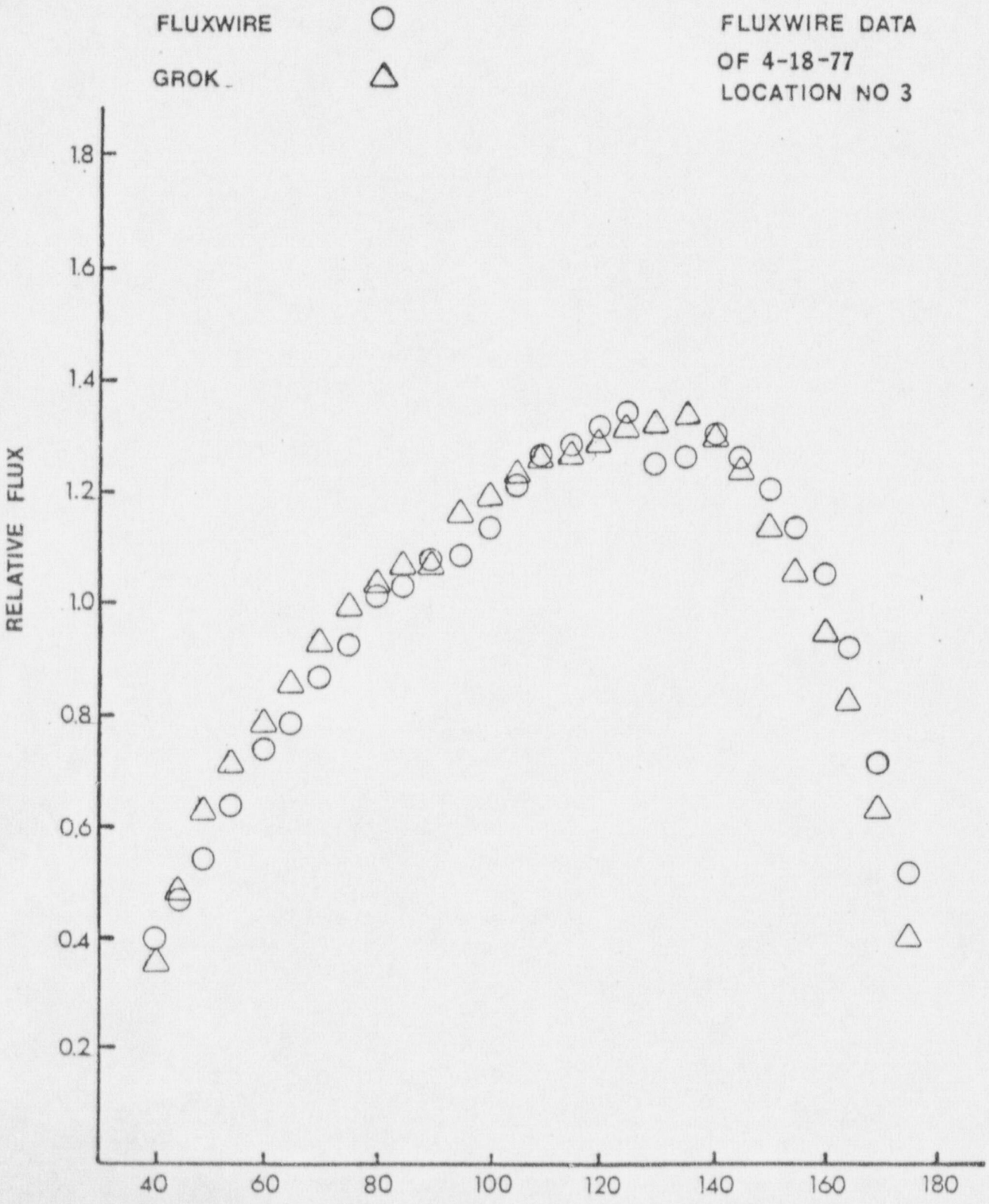


FIGURE 4 - 14

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

EOC 14

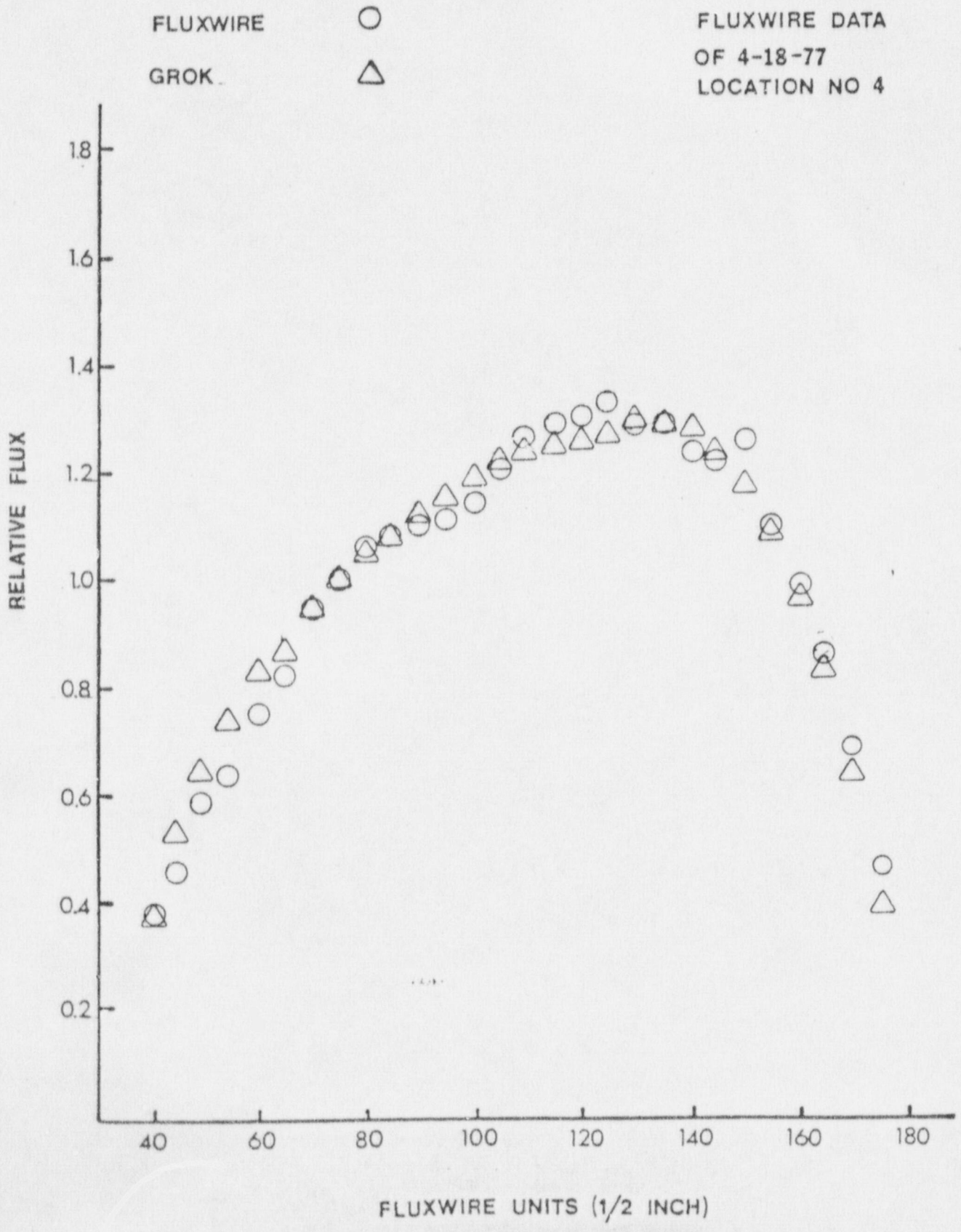


FIGURE 4 - 15

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
EOC 14

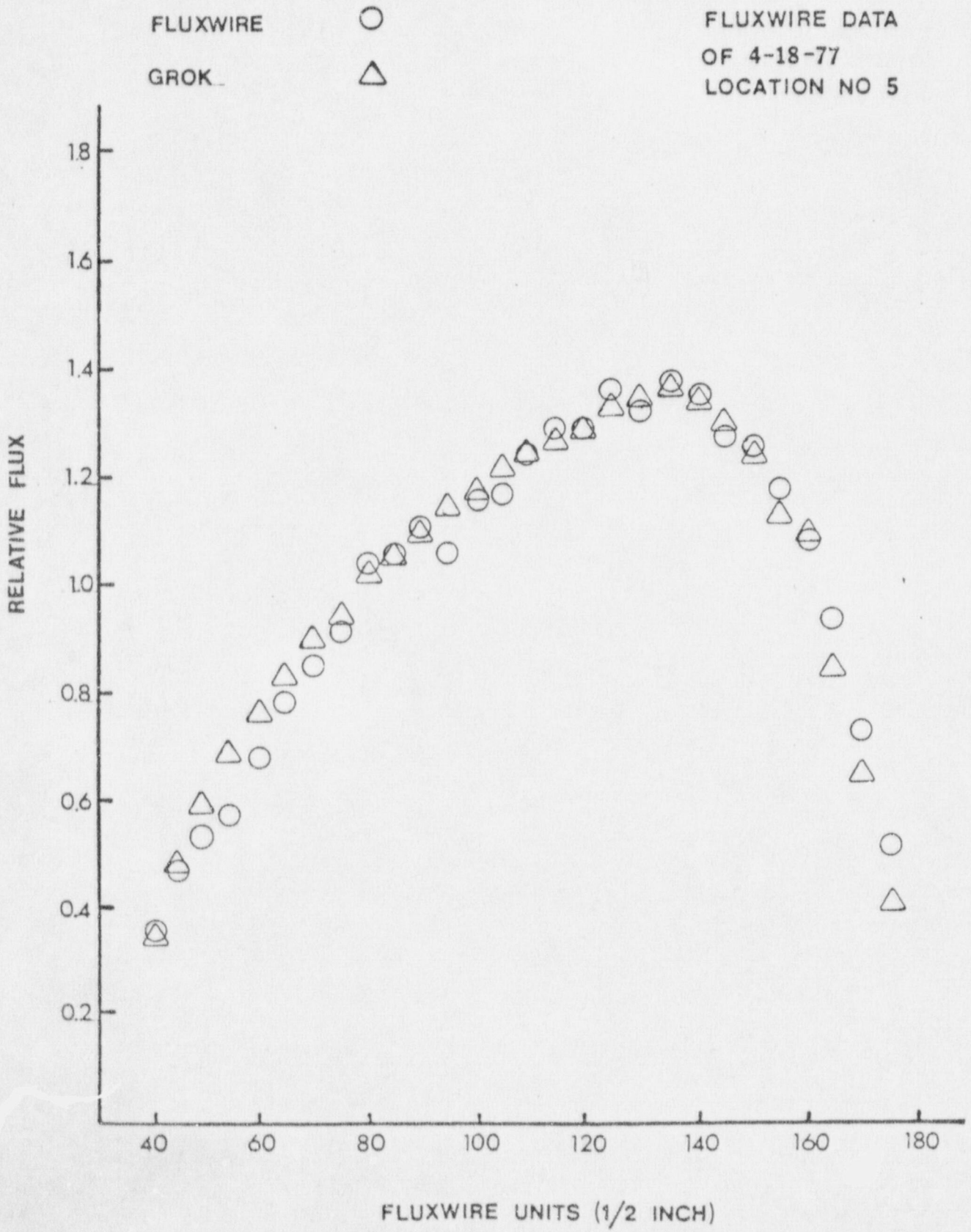


FIGURE 4-16

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

EOC 14

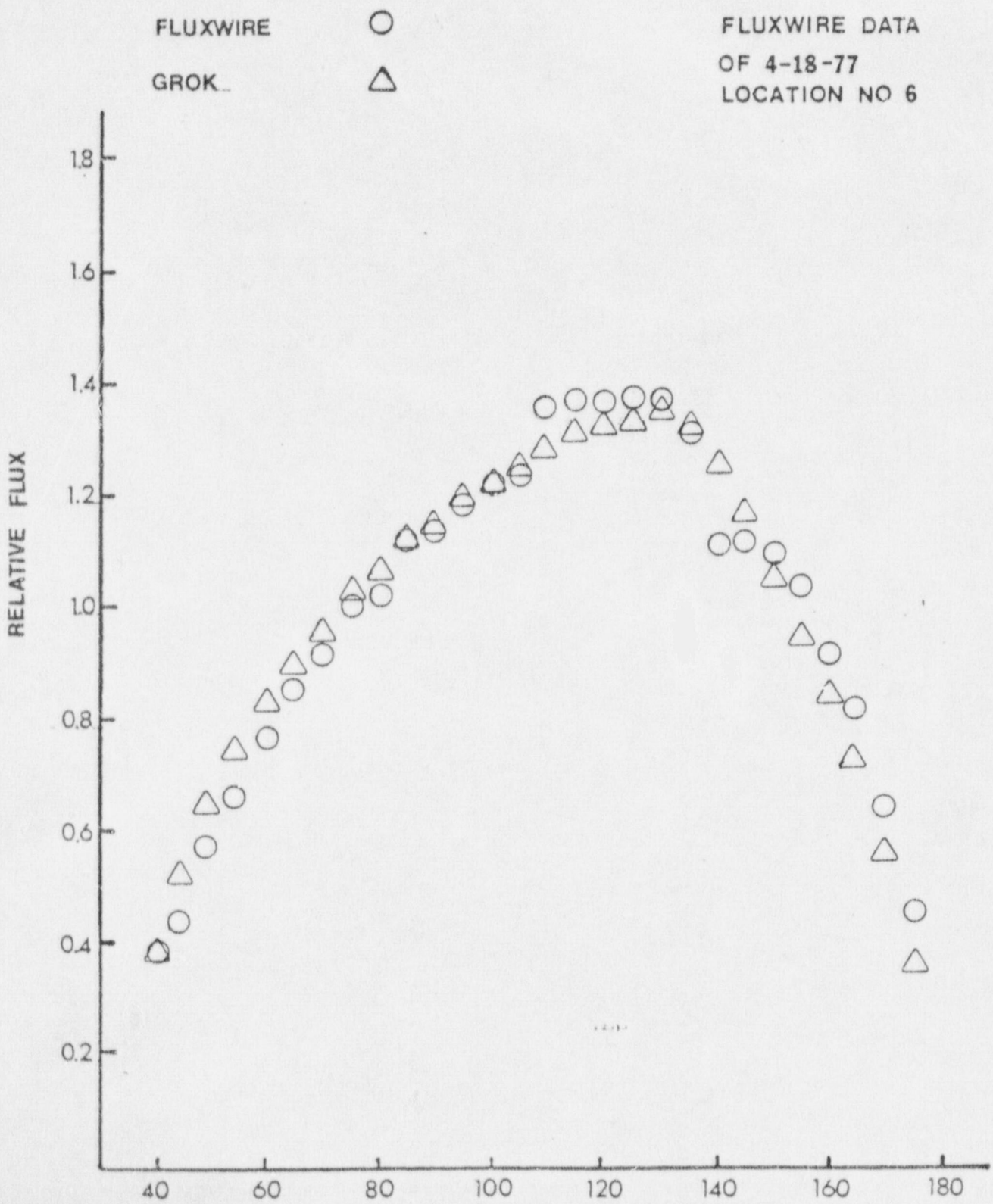


FIGURE 4-17

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
EOC 14

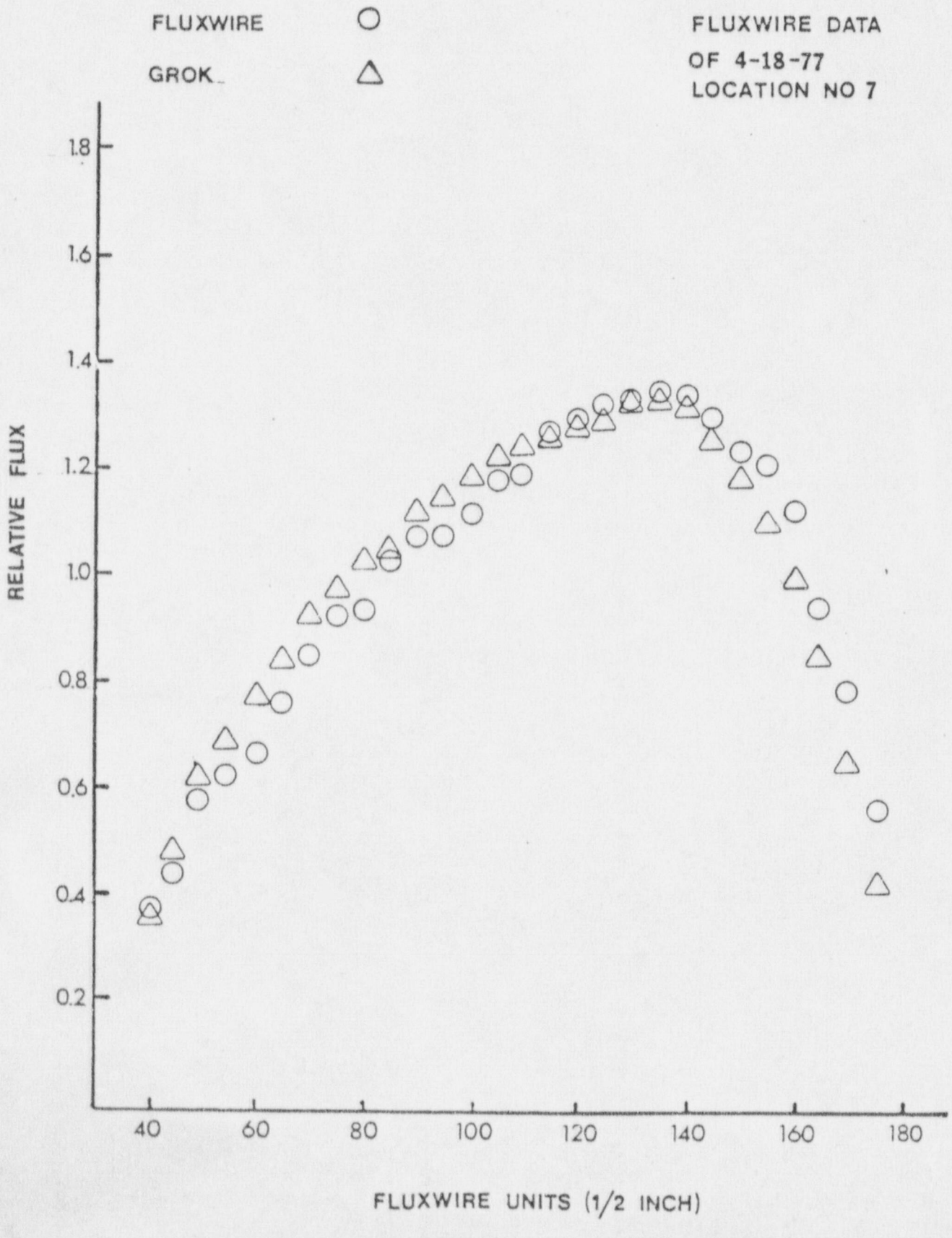


FIGURE 4-18

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
EOC 14

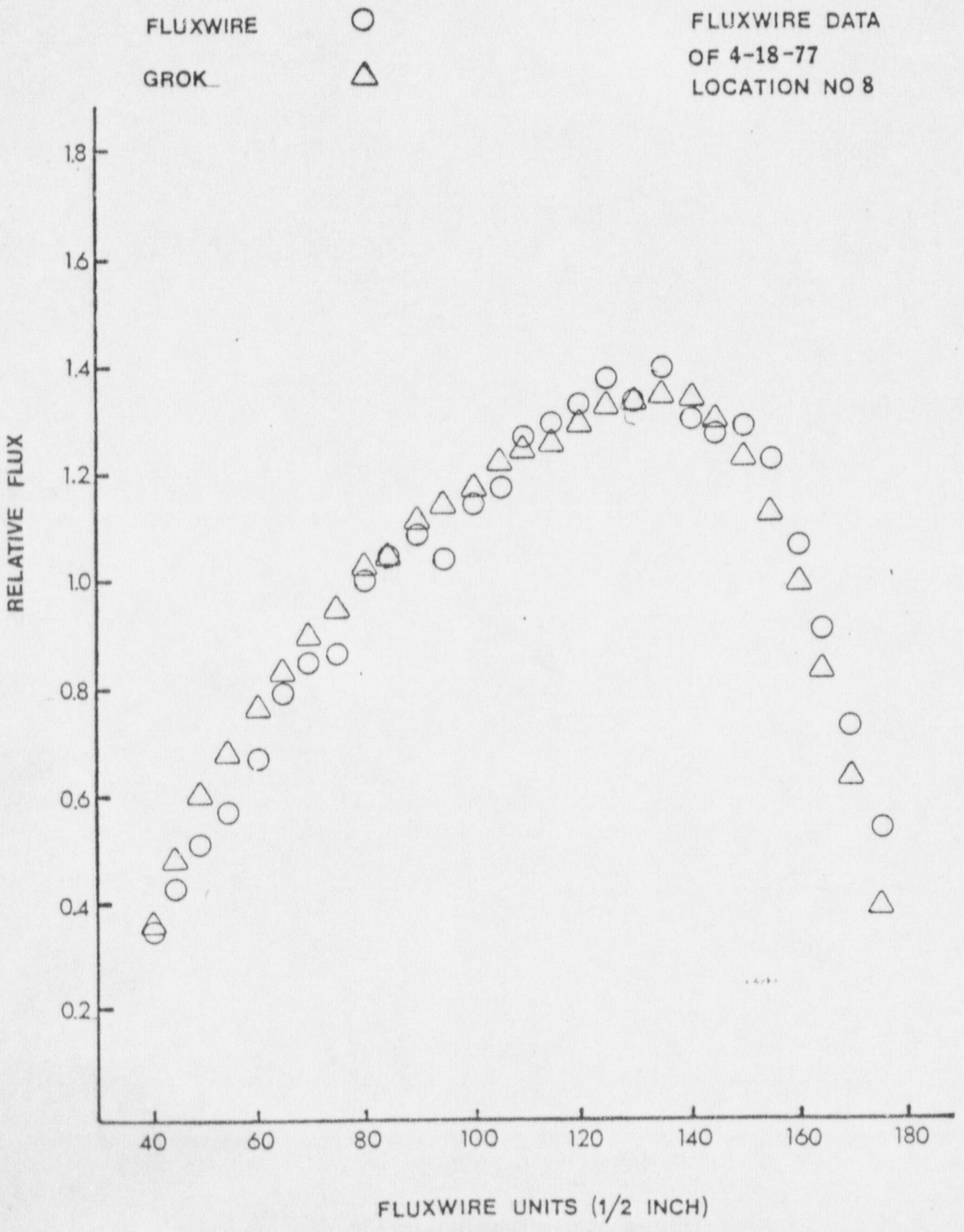


FIGURE 4-19

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 15

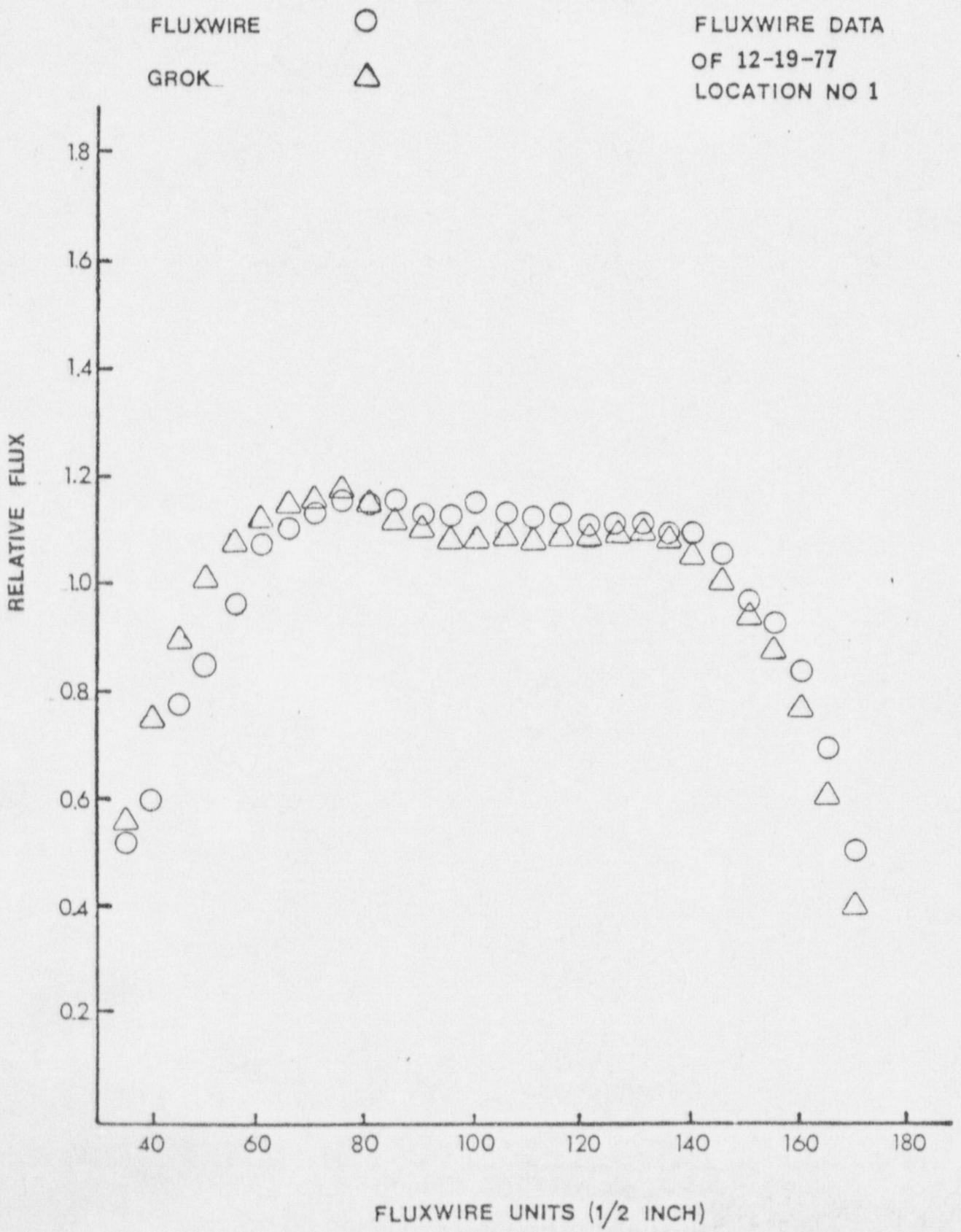


FIGURE 4-20

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 15

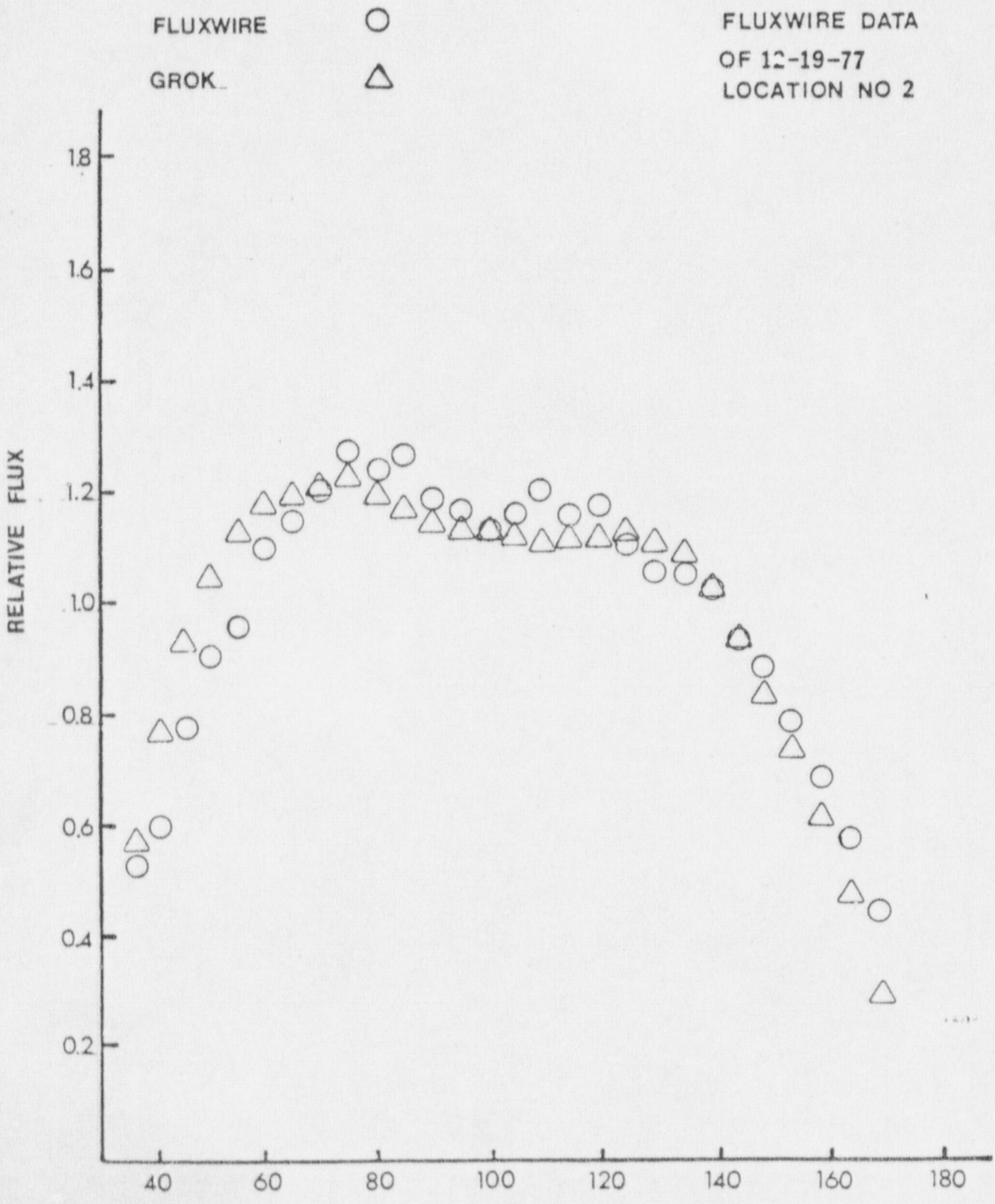


FIGURE 4-21

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 15

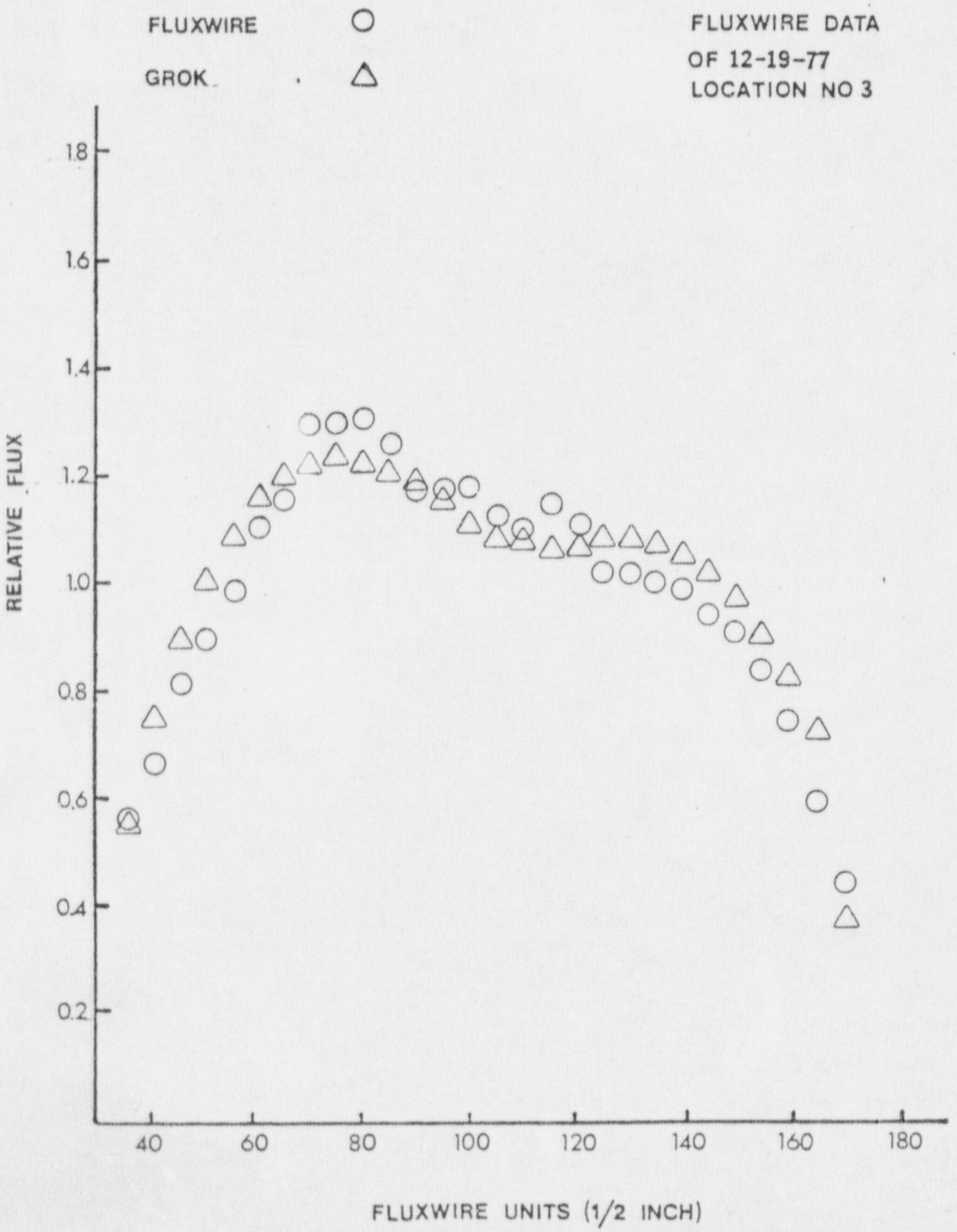


FIGURE 4-22

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 15

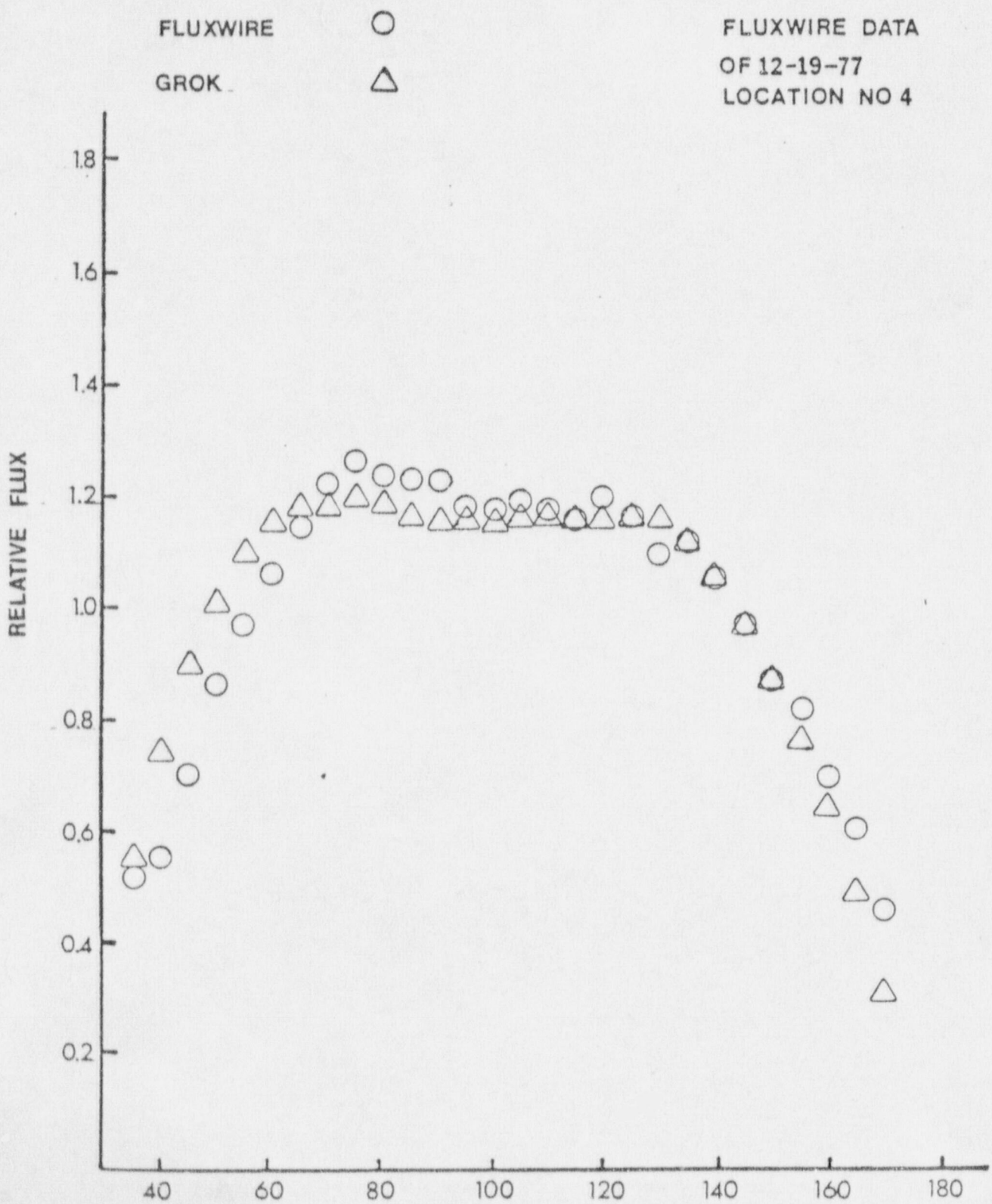


FIGURE 4-23

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 15

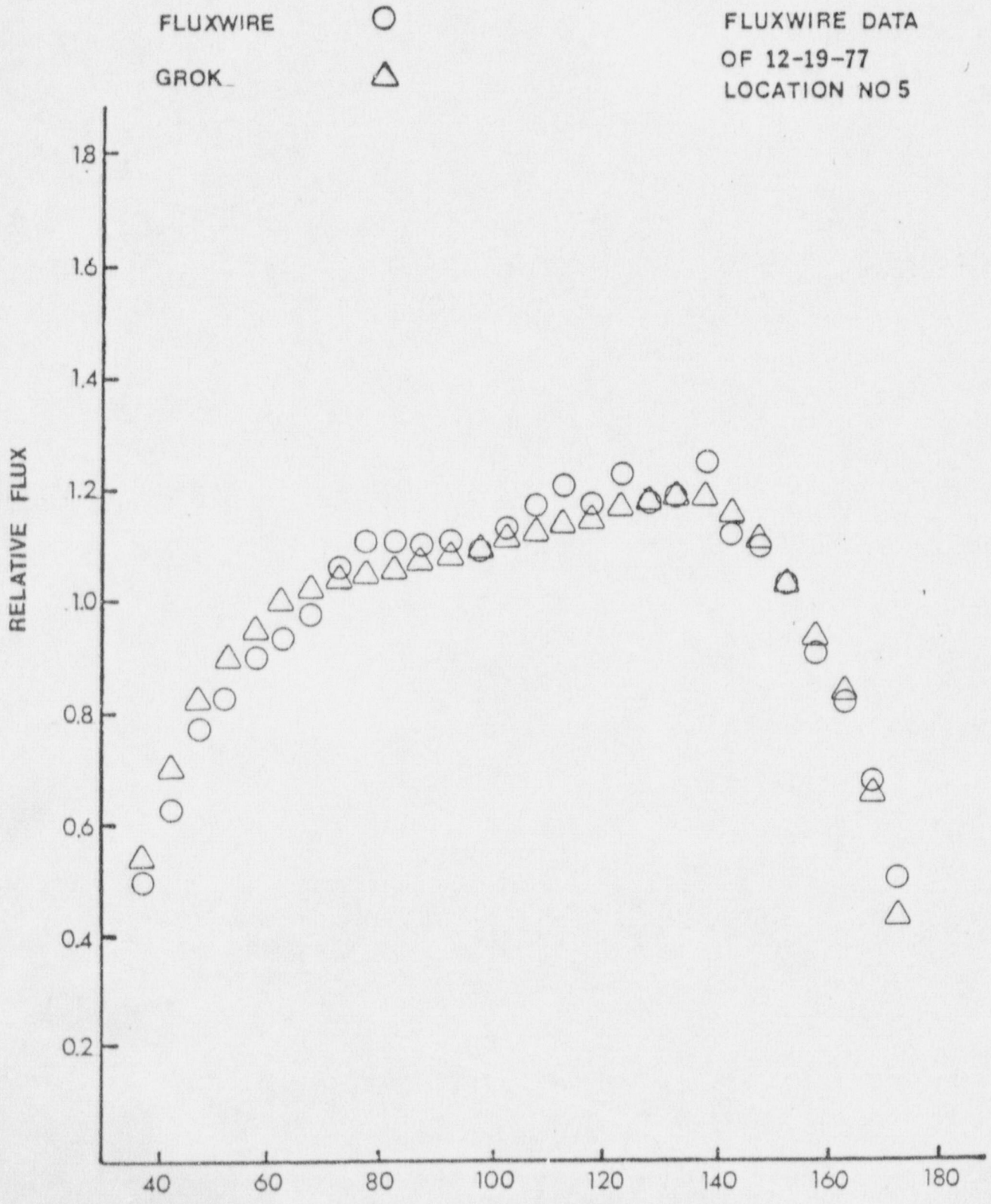


FIGURE 4-24

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 15

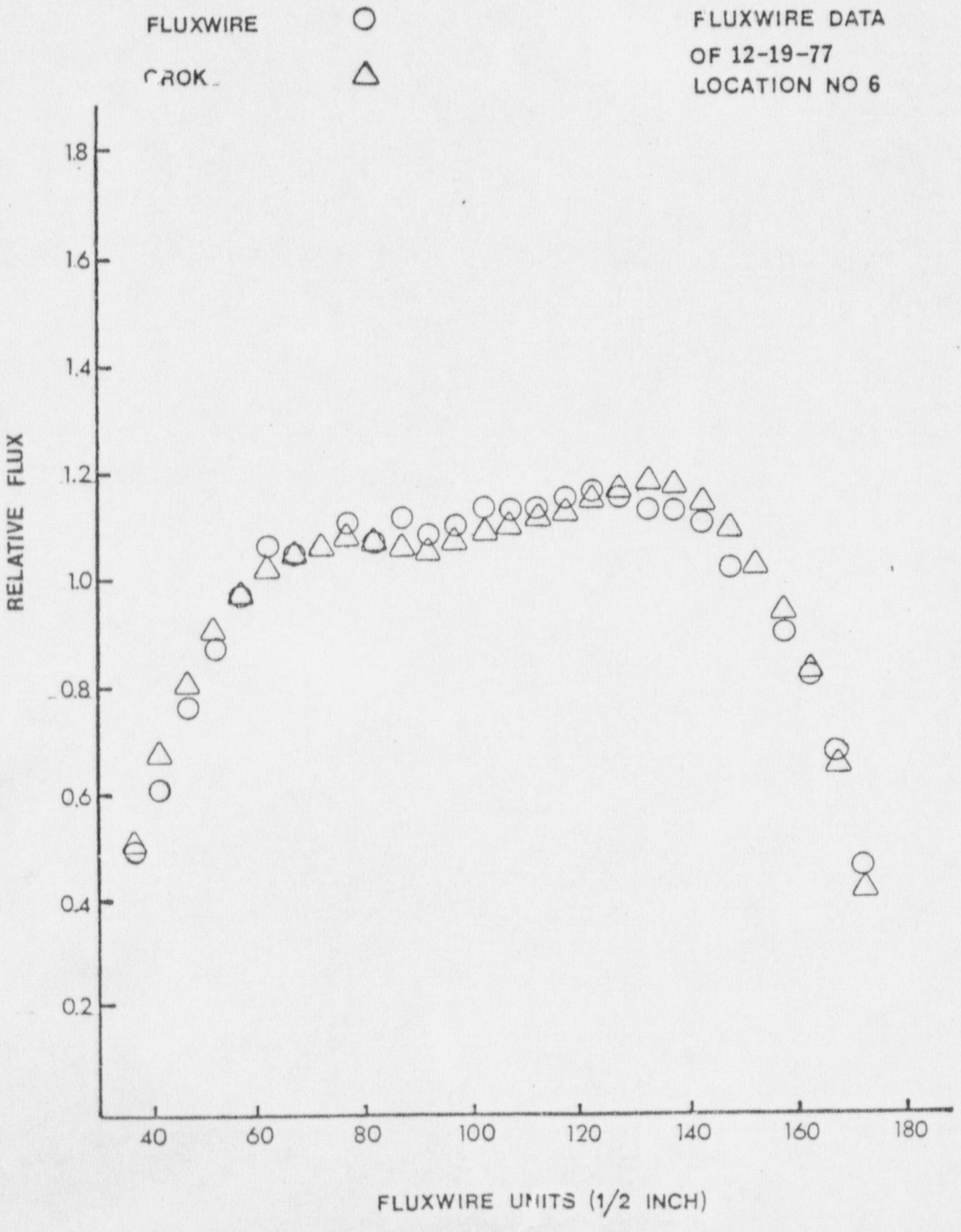


FIGURE 4-25

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

BOC 15

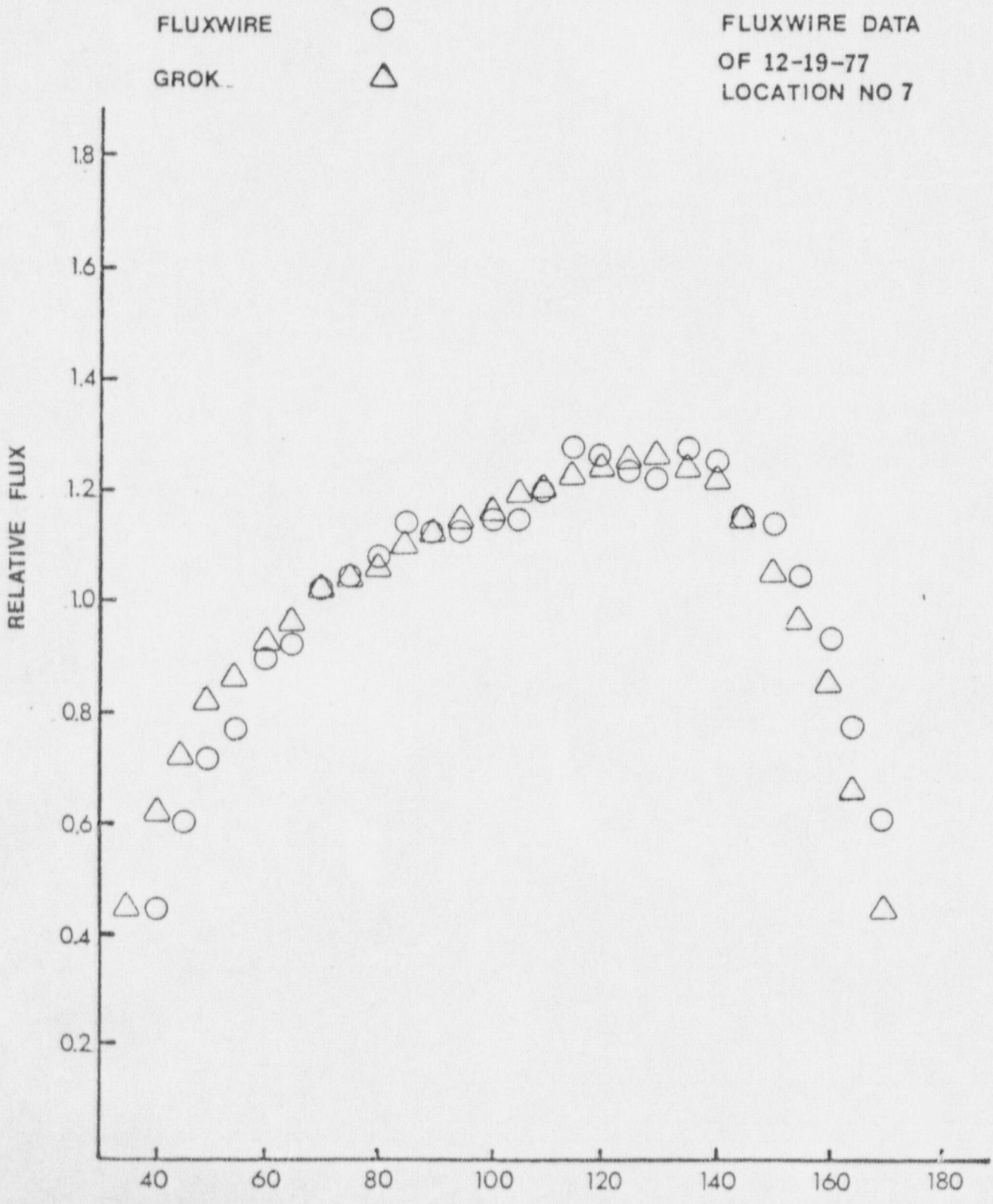


FIGURE 4-26

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
BOC 15

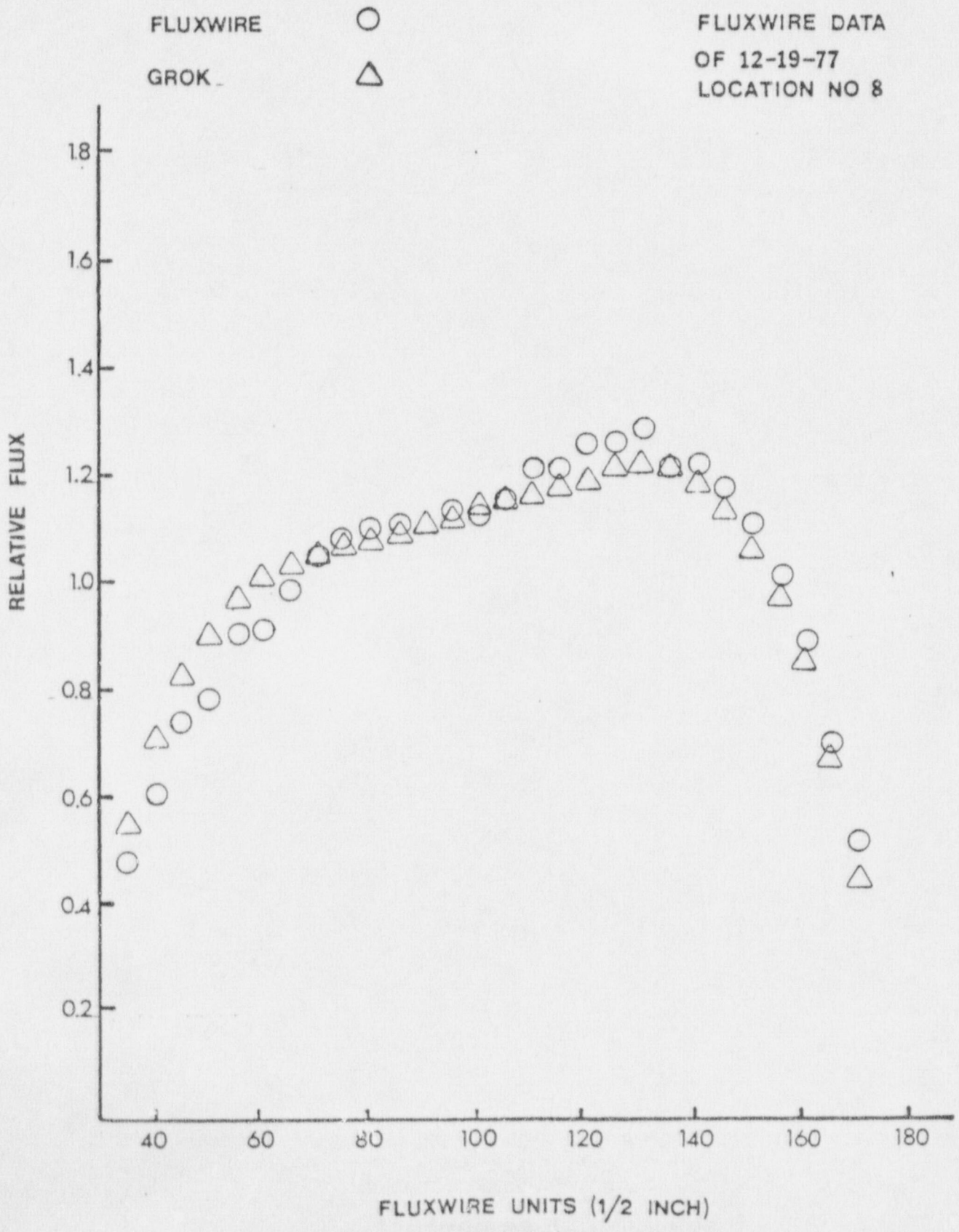


FIGURE 4-27

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
MOC 15

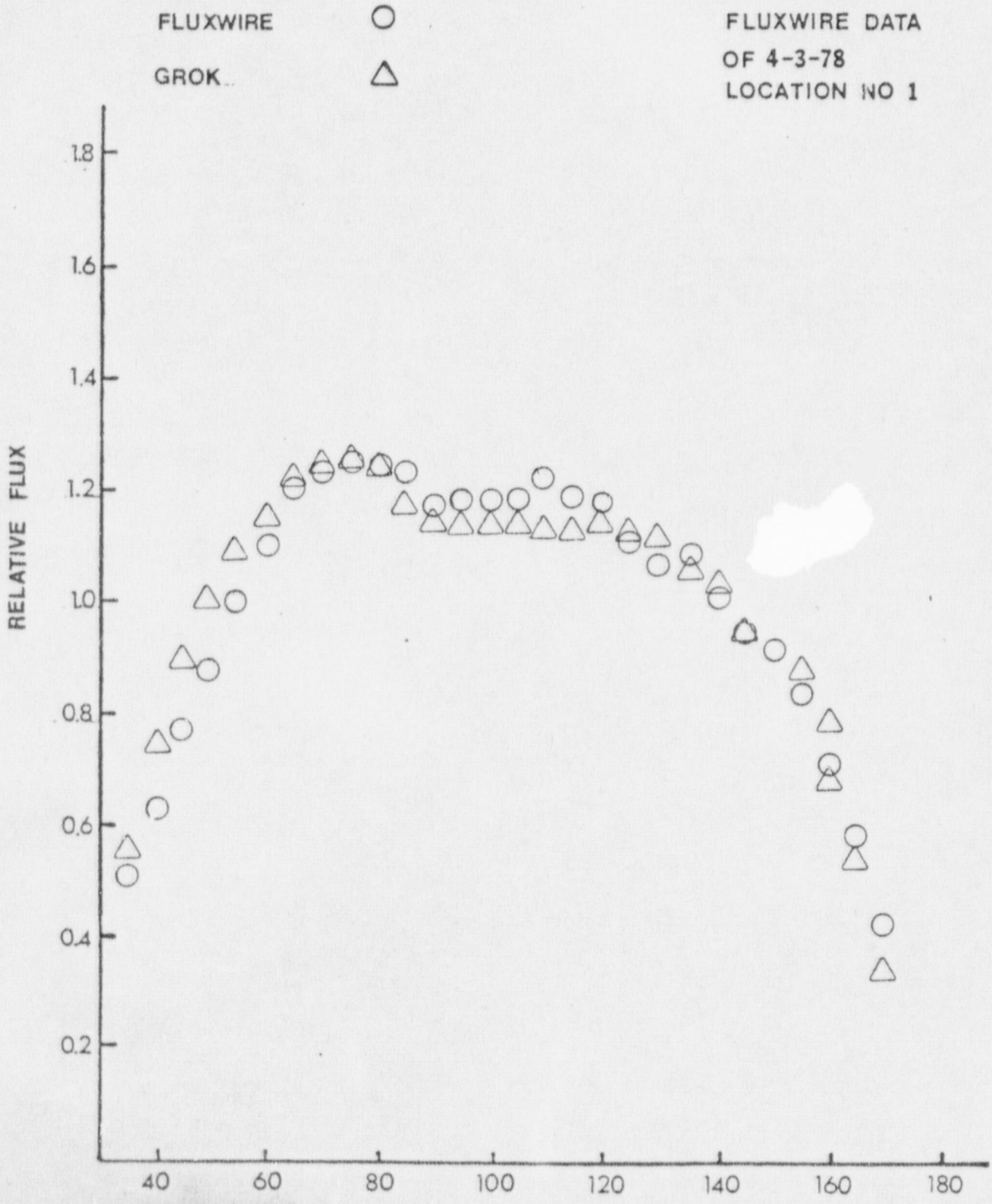


FIGURE 4-28

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
MOC 15

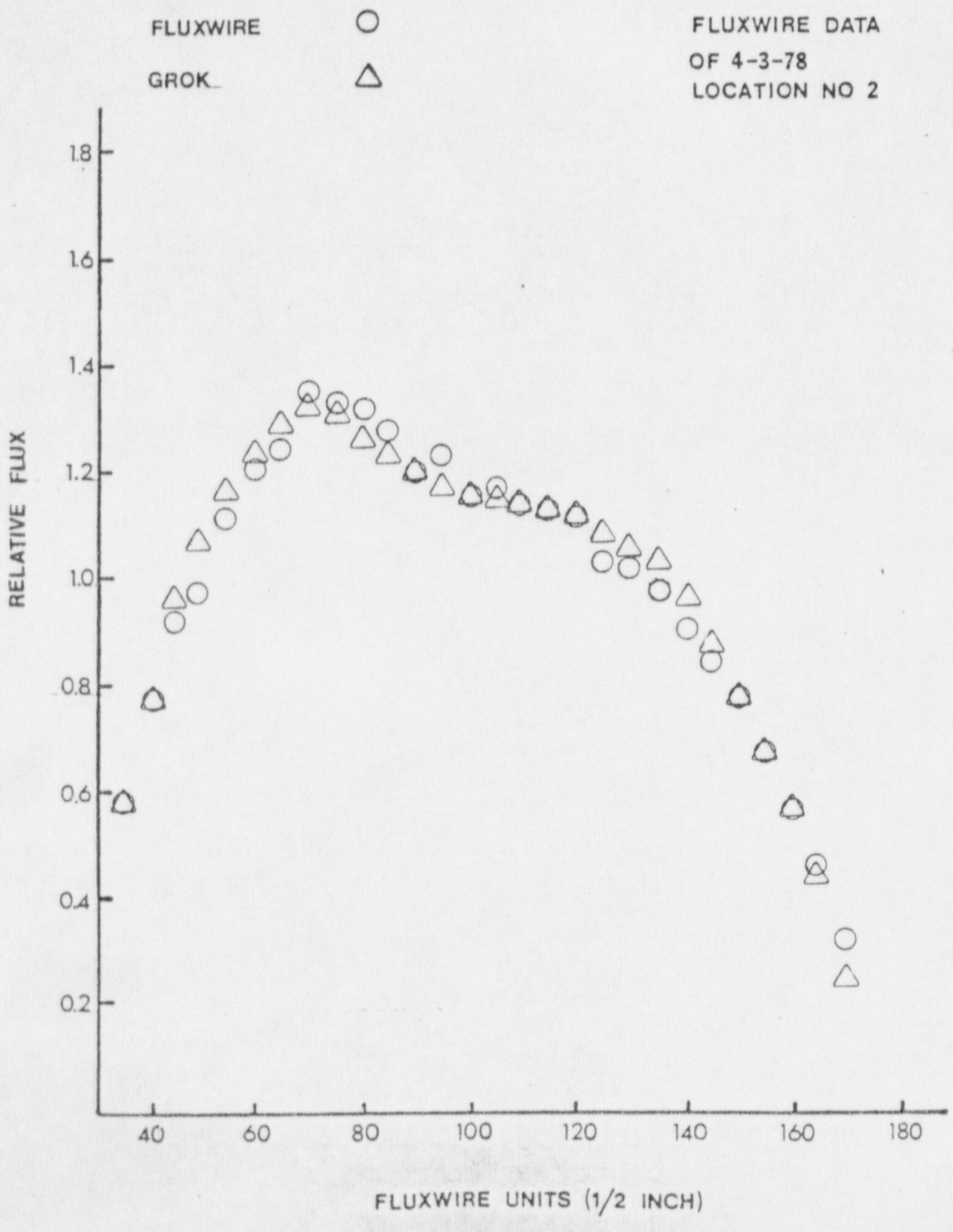


FIGURE 4-29

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
MOC 15

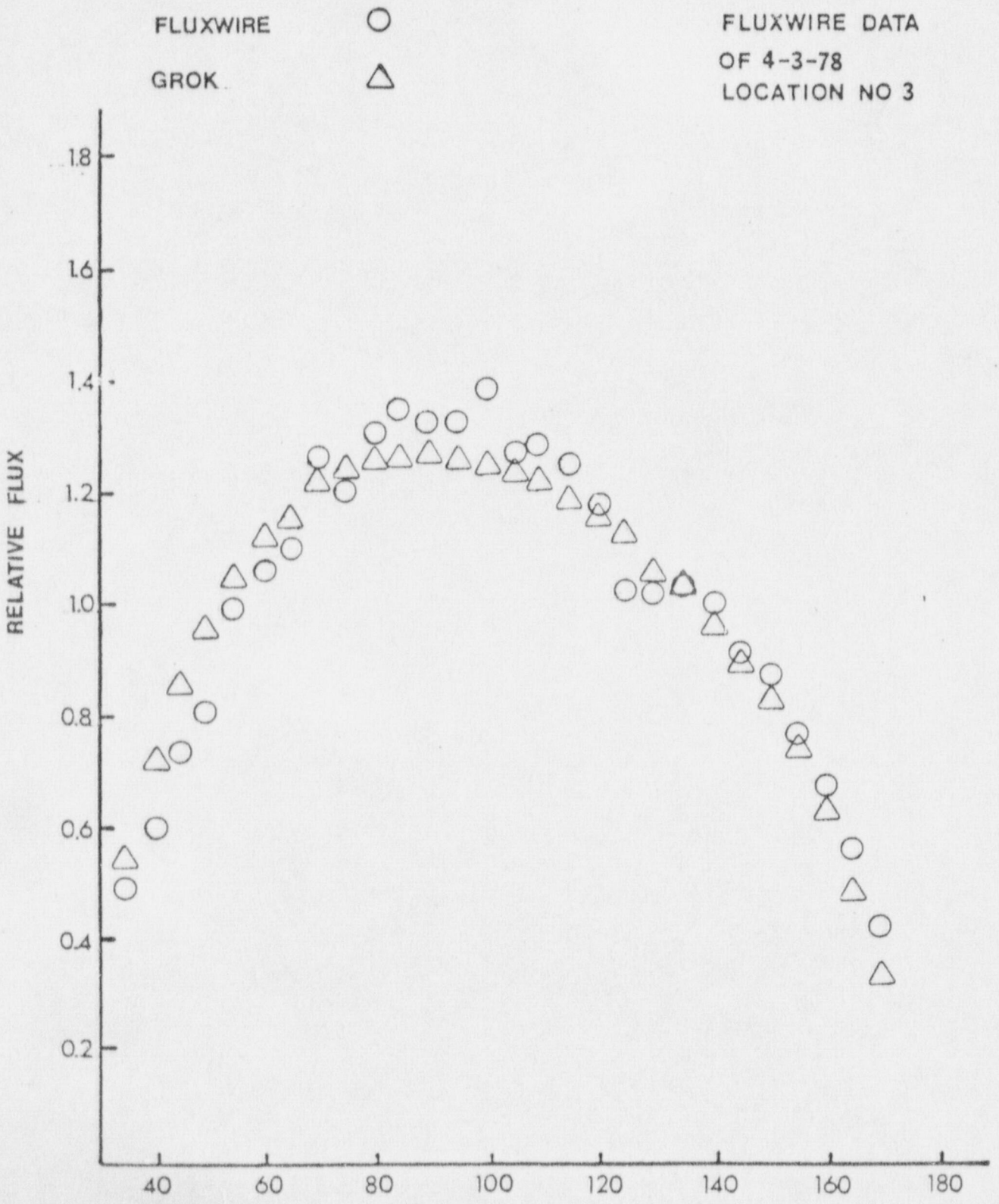


FIGURE 4-30

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
MOC 15

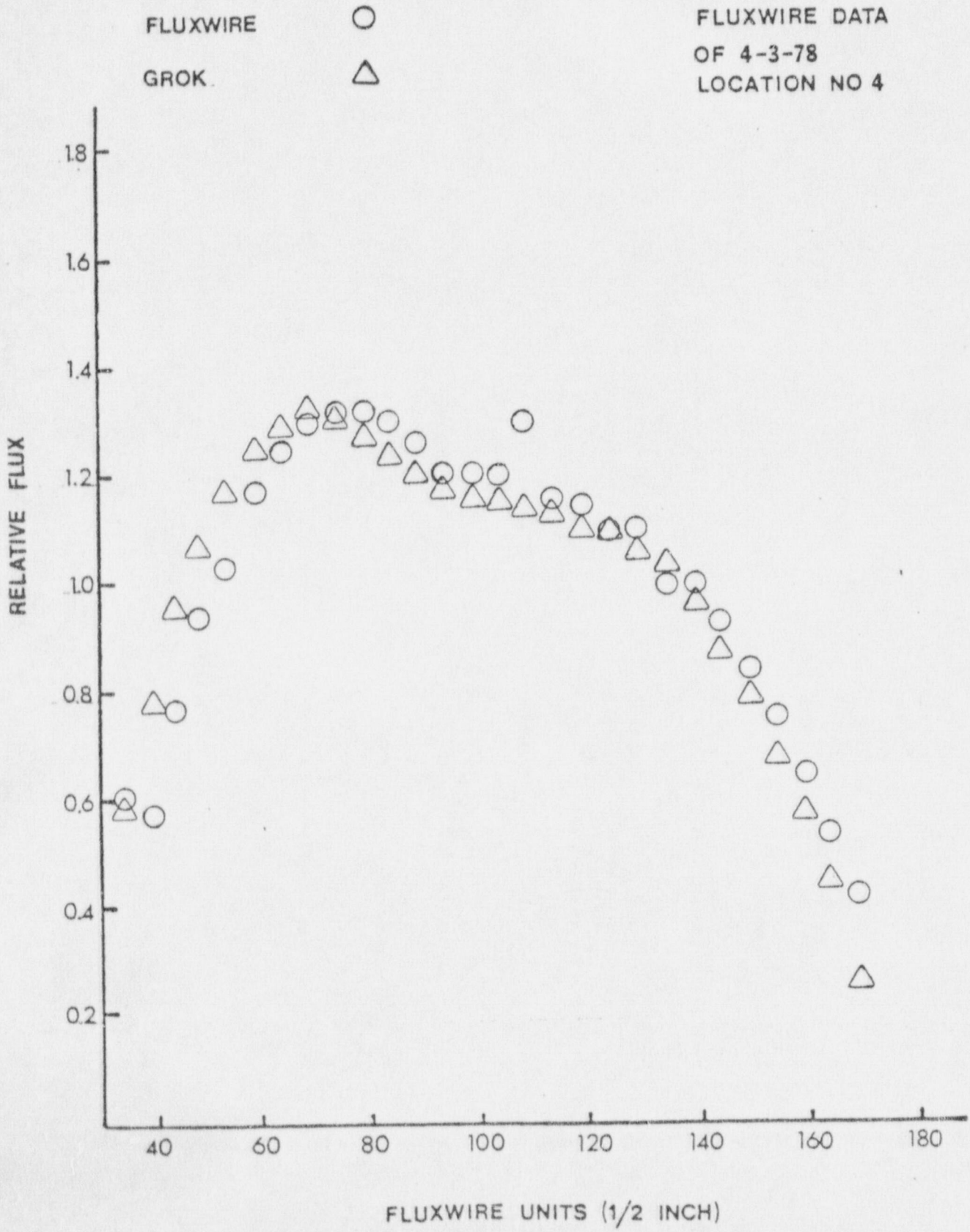


FIGURE 4-31

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
MOC 15

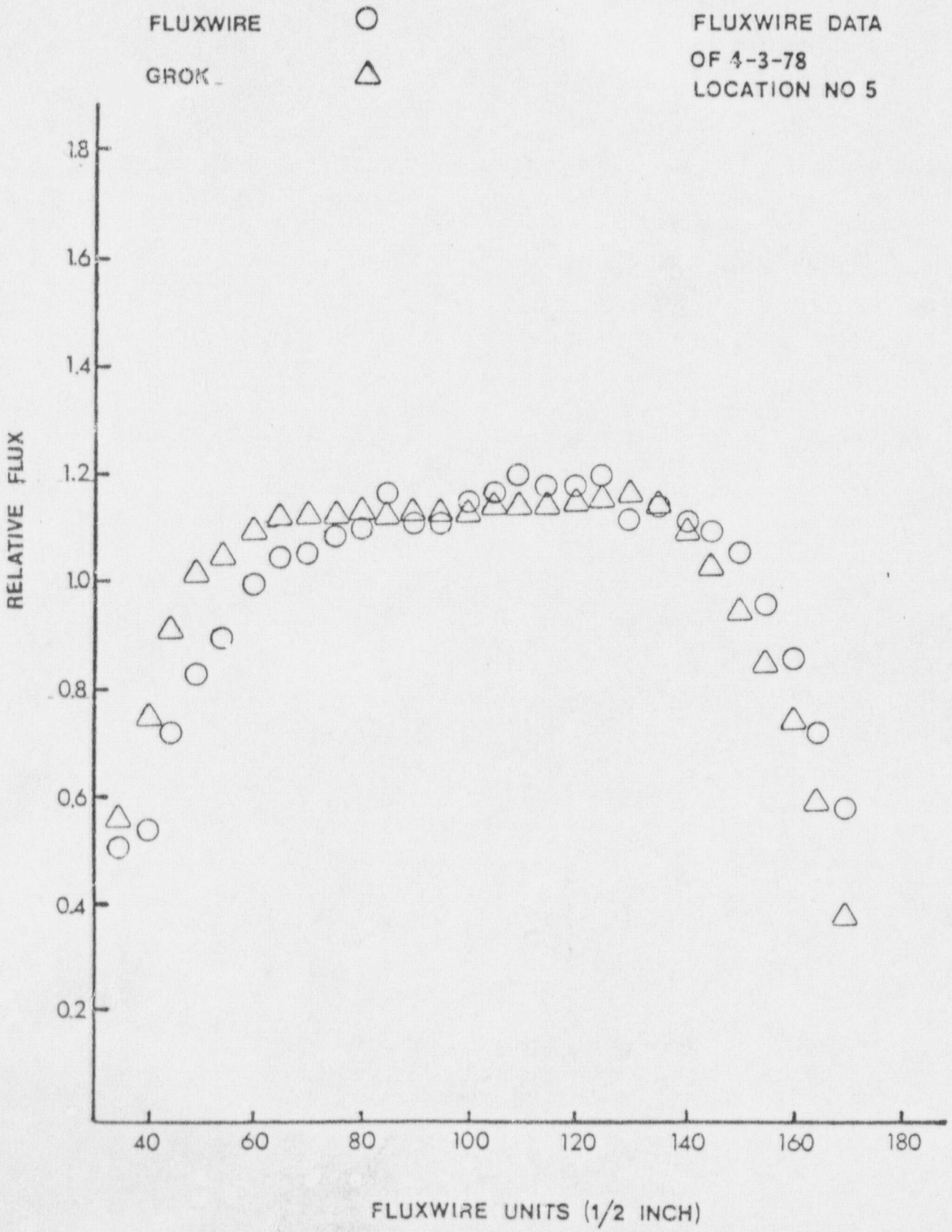


FIGURE 4-32

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

MOC 15

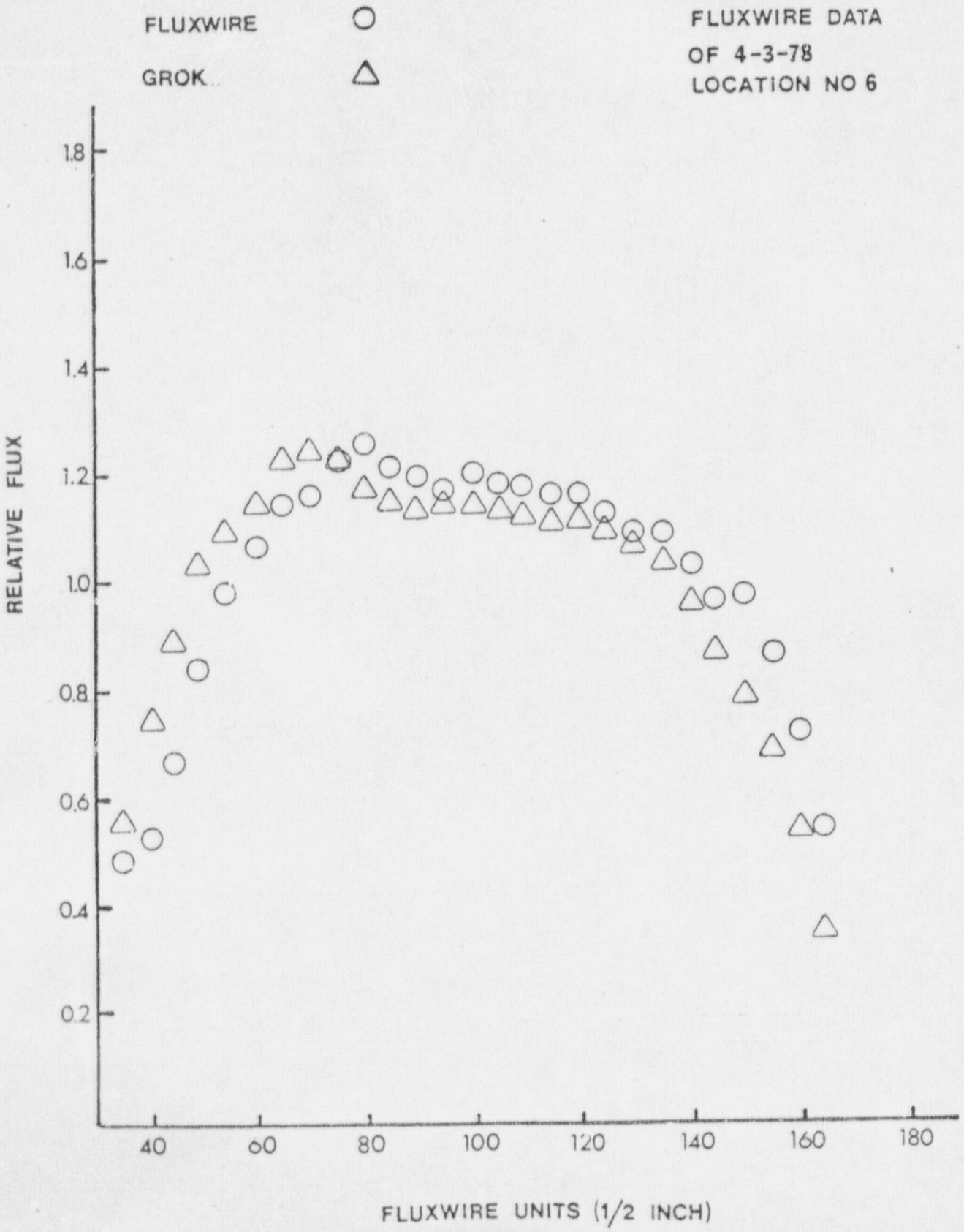


FIGURE 4-33

NORMALIZED AXIAL PROFILE

BIG ROCK POINT
MOC 15

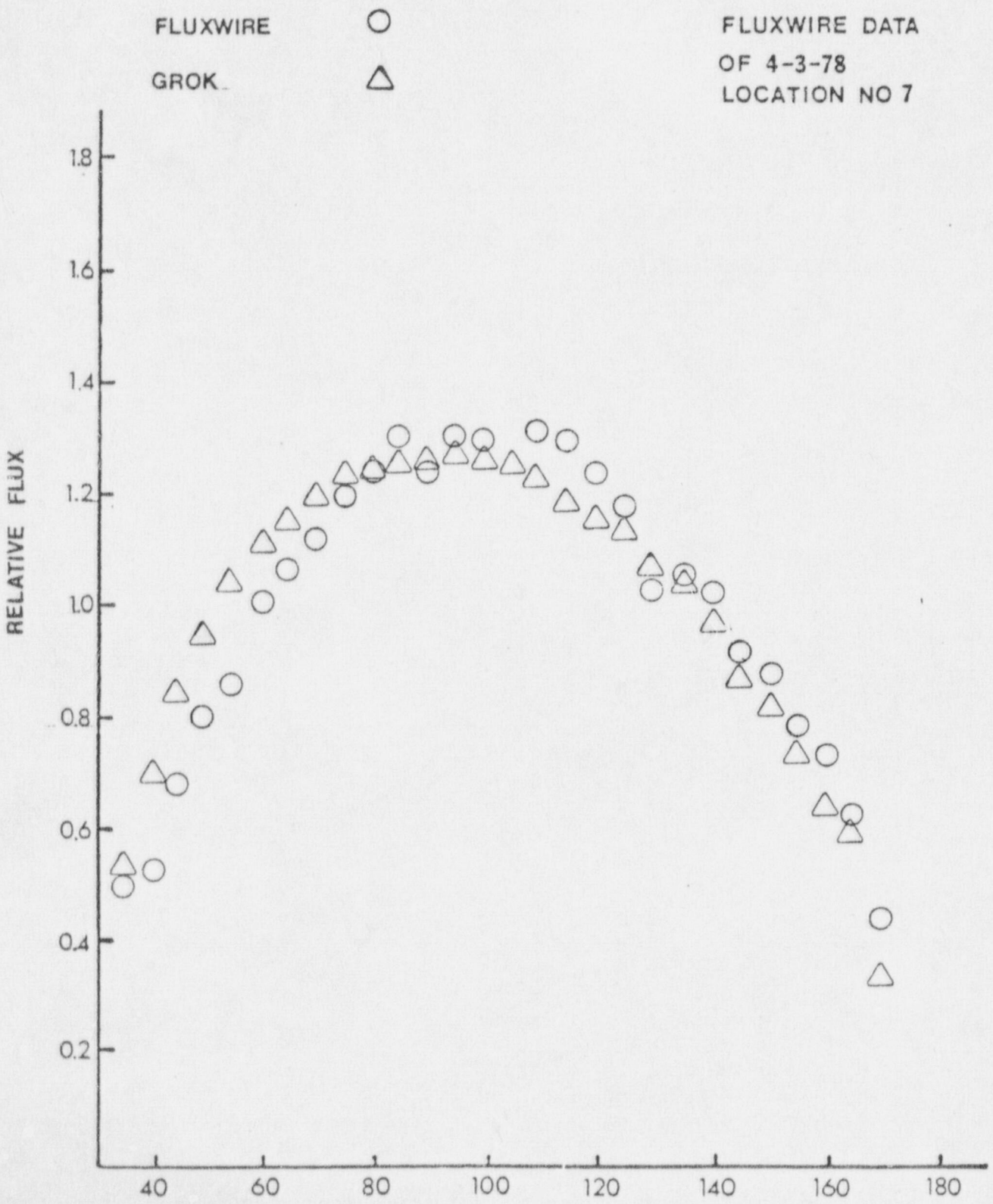


FIGURE 4-34

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

MOC 15

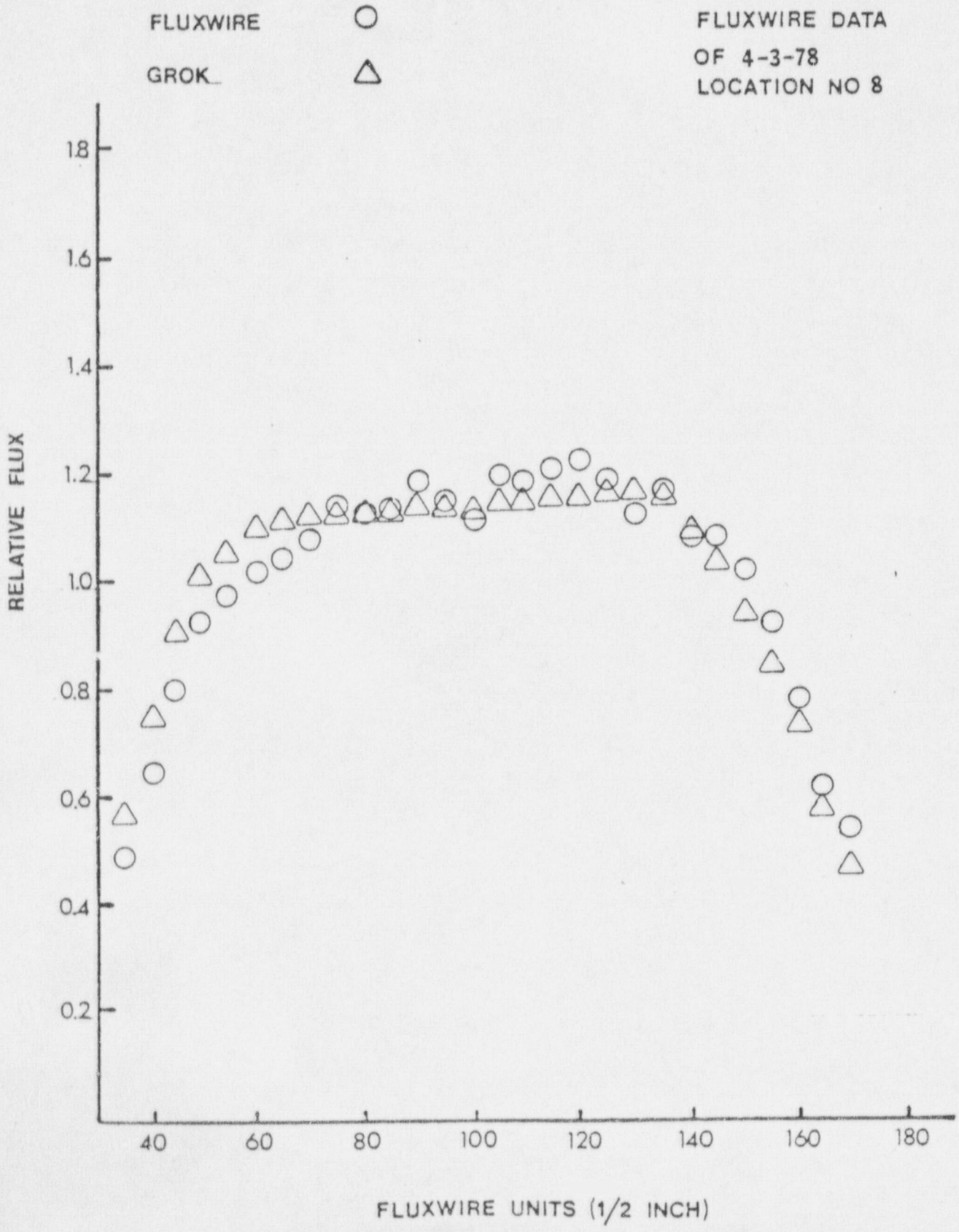
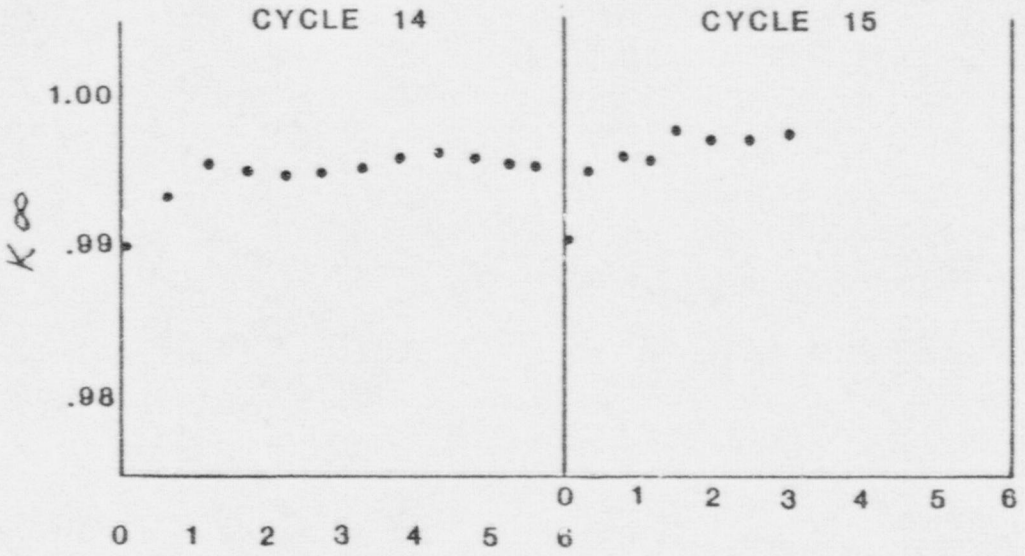


FIGURE 4-35

CALCULATED CORE MULTIPLICATION FACTOR VS BURNUP



## 5.0 SUMMARY AND CONCLUSIONS

The reactor physics methods employed at Consumers Power Company are very similar to methods used elsewhere in the industry. The computer models are or are derived from widely accepted codes which are well tested and documented.

Agreement with measured data has demonstrated the accuracy and applicability of the methodology. Reactivity is consistently predicted at both cold and hot operating conditions, and power distributions agree well with the measurements, indicating that the various neutronic effects are being properly modeled.

## REFERENCES


1. A. Ahlin and M. Edenius, "CASMO - The Fuel Assembly Burnup Program", AE-RF-76-4158, AB ATOMENERGI, Sweden (1976).

2. W.R. Cadwell, "PDQ-7 Reference Manual", WAPD-TM-678 (1967).

3. F.J. Skogen, "XPOSE - The Exxon Nuclear Revised LEOPARD", XN-CC-21, Revision 2, Exxon Nuclear Company (April 1975).

4. D.L. Delp, et al, "FLARE - A Three Dimensional Boiling Water Reactor Simulator", GEAP-4598 (July 1964).

5. B.D. Webb, "Power Peaking at the Tip of a Cruciform Control Rod", Consumers Power Company, (January 6, 1972).

6. "Jersey Nuclear Company Development Fuel Assemblies for Loading in the Consumers Power Company Big Rock Point Reactor Cycle 9", JN-70-1, Jersey Nuclear Company, (November 15, 1970), pp. 100-105, corrected via conversation between SVanVolkinburg and GFPratt, Consumers Power Company, (August 29, 1977).

7. K. Galbraith and J. Jaech, "The XN-2 Critical Power Correlation", XN-75-34, Exxon Nuclear Company, (August 1, 1975).

8. A.A. Armand, "The Resistance During the Movement of a Two-Phase System in Horizontal Pipes", Translated by V. Beak, AERE Trans 828. Izvestiya Vsesojuznogo Teplotekhnicheskogo Instituta (1), pp 16-23, (1946).

9. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K", NEDO-20566, (January, 1976).

10. R.F. Barry, "LEOPARD - A Spectrum Dependent, Non-Spatial Depletion Code for the IBM-7094", WCAP-3269-26, Westinghouse Electric Corporation, (1963).

11. J.E. Suich & H.C. Honeck, "The HAMMER System", DP-1064, January 1967.

12. W.J. Eich et al, "Advanced Recycle Methodology Program System Documentation", EPRI-RP118-1, (October, 1976).

13. K.P. Galbraith, "GAPEX, A Computer Program for Predicting Pellet to Cladding Heat Transfer Coefficients", XN-73-25, Exxon Nuclear Company (August 13, 1973).