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SUBJECT: Calculation of Radial Neutron-Flux Distribution in EGCR Lattice Cell

TO: Listed Distribution

FROM: T. K. DeBoer

Abstract

The neutron-flux distributions in an EGCR cell containing seven rod clusters of 2.0 and 2.6% enriched uranium oxide have been obtained by using a one-velocity, one-dimensional P-3 solution to the neutron-transport equation and adjusting fluxes in the fuel cluster in a manner which is consistent with previous comparisons of experimental and calculated distributions. Flux traverses in the outer rod perpendicular to a diameter of the cluster are also presented.

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DISCLAIMER

CALCULATION OF RADIAL NEUTRON-FLUX DISTRIBUTION IN EGCR LATTICE CELL

Introduction

The power-density distribution within the EGCR fuel cluster will have an important influence on the heat-transfer and thermal-stability characteristics[1] of the fuel element, and the method described herein to predict this distribution is a model which appears to yield results that are quite reliable.

In an effort to determine the validity of the various models used to predict neutron-flux distributions in seven rod fuel clusters, a comparison was made between theoretical distributions and detailed experimental results obtained at Atomics International.[2,3] As a result of this comparison, it was found that the fluxes obtained from an annular one-velocity, one-dimensional P-3 solution to the neutron-transport equation can be used to construct the power distribution in fuel rod clusters.

The model used for the P-3 calculation represented the outer six rods as a concentric annulus having a centerline radius equal to the centerline of the outer rods, a thickness equal to one fuel rod (unclad) diameter, and an appropriately reduced uranium density. The stainless steel cladding

[1] GCR Semiann. Prog. Rep. June 30, 1959, ORNL-2767, p 15.

[2] W. W. Brown et al., Exponential Experiments with Graphite Lattices Containing Multirod Slightly Enriched Uranium Fuel Clusters, NAA-SR-3096 (Jan. 15, 1959).

[3] R. A. Laubenstein et al., Sodium Graphite Reactor Quar. Prog. Rep. Jan.-Mar. 1955, NAA-SR-1347, pp 18-22 (Oct. 1, 1955) (classified).

for the outer rods was represented by annuli adjacent to the outer fuel annulus and having a thickness equal to that of the actual clad. Effective neutron temperatures used for the calculations were computed from the formula of Coveyou, Bate, and Osborne[4] with flux weighted macroscopic cross sections at 2200 m/s, and were found to be 945° and 985°K respectively for the 2.0 and 2.6% cases. These temperatures were employed in determining the Maxwell-Boltzmann-averaged macroscopic cross sections for the P-3 calculations.

The flux distribution from the P-3 annular model gives a fairly reliable estimate of the flux difference from the outer to inner surfaces of the outer rods, but does not provide an accurate estimate of the distribution within the rod. In particular, the annular model does not show the flattening or slight increase of the flux between the outer rod axis and the inner surface of the central rod.

From a comparison of calculated and experimental flux distributions in the outer rods, it was observed that the calculated values overestimated the flux from the rod axis along a cluster diameter to the outer edge of the rod. To obtain a more reliable estimate of the flux distribution in this portion of the rod, a radial plot (see Fig. 1) of $\phi_{exper.}/\phi_{calc.}$ was made as a function of the macroscopic absorption cross section in the outer fuel annulus. Values of $\phi_{exper.}/\phi_{calc.}$ for the EGCR cases (Σ_2 = 0.072 and 0.088) were obtained by extrapolation, and were used

[4] R. R. Coveyou, R. R. Bate and R. K. Osborne, <u>Effect of Moderator Temperature Upon Neutron Flux in Infinite, Capturing Medium</u>, ORNL-1958 (Sept. 29, 1955).

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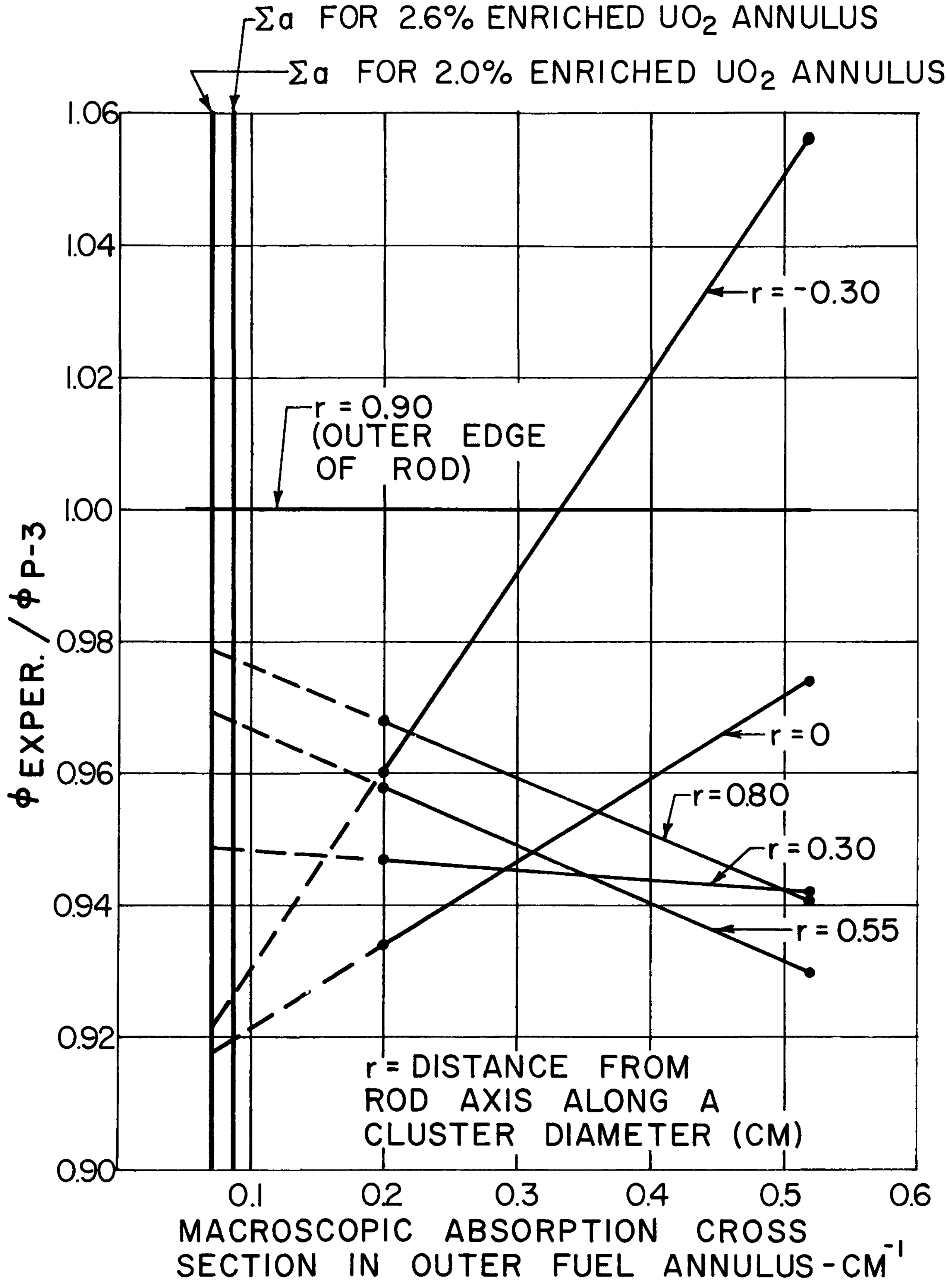

FIG. 1 - ϕ EXPERIMENTAL / ϕ P-3 IN OUTER RODS OF SEVEN ROD CLUSTERS

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to reconstruct the flux distribution. Experimental flux distributions for 2.78% enriched and natural uranium metal rods revealed an upswing in the flux between the rod axis and the inner surface of the rod. An extrapolation of amount of upswing vs Σ_a(fuel) indicated that the flux would be relatively constant in this region for the cases considered.

It was noted from experimental data that the flux traverse in the outer rod perpendicular to a diameter of the cluster can be estimated quite accurately by averaging the outer and inner fluxes at a given radius along a cluster diameter. The 45-deg distribution is similar to the 0-deg trace, with the flux values lying approximately three-quarters of the way from the 90-deg to 0-deg traces.

Results

The primary objective of this study was to determine the flux distribution in the outer rods of the seven rod fuel cluster, and the distribution shown in Fig. 2 represents the best estimate of the distribution that can be made at this time. The method of reconstructing the flux in the outer rods is subject to questioning, particularly with respect to the validity of the two-point extrapolation, but at present this is the best that can be done.

Figure 3 shows the flux distribution in the lattice cell. Values of

$$\left(\frac{\phi_{\text{fuel surface}}}{\bar{\phi}_{\text{fuel}}}\right)_{\text{center rod}}, \quad \frac{\bar{\phi}_{\text{outer rod}}}{\bar{\phi}_{\text{center rod}}}, \text{ and } \frac{\bar{\phi}_{\text{moderator}}}{\bar{\phi}_{\text{fuel}}}$$

are presented in Table 1.

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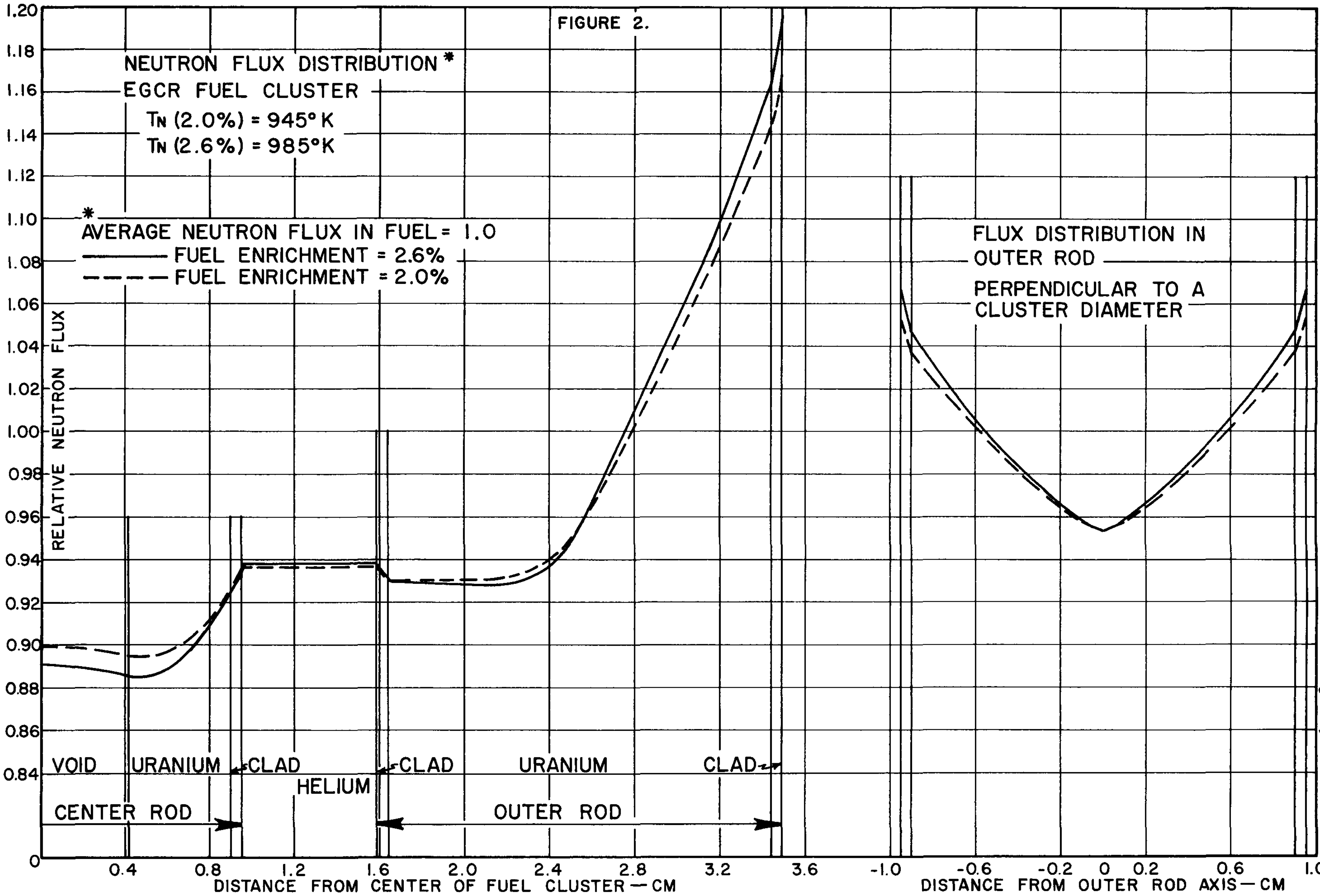
Table 1. Neutron-Flux Values in EGCR Lattice Cell

Enrichment (%)	$\left(\dfrac{\phi_{\text{fuel surface}}}{\bar{\phi}_{\text{fuel}}}\right)_{\text{center rod}}$	$\dfrac{\bar{\phi}_{\text{outer rod}}}{\bar{\phi}_{\text{center rod}}}$	$\dfrac{\bar{\phi}_{\text{moderator}}}{\bar{\phi}_{\text{fuel}}}$
2.0	1.022	1.11	1.52
2.6	1.028	1.13	1.61



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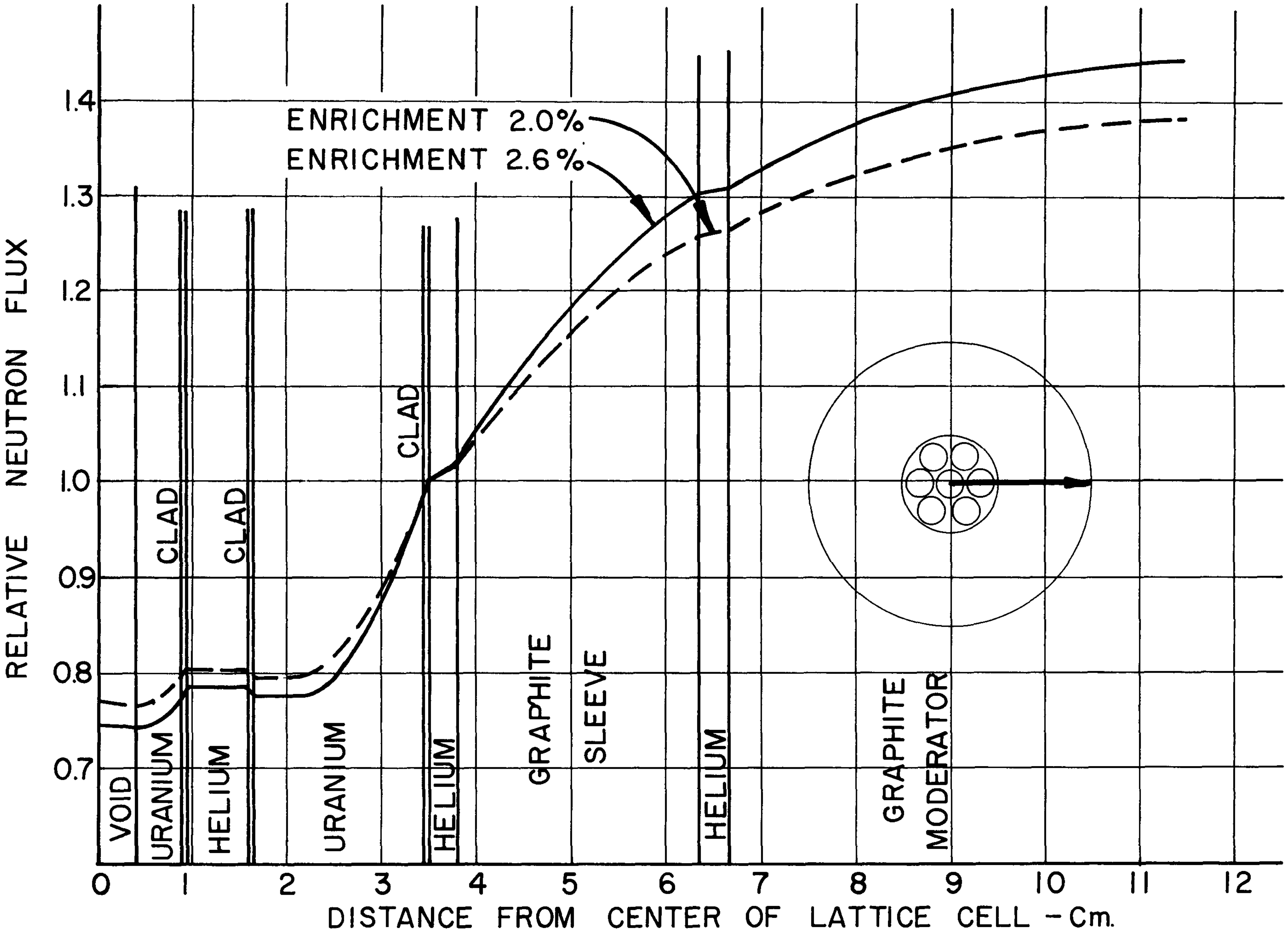

FIG. 3 - NEUTRON FLUX DISTRIBUTION - EGCR LATTICE CELL

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