ANALYSIS OF NEUTRON FLUX LEVELS AND SURVEILLANCE CAPSULE LEAD FACTORS FOR THE INDIAN POINT UNIT 2 REACTOR

S. L. Anderson
Westinghouse Electric Corp.
Nuclear Technology Division
July 1979

8701210148 870112
PDR ADOCK 05000247
P PDR

INTRODUCTION

Knowledge of the neutron environment within the pressure vessel-surveillance capsule geometry is required as an integral part of LWR pressure vessel surveillance programs for two reasons. First, in the interpretation of radiation induced properties changes observed in materials test specimens, the neutron environment to which the test specimens were exposed must be known. Second, in relating the changes observed in the test specimens to the condition of the reactor pressure vessel, a relationship between the environment at various positions within the reactor vessel and that experienced by the test specimens must be established. The former requirement is normally met by employing a combination of rigorous analytical techniques and measurements obtained with passive neutron flux monitors contained in each of the surveillance capsules. The latter information, on the other hand, is derived solely from analysis.

This report describes a discrete ordinates Sn transport analysis performed for the Indian Point Unit 2 reactor to determine the fast neutron (E>1.0MeV) flux and fluence levels within the reactor vessel and surveillance capsules; and, in turn, to develop lead factors for use in relating neutron exposure of the pressure vessel to that of the surveillance capsules.

METHODS OF ANALYSIS

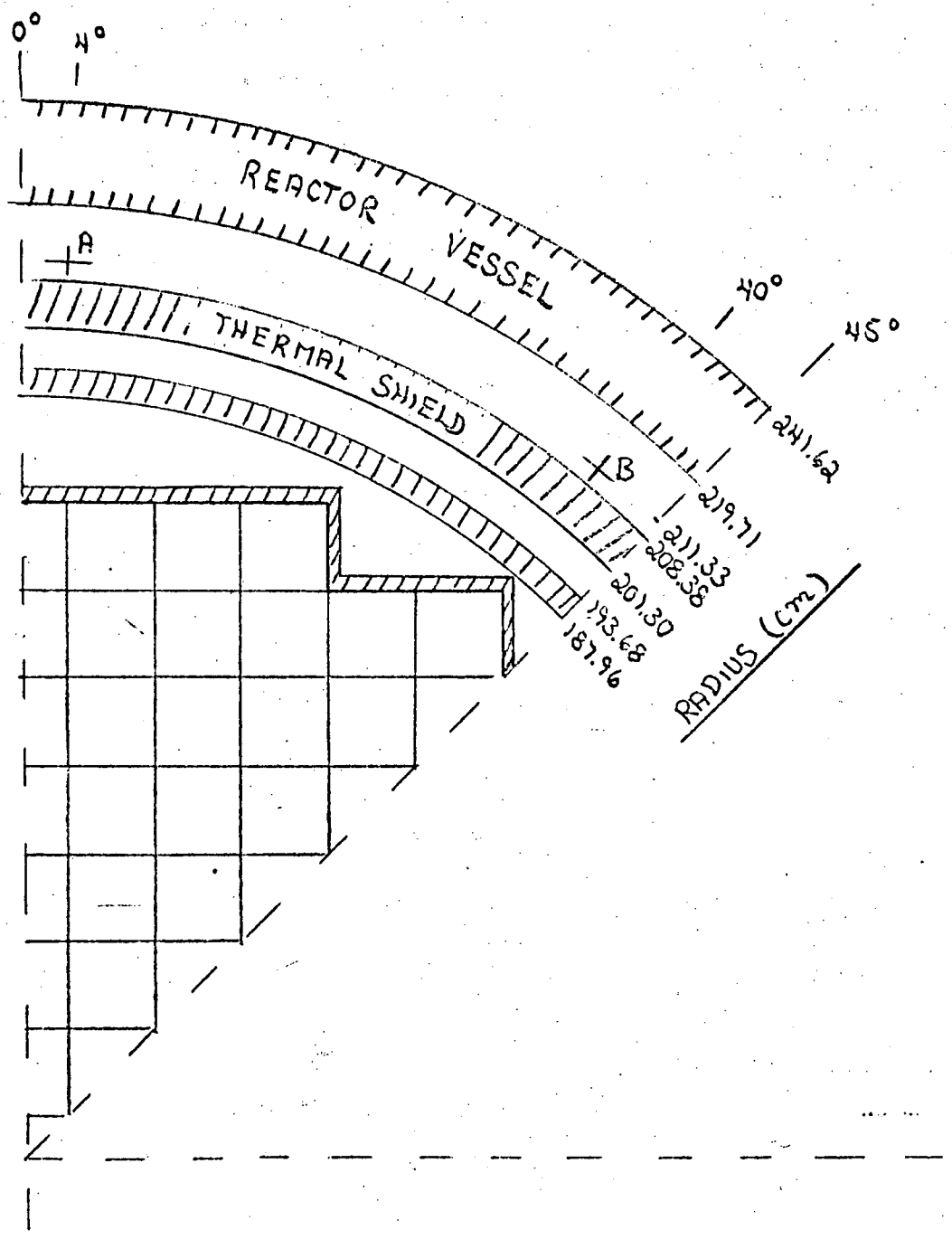
A. Reactor Geometry

A plan view of the Indian Point Unit 2 reactor geometry at the core midplane is shown in Figure 1. Since the reactor exhibits 1/8th core symmetry only 0° - 45° sector is depicted. Eight irradiation capsules attached to the thermal shield are included in the design to constitute the reactor vessel

TITLE				PAGE 4 OF
PROJECT	AUTHOR DATE	CHK'D. BY DATE	CHK'D. BY DATE	
S.O.	CALC. NO.	FILE NO.	GROUP	

FIGURE 1

R, θ GEOMETRY OF THE INDIAN POINT UNIT 2 REACTOR



REV. NO.	REV. DATE	AUTHOR DATE	CHK'D. BY DATE	CHK'D. BY DATE

WESTINGHOUSE FORM 55213D

surveillance program. Four capsules are located symmetrically at 4° from the cardinal axis while four more are positioned 40° from the cardinal axis. The relative locations of the two sets of capsules are indicated by points A and B on Figure 1.

A plan view of a single surveillance capsule attached to the thermal shield is depicted in Figure 2. The stainless steel test specimen container is one inch square and approximately 38 inches in length. The containers are positioned axially such that the specimens are centered on the core midplane, thus, spanning the central three feet of the twelve foot high reactor core.

From a neutronic standpoint the surveillance capsule structures are significant. In fact, as will be shown later, they have a marked impact on the neutron flux distributions in the water annulus between the thermal shield and the reactor vessel. Thus, in order to properly ascertain the neutron flux at the test specimen locations, the capsules themselves must be included in the analytical model. Use of a two-dimensional transport code is, therefore, mandatory.

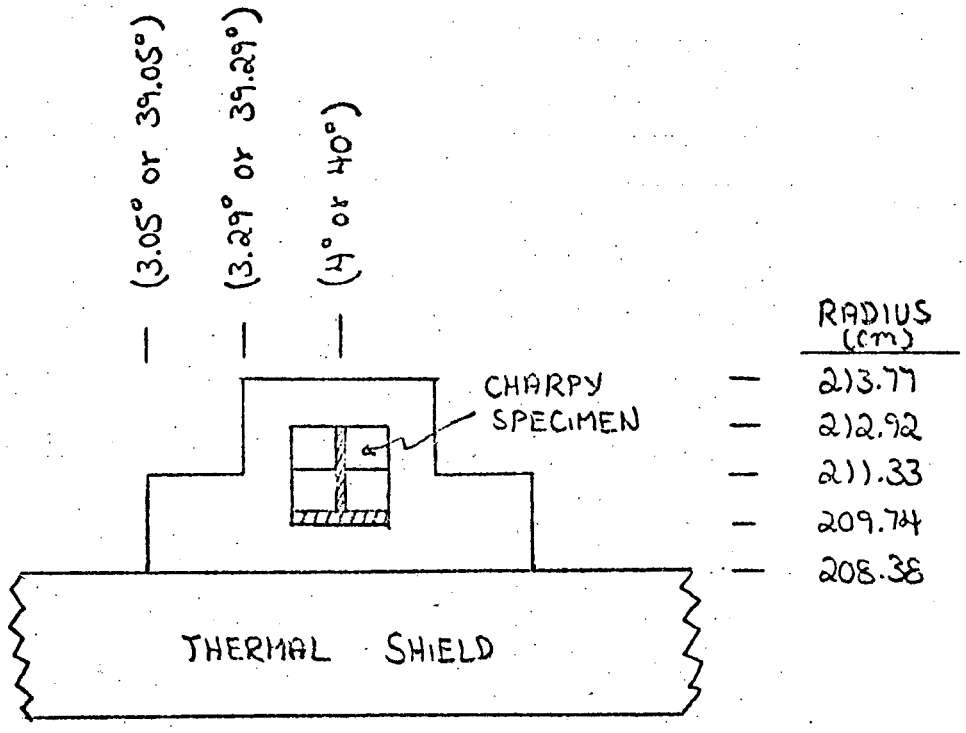
B. Transport Calculations

In the analysis of the neutron environment within the Indian Point Unit 2 reactor geometry, predictions of neutron flux magnitude and energy spectrum were made with the DOT[1] two-dimensional discrete ordinates code. The radial and azimuthal distributions were obtained from an R,θ computation wherein the geometry shown in Figures 1 and 2 was described in the analytical model. In addition to the R,θ computation, a second calculation in R,Z geometry was also carried out to obtain relative axial variations of neutron flux throughout the geometry of interest. In the R,Z analysis the reactor core was treated as

TITLE			PAGE 6 OF
PROJECT	AUTHOR DATE	CHK'D. BY DATE	CHK'D. BY DATE
S.O.	CALC. NO.	FILE NO.	GROUP

FIGURE 2

PLAN VIEW OF A REACTOR VESSEL SURVEILLANCE CAPSULE FOR INDIAN POINT UNIT 2



REV. NO.	REV. DATE	AUTHOR DATE	CHK'D. BY DATE	CHK'D. BY DATE

WESTINGHOUSE FORM 55213D

an equivalent volume cylinder of 168.54 cm. radius; and, of course, the surveillance capsules themselves were not included in the model.

Both the R,θ and the R,Z analyses employed 21 neutron energy groups, an S_8 angular quadrature, and a P_1 cross-section expansion. The cross-sections were generated via the Westinghouse GAMBIT[2] code system with processing by the APPROPOS[3] and ANISN[4] codes. The energy group structure used in the analysis is listed in Table 1.

A key input parameter in the analysis of integrated fast neutron exposure of the reactor vessel is the core power distribution. For this analysis, input power distributions included rod by rod variations for all peripheral assemblies. These were derived from statistical studies of power distributions corresponding to long-term operating conditions.

A key input parameter in the analysis of integrated fast neutron exposure of the reactor vessel is the core power distribution. For this analysis, power distributions representative of time averaged conditions derived from statistical studies of long term operation were employed.[5] These input distributions included rod by rod spatial variations for all peripheral fuel assemblies.

Having the results of the R,θ and R,Z calculations, three dimensional variations of neutron flux may be generated within the pressure vessel-surveillance capsule geometry by assuming that the following relation holds for the applicable regions of the reactor

$$\phi(R,Z,\theta,E_g) = \phi(R,\theta,E_g)\, F(Z,E_g)$$

TABLE 1

21 GROUP ENERGY STRUCTURE

GROUP	LOWER ENERGY (MeV)
1	7.79*
2	6.07
3	4.72
4	3.68
5	2.87
6	2.23
7	1.74
8	1.35
9	1.05
10	0.821
11	0.388
12	0.111
13	4.09×10^{-2}
14	1.50×10^{-2}
15	5.53×10^{-3}
16	5.83×10^{-4}
17	7.89×10^{-5}
18	1.07×10^{-5}
19	1.86×10^{-6}
20	3.00×10^{-7}
21	0.0

*Upper energy of group 1 is 10.0 MeV

where: $\phi(R,Z,\theta,E_g)$ = neutron flux at point R,Z,θ within energy group g

$\phi(R,\theta,E_g)$ = neutron flux at point R,θ within energy group g obtained from the R,θ calculations.

$F(Z,E_g)$ = relative axial distribution of neutrons within energy group g obtained from the R,Z calculation.

RESULTS OF ANALYSIS

Results of the neutron transport analysis of the Indian Point Unit #2 reactor are summarized in Figures 3 through 7 and Tables 2 and 3.

In Figure 3, the calculated maximum fast neutron flux (E>1.0MeV) levels at the pressure vessel inner radius, 1/4 thickness, and 3/4 thickness locations are presented as a function of azimuthal angle. The relative axial variation of neutron flux within the vessel is shown in Figure 4. Axial variations of fast neutron (E≥1.0MeV) flux can be obtained using Figures 3 and 4.

In Figure 5, the predicted fast neutron (E>1.0MeV) fluence (for a period of 32 Effective Full Power Years of Reactor Operation at a power level of 2700 Mwt) for Indian Point Unit 2 is given as a function of radial position within the vessel wall.

2760

The neutron flux (E>1.0MeV) distributions within the surveillance capsules are highlighted in Figures 6 and 7. Figure 6 depicts the azimuthal variation of fast neutron flux at a radius equal to that of the surveillance capsule centerline (R=211.33). The perturbation introduced by the presence of the capsule material is clearly evident. In Figure 7, the radial gradient of neutron flux

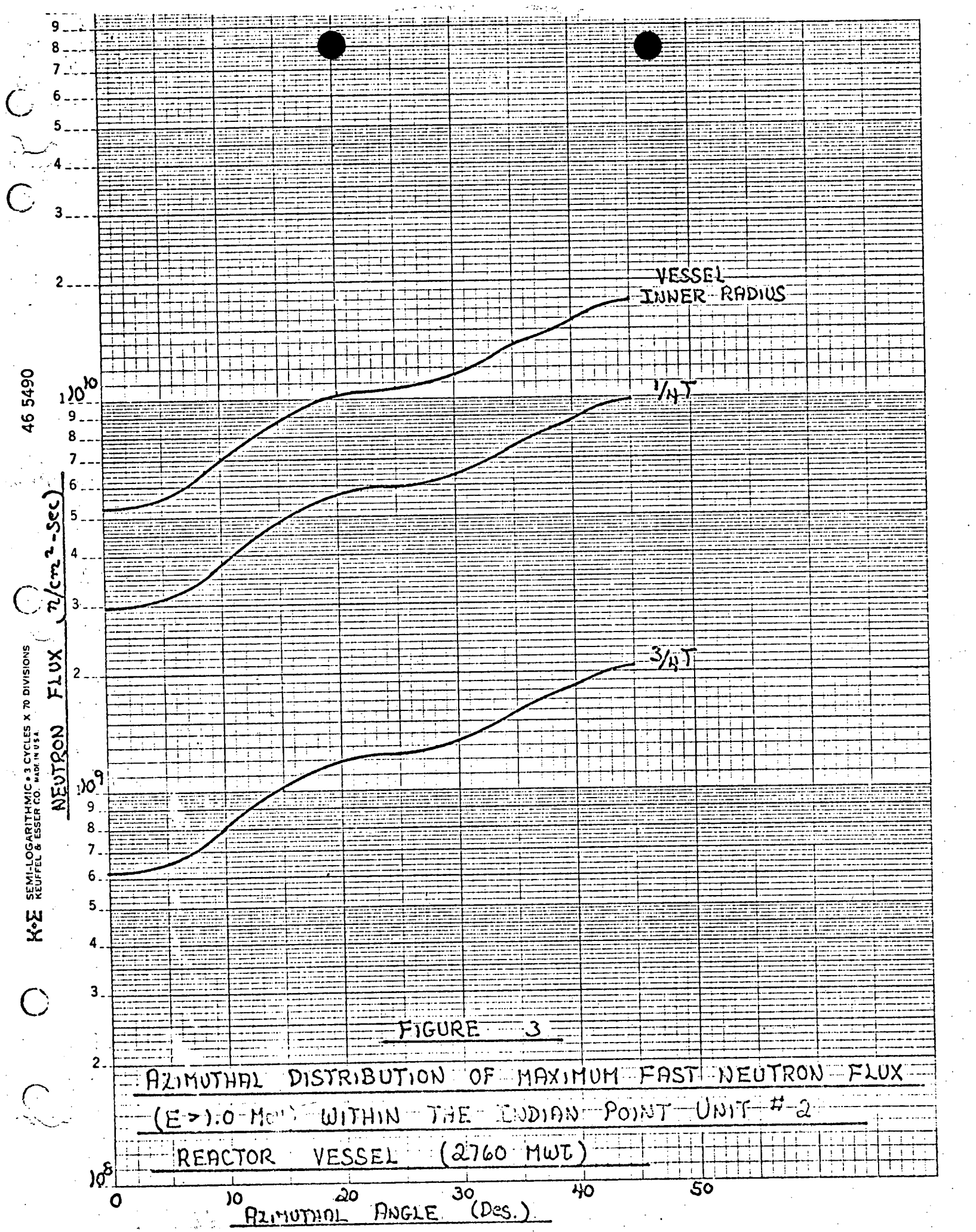


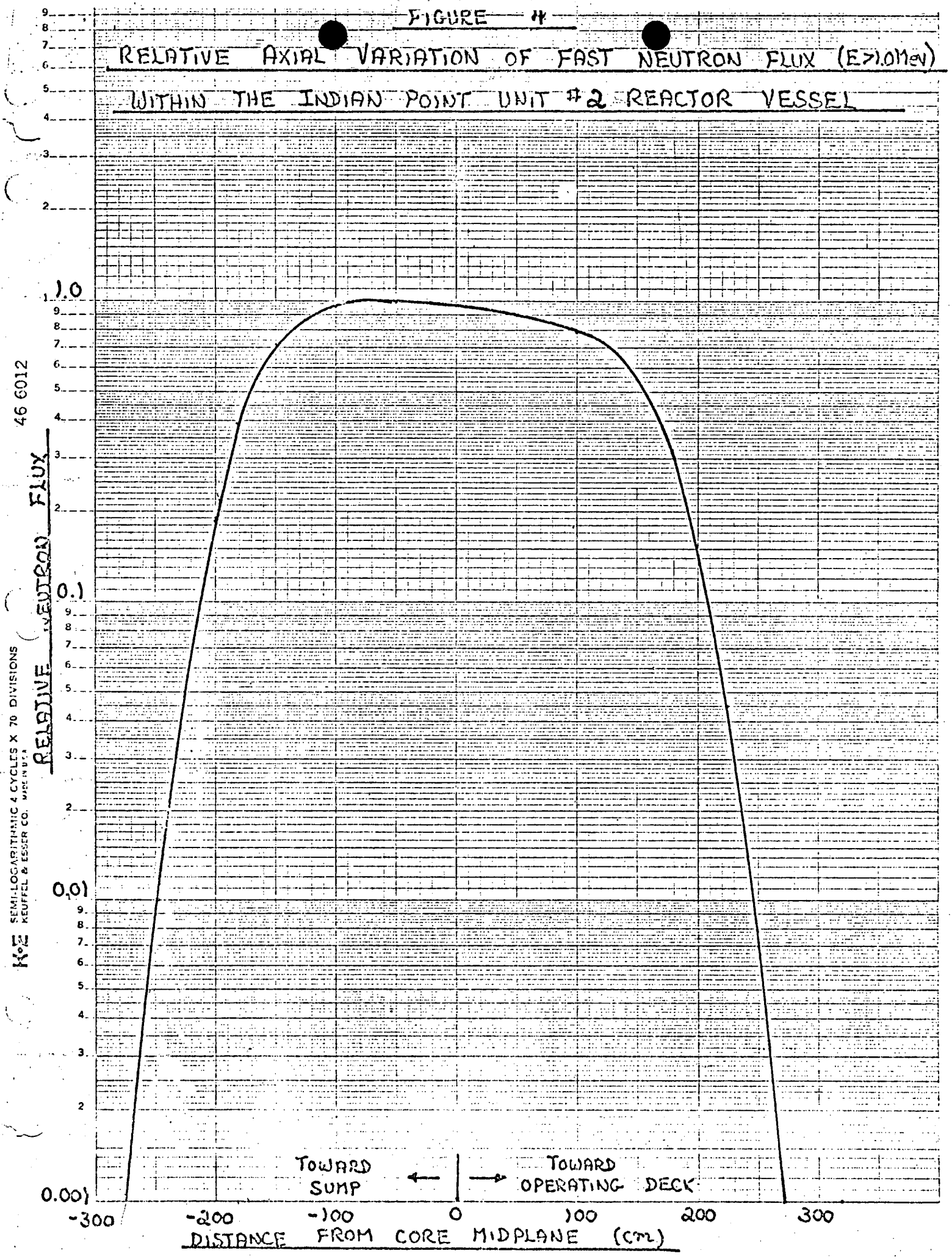
FIGURE 3

AZIMUTHAL DISTRIBUTION OF MAXIMUM FAST NEUTRON FLUX (E>1.0 MeV) WITHIN THE INDIAN POINT UNIT #2 REACTOR VESSEL (2760 MWt)

FIGURE #

RELATIVE AXIAL VARIATION OF FAST NEUTRON FLUX (E>1.0MeV)

WITHIN THE INDIAN POINT UNIT #2 REACTOR VESSEL



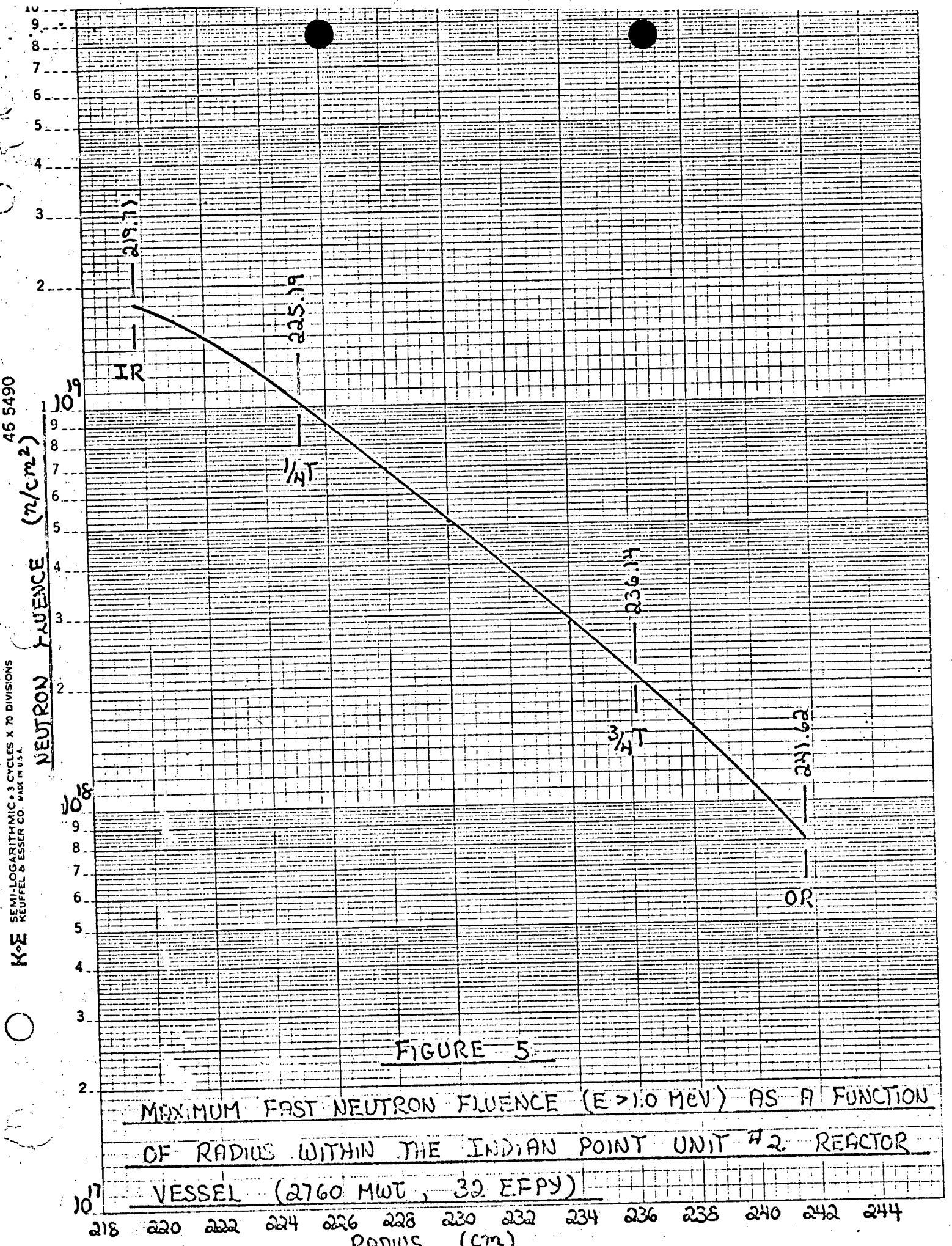
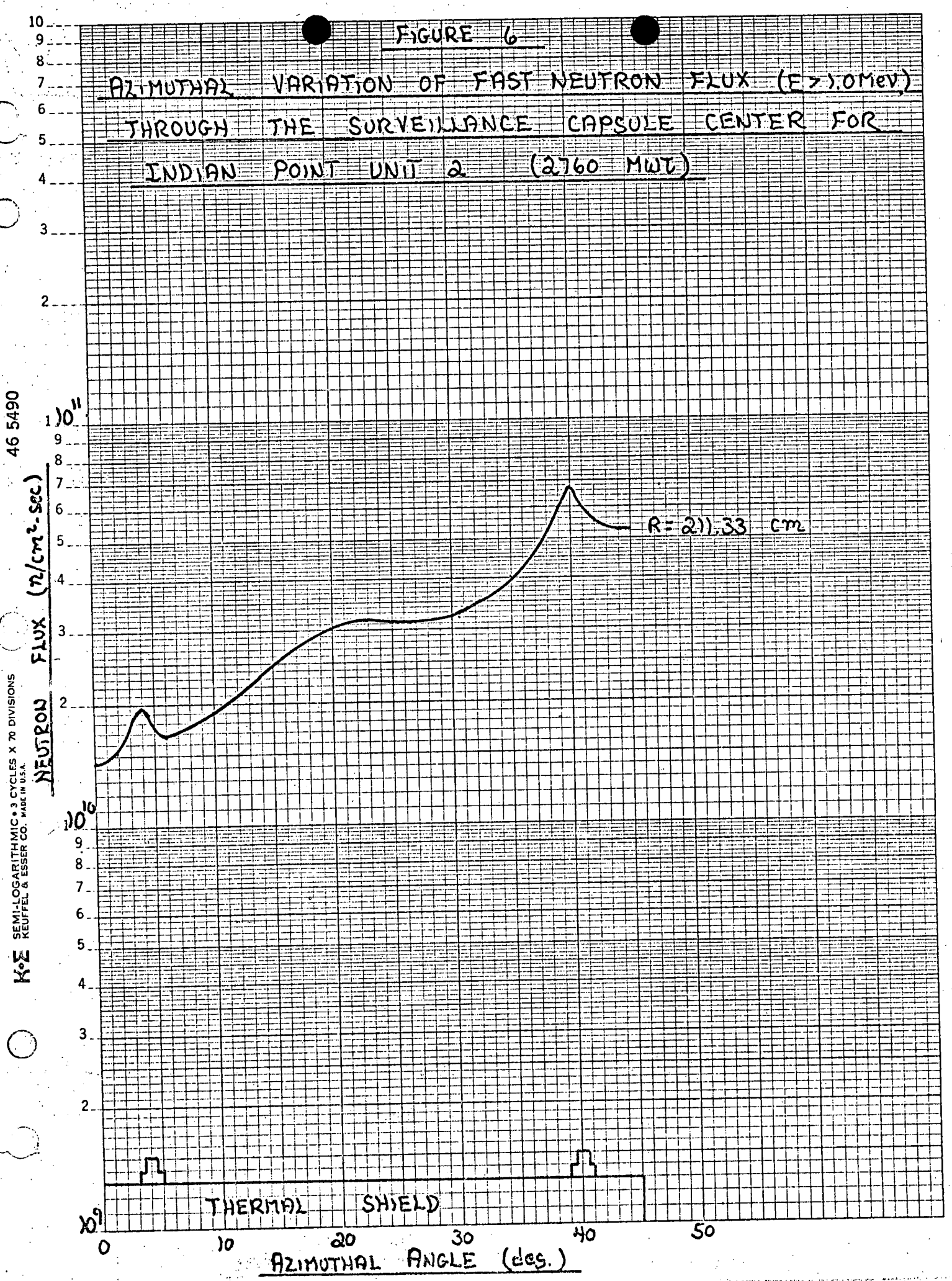


FIGURE 5

MAXIMUM FAST NEUTRON FLUENCE (E > 1.0 MeV) AS A FUNCTION OF RADIUS WITHIN THE INDIAN POINT UNIT #2 REACTOR VESSEL (2760 MWt, 32 EFPY)

FIGURE 6

AZIMUTHAL VARIATION OF FAST NEUTRON FLUX (E > 1.0 MeV) THROUGH THE SURVEILLANCE CAPSULE CENTER FOR INDIAN POINT UNIT 2 (2760 MWt)



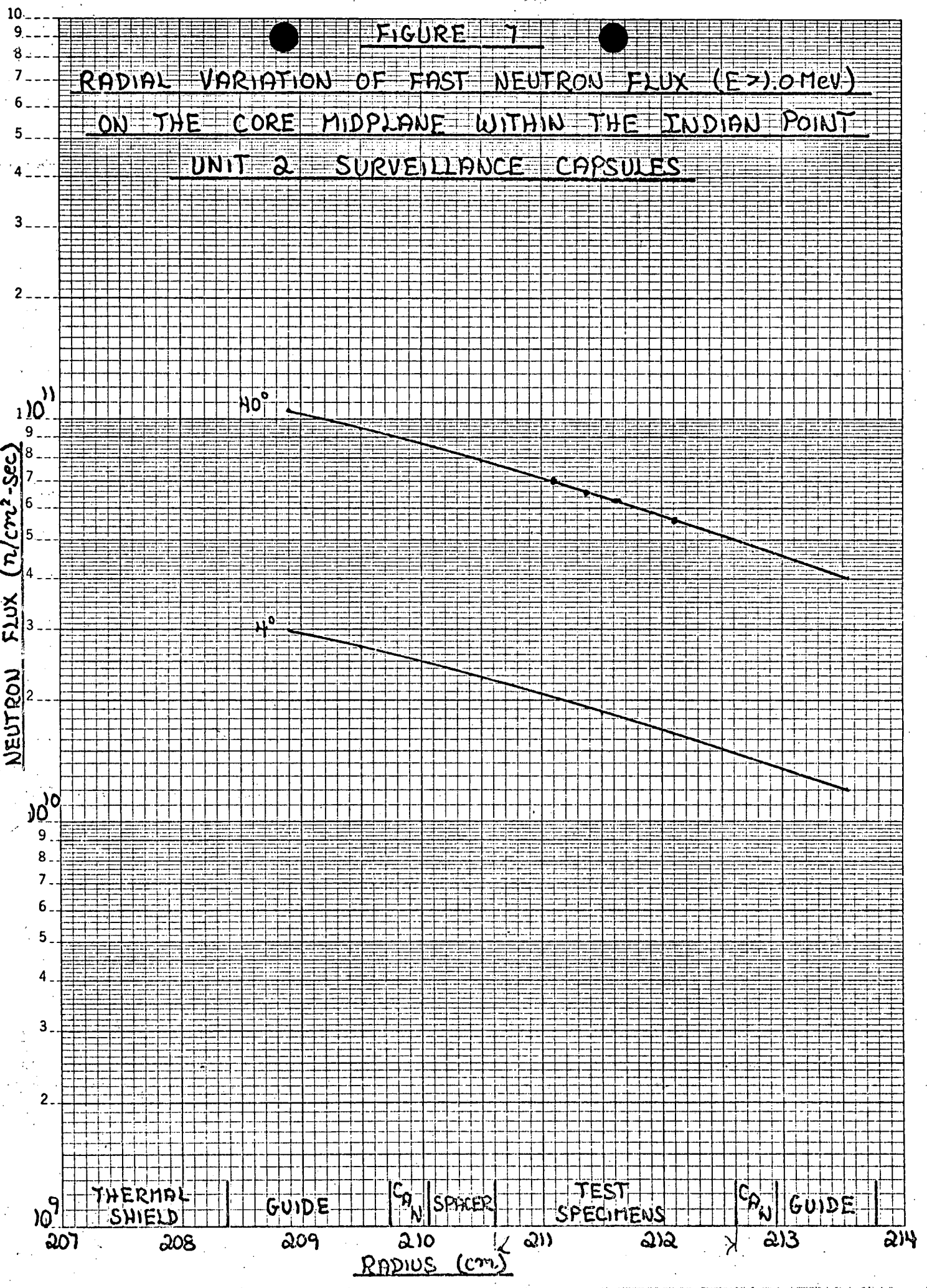


FIGURE 7
RADIAL VARIATION OF FAST NEUTRON FLUX (E>1.0 MeV)
ON THE CORE MIDPLANE WITHIN THE INDIAN POINT
UNIT 2 SURVEILLANCE CAPSULES
NEUTRON FLUX (n/cm²-sec)
10¹¹
10¹⁰
10⁹
40°
4°
THERMAL SHIELD
GUIDE
CAN
SPACER
TEST SPECIMENS
CAN
GUIDE
207
208
209
210
211
212
213
214
RADIUS (cm.)
46 5490
K+E SEMI-LOGARITHMIC • 3 CYCLES X 70 DIVISIONS
KEUFFEL & ESSER CO. MADE IN U.S.A.

(E>1.0MeV) at the aximuthal center of the capsules is depicted. It is of interest to note that from the core side of the test specimens to the pressure vessel side of the test specimens the fast neutron flux drops approximately 30%.

Lead factors for each of the Indian Point Unit 2 surveillance capsules are presented in Table 2. The lead factor is defined as

$$LF = \frac{\phi_C}{\phi_{PV}}$$

where: ϕ_C = neutron flux (E>1.0MeV) within the surveillance capsule

ϕ_{PV} = maximum neutron fluxe (E>1.0MeV) at the pressure vessel inner radius.

Since the Westinghouse surveillance capsules contain passive neutron flux monitors located at the center of the core side row of charpy specimens, the geometric center of the capsule, and the center of the pressure vessel side row of charpy specimens, lead factors are presented for each of these three radial positions.

In Table 3, the relative neutron flux spectra calculated to exist at the center of the surveillance capsules are listed. The data in each case have been normalized to a total neutron flux of 1 n/cm^2-sec. This spectral information may be used to develop spectrum averaged reaction cross-sections for use in analyzing neutron dosimeter packages contained within the surveillance capsules. A set of cross-sections in the group structure listed in Table 3 are given in Table 4 for selected fast neutron reactions. These data are based on information found in the ENDF IV dosimetry file.

TABLE 2

LEAD FACTORS FOR INDIAN POINT UNIT 2

SURVEILLANCE CAPSULES

LOCATION* WITHIN CAPSULE	4° CAPSULES (U,V,W,X)	40° CAPSULES S,T,Y,Z)
R = 211.10 cm	1.14	3.90
R = 211.33 cm	1.09	3.72
R = 212.10 cm	0.925	3.14

Avg. $= \frac{3.90 + 3.14}{2}$

$= \frac{7.04}{2} = 3.52$

*Locations correspond to the center of front charpies, center of capsule, and center of rear charpies.

ADEQUACY OF THE ANALYTICAL METHOD

The accuracy of the calculations described in this report depends on the ability to define appropriate core power distributions, the adequacy of the cross-section libraries, and on the appropriateness of the transport model itself. Taken as a whole, these factors combine to yield an overall uncertainty in the analytical results. The overall Westinghouse analytical method described herein has been compared with measurements obtained at a number of operating PWR's. These comparisons have been described in the open literature[6,7], but will also be summarized here.

Figures 8 and 9 depict a comparison of the calculated activity of four fast neutron detectors with measurements obtained from surveillance capsules removed from nine 2-loop PWR's. The calculations are in good agreement with the measurements. Since these monitors respond to different portions of the neutron energy spectrum, the agreement is an indication that both the energy spectra and flux magnitudes are being determined well by the analytical method.

Figures 10 and 11 also show comparisons of calculations with measurements of fast neutron flux. In this case, the data were obtained external to the reactor vessel in the air gap between the vessel and the biological shield. The data shown in Figure 10 was obtained at a 4-loop reactor similar to Indian Point Unit 2; while the data in Figure 11 was obtained at a 3-loop reactor. Again, the agreement between calculation and measurement is excellent.

Based on the comparisons presented in Figures 8 through 11, it is concluded that the Westinghouse design method is appropriate for the analysis of large PWRs and that the results presented herein are applicable to the Indian Point Unit 2 reactor.

TABLE 3

RELATIVE NEUTRON FLUX SPECTRUM AT THE CENTER OF THE SURVEILLANCE CAPSULES

GROUP NO.	RELATIVE NEUTRON FLUX 4° CAPSULES	RELATIVE NEUTRON FLUX 40° CAPSULES
1	.00163	.000852
2	.00552	.00283
3	.00743	.00447
4	.00672	.00467
5	.0103	.00793
6	.0191	.0155
7	.0255	.0219
8	.0332	.0299
9	.0403	.0373
10	.0406	.0382
11	.125	.122
12	.141	.144
13	.0564	.0586
14	.0420	.0438
15	.0311	.0326
16	.0750	.0800
17	.0560	.0595
18	.0607	.0661
19	.0466	.0508
20	.0489	.0530
21	.126	.126
ϕTotal	1.30×10^{11}	5.34×10^{11}

TABLE 4

REACTION CROSS-SECTIONS FOR ANALYSIS OF FAST NEUTRON MONITORS

		σ(E) barns	
GROUP	$Fe^{54}(n,P)Mn^{54}$	$Ni^{58}(n,P)Co^{58}$	$Cu^{63}(n,\alpha)Co^{60}$
1	.592	.607	.035
2	.572	.608	.0098
3	.464	.535	.00085
4	.325	.388	0.0
5	.145	.222	0.0
6	.0494	.113	0.0
7	.0194	.0371	0.0
8	.0089	.0112	0.0
9	.0015	.0043	0.0
10-21	0.0	0.0	0.0
$\bar{\sigma}(4°)$	.087	.113	.000785
$\bar{\sigma}(40°)$	.067	.0899	.000490

$$\bar{\sigma} = \frac{\int_0^\infty \sigma(E)\,\phi(E)\,dE}{\int_{1\ \mathrm{MeV}}^\infty \phi(E)dE}$$

In regard to the use of a P_1 vs P_3 scattering approximation in the analysis, the comparisons in Figures 8 through 11 demonstrate the adequacy of a P_1 approach. The variability of measured flux levels from plant to plant and cycle to cycle outweighs the uncertainty in the calculations, thus, the use of a P_3 scattering approximation would not result in a significant improvement in analytical vs. experimental comparisons.

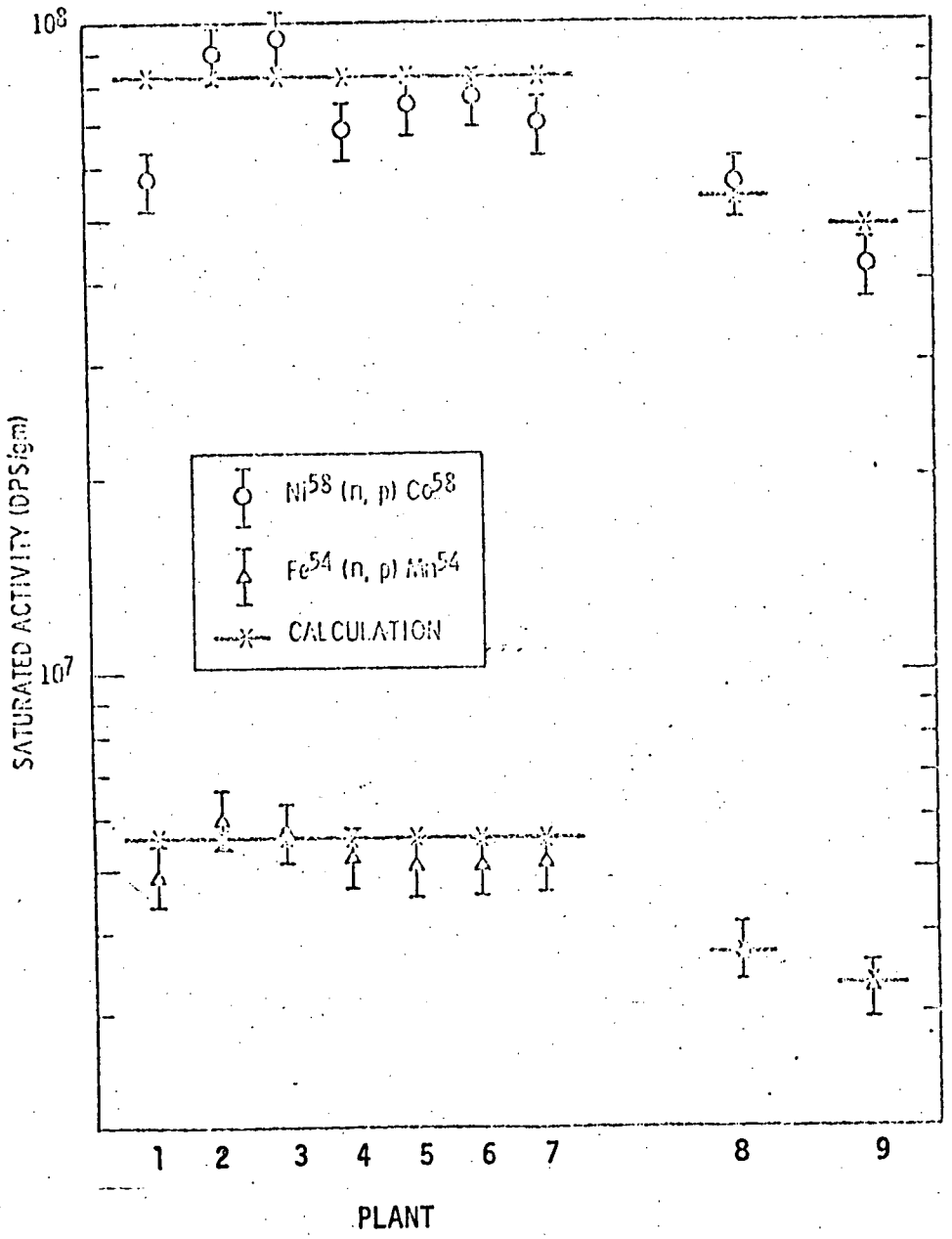


FIGURE 8 Activation Neutron Flux Monitor Saturated Activity at the Dosimeter Block Location for Nine 2-Loop PWR Power Plants. (Normalized to 1650 MWT)

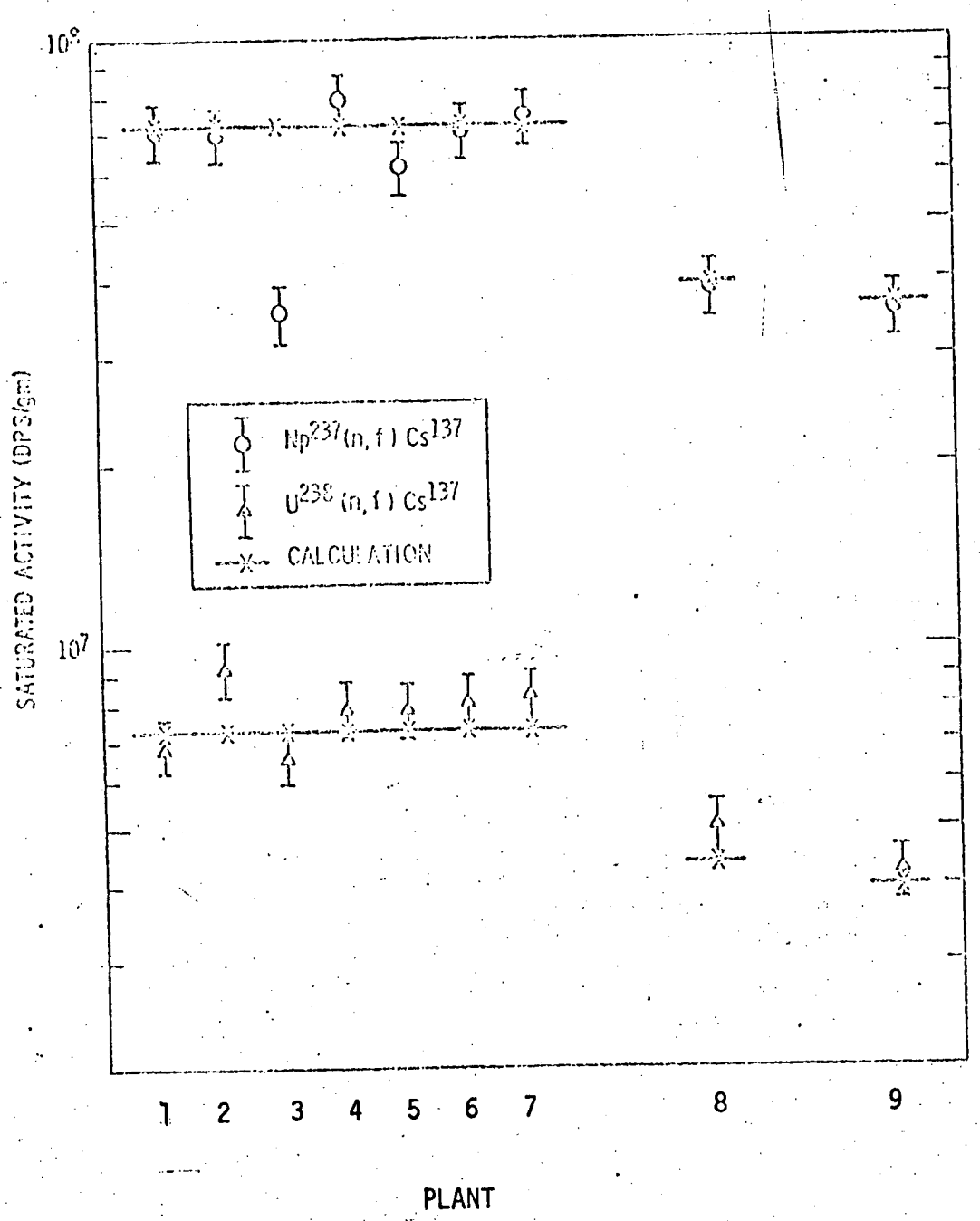


Figure 9 - Fission Neutron Flux Monitor Saturated Activity at the Dosimeter Block Location for Nine 2-Loop PWR Power Plants. (Normalized to 1650 MWT)

FIGURE 10

CALCULATED AND MEASURED AZIMUTHAL VARIATION OF $Fe^{54}(n,P)Mn^{54}$ RESPONSE ON THE CORE MIDPLANE - PLANT A

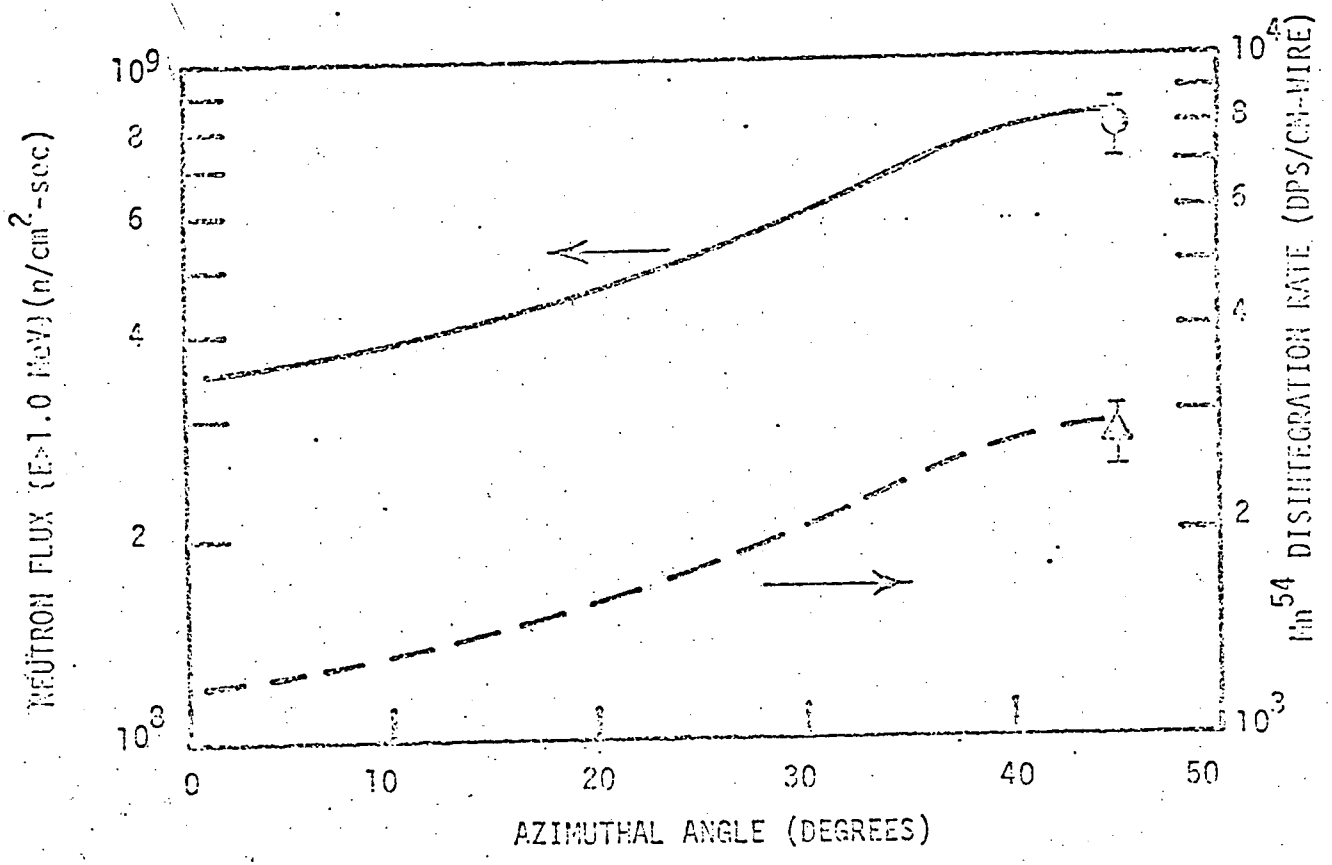
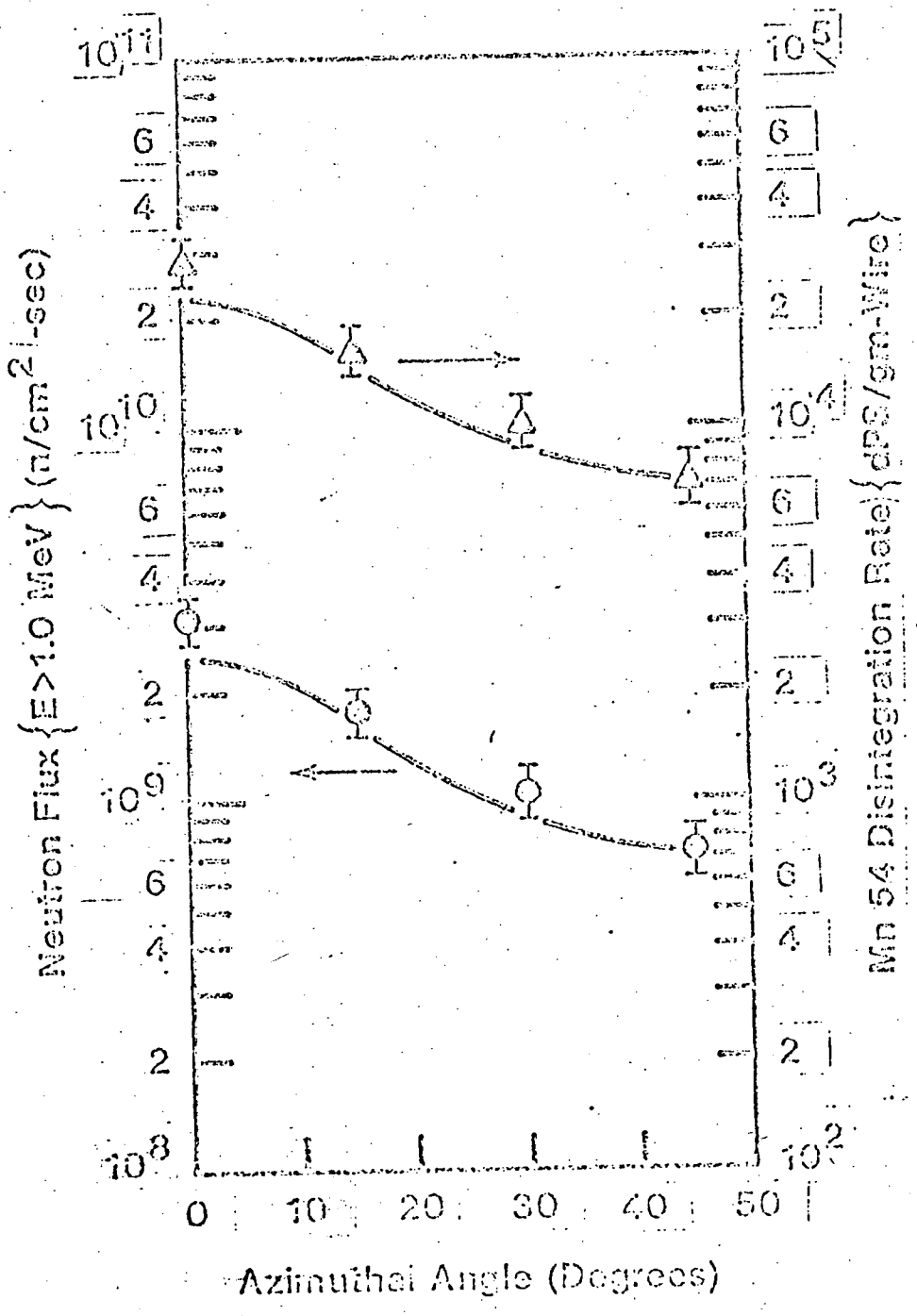


FIGURE 11

CALCULATED AND MEASURED AZIMUTHAL VARIATION OF $Fe^{54}(n,P)Mn^{54}$ RESPONSE ON THE CORE MIDPLANE - PLANT B



REFERENCES

1. Soltesz, R. G. Et al, "Nuclear Rocket Shielding MEthods, Modification, Updating and Input Data Preparation - Volume 5 - Two-Dimensional Discrete Ordinates Transport Technique," WANL-PR-(LL)-034, August 1970.

2. Collier, G., et al, "Second Version of the GAMBIT Code", WANL-TME-1969, November 1969.

3. Soltesz, R. G., et al, "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation - Volume 3, Cross-Section Generation and Data Processing Techniques", WANL-PR-(LL)-034, August 1970.

4. Soltesz, R. G. et al, "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation - Volume 4 - One-Dimensional Discrete Ordinates Transport Technique," WANL-PR-(LL)-034, August 1970.

5. Internal memorandum (Westinghouse Proprietary)

6. McElroy, W.N. et al, "Development and Testing of Standardized Procedures and Reference Data for LWR Surveillance," IAEA Specialists Meeting, Vienna, Austria, February 1979.

7. Anderson, S. L and Capo, M. A., "Measurements and Calculatuions of Neutron Streaming within a Pwr primary Shield Annulus", Trans. AM. Nucl. Soc., 617 (1976).

Attachment B

Analysis of Neutron Flux Levels and Surveillance Capsule Lead Factors for the Indian Point Unit 2 Reactor Using a Low Leakage Core