Criticality Studies of a Neutron Multiplier Lattice

by
R. C. Lloyd, S. R. Bierman,
E. D. Clayton and B. M. Durst

April 1976

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TABLE OF CONTENTS

	<u>Page</u>
INTRODUCTION.	1
EXPERIMENTAL DESCRIPTION.	1
MEASUREMENT DATA	5
Critical Approach Measurements and Temperature Effects	5
Effect of Voids in the Central Region	5
Effect of Various Materials in Central Region.	12
Effect of Voids and Various Materials Outside Lattice Region	12
Pulsed Neutron Source Measurement Results	12
Neutron Multiplication and Relative Flux Inside the Central Flux Trap	19
Effect of Rod Removal from the Lattice Region	25
Typical Calculation of an Operation Loading	25
CONCLUSIONS	28
ACKNOWLEDGEMENTS	29
REFERENCES	30

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INTRODUCTION

There has been considerable need for neutron radiation units with high neutron yield.^(1,2) Sources of various types have been used for this purpose for many years, but have lacked the intensity and volume, with level flux, for larger sample irradiation. While the research reactor can produce desirable neutron flux, the operation is rather costly. A neutron multiplier has been designed using a subcritical lattice of enriched uranium rods with a central cylindrical flux trap.^(3,4) This lattice unit was loaded at the Critical Mass Laboratory to study aspects of criticality including temperature, voids, and added material. Data from these experiments are presented in this report.

EXPERIMENTAL DESCRIPTION

The experimental system consisted of a lattice assembly, reflected and moderated with water, in a cylindrical tank having a diameter of 102 cm. Measurements were made at lattice spacings of 0.55 and 0.60 inch. A photograph of the assembly is shown in Fig. 1. The canned fuel rods were a U+Al alloy having a ^{235}U enrichment of 93.2 wt%; they were 0.40 inch in diameter and 23-3/4 inches in length. Detailed dimensions are shown in Fig. 2.

The lattices were safely loaded, using the critical approach method; this also provided the extrapolation for the critical number of rods. At selected fuel loadings, pulsed neutron source measurements were made to determine reactivity and prompt neutron decay rates. The effect of voids and various materials placed in the central region and outside the loaded lattice was measured in terms of the number of rods required for criticality. Dimensions of these materials are given in Table I. To measure the reactivity

FUEL ROD

CLAD MATERIAL: ALUMINUM, 1100

LOAD MATERIAL: 28 w/o U IN Al, 93.2 w/o ^{235}U IN U, 1 gm U (93.2)/cm³. 3 FUEL PIECES PER ROD. 22.224 gm ^{235}U PER ROD

FIGURE 2

TABLE ITEST MATERIAL DETAILS

<u>Material</u>	<u>Dimensions</u>
Al Void Tubes	1.00 in. OD x 0.063 in. wall
Al Void Tubes	2.00 in. OD x 0.063 in. wall
Al Void Tubes	3.00 in. OD x 0.094 in. wall
Al Void Tubes	5.00 in. OD x 0.094 in. wall
Graphite Rod	1.75 in. diam, 1.65 g/cc
Plexiglas Rod	1.25 in. diam, 1.18 g/cc
Wood Dowel	1.25 in. diam, 0.72 g/cc
Bi Metal (granular)	1.341 in. ID, 1.401 in. OD Al tube, 5.44 g/cc
Mineral Oil	1.341 in. ID, 1.401 in. OD Al tube

effect of temperature, the water was heated using an electrical Calrod unit and a steam coil. Temperatures were monitored using thermocouples, and the number of rods required for criticality was determined as a function of water temperature.

MEASUREMENT DATA

Critical Approach Measurements and Temperature Effects

Critical approach measurements were carried out for the 0.55 and 0.60 in. lattice spacings at room temperature ($\sim 21^{\circ}\text{C}$) and for the 0.60 in. lattice at temperatures up to 71°C . Table II and Fig. 3 show the results of these critical approach measurements made at various temperatures with the 0.60 in. lattice. The critical number of fuel rods at 21°C , with a 2 in. central void and a 37 rod equivalent flux trap, was 271.4. The same system with the 2 in. diameter central tube half filled with water was 276.3. When either system was heated, the maximum decrease in critical number of rods, heating from 21°C , was about 2.6, while the system having the 2 in. diameter void leveled off at a somewhat lower temperature.

The 0.55 in. lattice spacing showed a critical number of rods to be 340.9 with a 37 cell flux trap, and 360.2 with a 55 cell flux trap. Both lattices had a 2 in. central void.

Effect of Voids in the Central Region

Table III shows the worth of voids in the central region of the 0.55 in. lattice in terms of the effect on the critical number of rods. Fig. 4 shows the void effect in the 0.55 in. lattice with both the 37 and 55 cell flux traps. The minimum appears to occur slightly above 30%. Table IV lists the critical number of rods for the 0.60 in. lattice with various void tubes in the central region. Fig. 5 shows the critical number of rods as the void percent in the

TABLE II

CRITICALITY OF NEUTRON MULTIPLIER 0.60 IN. LATTICE
 (EFFECT OF TEMPERATURE CHANGE)

<u>Exp. No.</u>	<u>Temperature (C°)</u>	<u>N_c (Rods)</u>	<u>Max. Temp. Effect (Rods)</u>	<u>Condition</u>
007	21	271.42	2.62	2 in. Diam. Center Void, 37 Rod Equivalent Flux Trap
010	32	269.08		
011	50	268.80		
012	71	269.17		
015	23	276.25	2.61	2 in. Diam. Center Void - 1/2-filled with H ₂ O 37 Rod Equivalent Flux Trap
017	47	273.73		
018	70	273.64		

EFFECT OF TEMPERATURE CHANGE ON CRITICALITY OF NEUTRON MULTIPLIER

FIGURE 3.

TABLE III.

CENTER VOID EFFECT AS A FUNCTION OF
CHANGE IN THE FLUX TRAP SIZE
 (0.55 IN. LATTICE)

<u>Exp.</u> <u>No.</u>	<u>Void^(a)</u> <u>Diameter</u> <u>(in.)</u>	<u>N_c</u> <u>(Rods)</u>	<u>%</u> <u>Void</u>
(37 Tube Flux Trap)			
001A	0	348.43	2.64
	1	345.80	8.10
	2	341.76	32.4
	3	371.91	72.9
(55 Tube Flux Trap)			
002A	0	374.74	3.92
	2	360.38	21.8
	3	361.52	49.1

a. 0 void diameter was measured by filling the 2 in. Al center tube with water. Al assumed as void in % void calculation.

EFFECT OF CENTER VOIDS

FIGURE 4.

FIGURE 1

TABLE IV.

CENTER VOID EFFECT AS A FUNCTION OF TEMPERATURE
(0.60 IN. LATTICE)

<u>Exp.</u> <u>No.</u>	<u>Void^(a)</u> <u>Diameter</u> <u>(in.)</u>	<u>N_c</u> <u>(Rods)</u>	<u>% Void</u> <u>(Vol%)</u>
<u>(21°C Temperature)</u>			
009	0	281.87	3.30
	1	278.69	6.81
	2	271.35	27.2
	3	280.64	61.3
<u>(74°C Temperature)</u>			
013	0	275.12	3.30
	1	273.16	6.81
	2	268.95	27.2
	3	280.70	61.3

a. 0 void diameter was measured by filling the 2 in. Al center tube with water. Al assumed as void in % void calculations.

EFFECT OF CENTER VOIDS

FIGURE 5.

flux trap increases. Here, the minimum critical number of rods occurs at slightly less than 30% void.

Effect of Various Materials in Central Region

To determine the sensitivity of critical number of rods to materials being placed in the flux trap region (such as would occur during use of the multiplier), various materials were placed in the central region. Table V gives the worth of these various materials, in terms of rods, on reactivity increase or decrease. The 1/3 fuel rod had a worth of +4.4 fuel rods, the 1-3/4 in. OD graphite rod was the equivalent of +3.3 and the source mockup was -1.9. These materials were positioned as they would be in normal usage. To get a maximum ability to reduce reactivity, a 2 in. void was filled with an aqueous 6 g boric acid (H_3BO_3)/liter solution, providing about a 29 rod reduction. In actual usage, where only half the center hole would be available to add poison, the effect of this central poison would be reduced by 50%.

Effect of Voids and Various Materials Outside Lattice Region

Knowledge of the effect of irradiation tubes outside the lattice is needed for operation of the multiplier; also, the effect of various materials that might be inserted in these tubes. The effects that were measured are given in Table VI. A cylindrical void (empty aluminum tube) placed against the lattice plate had a considerable negative reactivity worth of about 2.3 rods for the 2 in. diameter tube and about 6 rods for the 5 in. diameter tube. It is also of interest that adding a 1-3/4 in. diameter graphite rod in the 2 in. diameter void increased the reactivity by the equivalent of 3.6 fuel rods. Even more interesting is the fact that this is a 1.3 fuel rod increase over having water in that region.

Pulsed Neutron Source Measurement Results

Pulsed neutron source measurements were performed on the lattice assembly having a 0.6-inch spacing and a center flux trap equivalent to 37 fuel rods.

TABLE V.EFFECT OF MATERIAL IN CENTER TUBE(Center Tube 1/2-Filled With H_2O - 0.60 in. Lattice Spacing)

<u>Material</u>	<u>Position</u>	<u>Reactivity Worth (Fuel Rods)</u>
1/3 Fuel Rod	In Void - Above Center	+4.37
1 Fuel Rod	Centered Vertically	+11.46
Graphite (1-3/4 in. OD)	In Void in Top Half	+3.29
Source Mockup (200 g SS)	In Water In Bottom Half	-1.86
1.34-in. Tube of Bismuth	In Void in Top Half	+0.87
1.34 in. Tube of Mineral Oil	In Void in Top Half	-2.40
17 Cd-Covered Cu Foils Spaced 2" Vertically	Centered Vertically	-6.68 ^(a)
17 Cu Foils Spaced 2" Vertically	Centered Vertically	-1.68 ^(a)
6 g/l Boric Acid-Filled 2" Void Tube	Centered Vertically	-29.1 ^(b)
BF ₃ Tube	Centered Vertically	-1.48 ^(b)

a. Part of effect due to change in water height in center tube.

b. Center tube completely void of H_2O .

TABLE VI.

<u>EFFECT OF OUTSIDE LATTICE VOID TUBES AND MATERIAL</u> (Center Tube 1/2-Filled With H ₂ O - 0.60 in. Lattice Spacing)		
<u>Void Diameter</u>	<u>Placement Description</u>	<u>Reactivity Worth^(b) (Fuel Rods)</u>
2 in.	Flush to Lattice Plate ^(a)	-2.13
2 in.	1 in. from Lattice Plate	-0.53
2 in.	2 in. from Lattice Plate	-0.09
5 in.	Flush to Lattice Plate	-5.97
2 in.	Void Filled with Graphite Rod (1-3/4 in.) Flush to Lattice Plate Compared to Void	+3.63
2 in.	Void Filled with Graphite Rod Flush to Lattice Plate	+1.25
2 in.	Flush to Lattice Plate (Recheck)	-2.38
2 in.	Flush to Lattice Plate (Recheck)	-2.40
2 in.	Void/1.34 Tube Mineral Oil Flush to Lattice Plate	-0.90
2 in.	Void/1.34 Tube Bismuth Flush to Lattice Plate	-1.31
2 in.	Void/1-1/4 in. Wood Flush to Lattice Plate	-1.37
2 in.	Void/1-1/4 in. Plexiglas Flush to Lattice Plate	-0.81
2 in.	Void Tube of Mineral Oil Flush to Lattice Plate	+0.06

a. Edge of lattice plate to edge of fuel loading was 0.690 in.

b. All worths compared to water except as noted with the graphite rod.

The measurements were conducted at room temperature ($\sim 21^{\circ}\text{C}$) and with the 2-in. void tube axially through the center of the flux trap. The pulsed neutron source was positioned in the reflector region away from the fuel rods with its tritium target at the mid-plane of the lattice. Two neutron detectors were used, independently, for data accumulation. The primary detector, a BF_3 tube, was located at the center of the lattice in the 2-in. void tube to minimize spatial effects in the data. However, as the delayed critical condition was approached, a secondary neutron detector (a boron-lined proportional counter), located in the reflector region opposite the pulsed neutron source, had to be relied upon for data accumulation.

Measurement data were obtained on five symmetrical fuel rod loadings nearest the delayed critical loading as predicted by the companion critical approach measurements. Each set of data was least squares fit to a single exponential term plus a constant to determine the prompt neutron decay rate for each fuel loading. Each set of data was also analyzed using the Garelis-Russell⁽⁵⁾ method and the Gozani⁽⁶⁾ method to determine the reactivity of each fuel loading. The results of these data analyses are shown graphically in Fig. 6. Experimentally determined corrections of about 0.5% in mass were applied to all the measurement results to account for the presence of the pulsed neutron source system. The number of fuel rods required for criticality was 271.42 without the pulsed neutron system and 272.9 with it in position.

As expected, no spatial effects were observed in the prompt neutron decay rates. Data from both detectors yielded decay rates which varied linearly over the fuel loading range, with no apparent dependency on the data source. By linearly extrapolating these subcritical data to the prompt critical condition, a prompt critical loading of 281.5 fuel rods is obtained.

PULSED NEUTRON SOURCE MEASUREMENT RESULTS - NEUTRON MULTIPLIER LATTICE

FIGURE 6.

Not unexpectedly, some spatial effects were observed in the reactivity analyses of these measurement data. As shown in Figure 6, analyses of the data accumulated with the detector located in the reflector region gives different reactivity values from those obtained from the data accumulated with the detector located in the center of the fuel lattice. However, as the delayed critical condition is approached, the reactivity values converge such that the data from the reflector detector gives the same value for the reactivity of the largest subcritical fuel loading as does an extrapolation of the data from the center detector. By extrapolating this subcritical data to the delayed critical condition, a delayed critical loading of 271.4 fuel rods is obtained.

By comparing the differences between the prompt critical loading and the delayed critical loading, a reactivity worth of 9.9 cents per rod can be determined for the fuel rods near the critical loading. This is equivalent to 0.00065 per rod in terms of the neutron multiplication constant (k_{eff}) if a reasonable value of 0.0065 is assumed for the effective delayed neutron fraction (β_{eff}).

Effective neutron multiplication constants (k_{eff}) have been determined from the reactivity data shown in Figure 6 by assuming a value of 0.0065 for β_{eff} . These k_{eff} values as a function of fuel rod loading are shown graphically in Figure 7. A curve is also shown in Figure 7 depicting the reactivity effect resulting from filling the lower half of the center void tube with water. This second curve was obtained by considering that filling the void tube half full of water increases the number of fuel rods required for criticality by 4.9 rods and by applying the rod worth of 0.65 milli-k uniformly over the lattice loadings shown.

k_{eff} FROM PULSED NEUTRON SOURCE MEASUREMENTS - NEUTRON MULTIPLIER LATTICE

FIGURE 7.

Neutron Multiplication and Relative Flux Inside the Central Flux Trap

A study was made of relative neutron flux levels inside the central flux trap of the multiplier using the arrangement shown in Figure 8.^(7,8) Seventeen small copper foils (3/8 in. OD, ~ 0.16 gm) were vertically inserted into the flux trap at 2 in. increments above and below the ^{252}Cf source. Axial traverse data were obtained for both bare and cadmium-covered foils with and without the fuel in place. The source was positioned alongside of the 2 in. flux trap tube at the midpoint height of the fuel region. Half of the foils were, therefore, exposed in air and half in water. From these data, the thermal neutron multiplication of the multiplier was then determined by:

$$M = \frac{(\text{bare activity} - \text{Cd activity})_{\text{fuel} + \text{source}}}{(\text{bare activity} - \text{Cd activity})_{\text{source only}}}$$

With the multiplier loaded to 264 rods and operating at a $k_{\text{eff}} = \sim 0.991$ the thermal multiplication is ~ 20 (see Table VII). The multiplier greatly enhances and broadens the axial flux profile (see Fig. 9). In the water region of the flux trap (below the source), the flux follows fairly well $\phi(z) \sim 49 \cos(0.121 z)$ (see Fig. 10). An upward flux tail is observed below the fuel, probably due to neutron reflection by the water. In the air side of the flux trap, the thermal flux can be seen to closely follow a profile of $\phi(z) \sim 49e^{-0.1z}$ between the fuel center and top. Above the fuel, the flux falls off linearly, probably due to neutron streaming from the void. The peak thermal flux occurs slightly below the center point due to the moderation and reflection by the water. The peak total flux also occurs here.

Table VIII displays a comparison of relative flux levels radially from the center point of the multiplier. One foil was placed in water ~ 1.8 in away from

FLUX TRAVERSE EXPERIMENT SETUP

FIGURE 8.

TABLE VII

RESULTS OF CENTRAL FLUX TRAP FOIL IRRADIATION ¹⁾

(0.60 Lattice Spacing)

Vert. Dist. of Foil From Center of Flux Trap (cm) +Above -Below	264 Rods+Source Bare Cu Foils Sat. Act. (cpm/gm)	264 Rods+Source 2) Cd Covered Foils Sat. Act. (cpm/gm)	Bare/Cd	264 Rods+Source Bare -Cd Sat. Act. (cpm/gm)	Source Only Bare Cu Foils Sat. Act. (cpm/gm)	Source Only Cd Covered Foils Sat. Act. (cpm/gm)	Bare/Cd	Source Only Bare -Cd Sat. Act. (cpm/gm)	Bare -Cd Bare -Cd Thermal Multiplication	264 Rods + Source Cd Only Fast Multiplication
-40.64	1910	30	63.7	1880	5					
-35.56	5383	100	53.8	5283	9					
-30.48	10992	250	44.0	10742	19					
-25.40	17361	460	37.7	16901	32					
-20.32	28396	780	36.4	27616	103					
-15.24	36654	1210	30.3	35444	329	3	110	324	109	242
-10.16	45631	1600	28.5	44031	928	11	84.4	917	48.0	145
- 5.08	51350	1710	30.0	49640	2198	31	70.9	2167	22.9	55.2
0.0	50807	1750	29.0	49057	2519	75	33.6	2444	20.1	23.3
5.08	38311	1730	22.1	36581	1965	35	56.1	1930	19.0	49.4
10.16	33082	1630	20.3	31452	1140	18	63.3	1122	28.0	90.6
15.24	27330	1350	20.2	25980	523	8	65.4	515	50.4	169
20.32	21351	980	21.8	20371	343					
25.40	15657	670	23.4	14987	185					
30.48	11530	370	31.2	11160	91					
35.56	7636	170	44.9	7466	44					
40.64	3335	80	41.7	3255	24					

¹⁾ All counts are normalized to count rates observed before foils were inserted into multiplier

²⁾ Cadmium activities were re-evaluated by curve fit through the cadmium foil data.

RELATIVE FLUX VERSUS AXIAL HEIGHT IN CENTER FLUX TRAP

FIGURE 9.

RELATIVE FLUX VERSUS AXIAL HEIGHT IN CENTER FLUX TRAP

FIGURE 10.

TABLE VIII.

RELATIVE RADIAL FLUX COMPARISON
(0.60 in. Lattice Spacing)

<u>Foil Position</u>	<u>Relative Flux Saturation Activity (cpm/gm)</u>	<u>$\frac{\phi \text{ Center}}{\phi \text{ Outside}}$</u>
1.8 in. Outside Fuel in Void	13693	3.17
Center at Water/Void Interface	43475	
1.8 in. Outside Fuel in Water	13685	3.18

the outside fuel edge and one foil was centered in a 2 in. void tube whose center point was ~ 1.8 in. away from the fuel. Another foil was, again, positioned in the center of the flux trap. The flux at the center was about 3.2 times the outside flux, and it is interesting to note, further, that the relative flux in both the void tube and water was nearly the same.

Effect of Rod Removal From the Lattice Region

For lattices of either a 0.55 or 0.60 in. spacing, the fuel assembly is undermoderated. The undermoderated lattice is required if the multiplier is to have a negative temperature coefficient. The removal of a fuel rod from the lattice creates a more moderated area within the lattice, which increases the reactivity; i.e., fuel rod removal from within the lattice region could cause the system to attain criticality. Table IX gives the increased effects of rod removal from approximately the mid-region of the loading. For the 0.55 in. lattice, a rod removed from the loaded region produces about the same effect on reactivity as a rod added to the outside of the lattice; in the 0.60 in. lattice, it would be equivalent to about a 1/2 rod addition. Obviously, care must be taken to not unload from within the lattice region.

Calculation of the Operation Loading

The mathematical determination of the loading suggested for operation is shown in Table X. The number of fuel rods (N) for $k_{\text{eff}} = 0.985$ was determined for the system as it would be used in operation; i.e., no credit was allowed for the source, added outside plates, void tubes, etc., that could be removed. This N value was 257.9 rods. Reducing the number of rods to compensate for the temperature effect and the effect of 5 g ^{235}U in the central tube gives the operation loading of 252.3. Using this loading, limits would have to be placed on materials used in the outside radiation tubes and on fuel rod removal from the lattice region.

TABLE IX.WORTH OF ROD REMOVAL FROM LATTICE

<u>Exp. No.</u>	<u>Lattice Spacing</u>	<u>Number of Inner Rods Removed</u>	<u>Worth (Rods)</u>	<u>Worth (1) (Rods/Rod)</u>
004 ⁽²⁾	0.55	6	4.30	0.72
	0.55	12	10.29	0.86
	0.55	18	16.94	0.94
	0.55	24	24.35	1.01
019 ⁽³⁾	0.60	6	3.51	0.59
	0.60	12	6.70	0.56

(1) Number of rods equivalent on outside of lattice for each removed from within lattice region.

(2) With 3 in. void tube, 55 rod equivalent flux trap; rods removed from 5th row.

(3) With 2 in. void tube - 1/2-filled with water; 37 rod equivalent flux trap; rods removed from 4th row.

TABLE X.TYPICAL CALCULATION OF AN OPERATION LOADING

(0.60 in. Lattice Spacing)

$$N_C \text{ (2 in. Void Center Tube)} = 271.4$$

$$N_C \text{ (2 in. 1/2 Filled/Water)} = 276.3$$

$$\text{Difference : } 276.3 - 271.4 = 4.9$$

$N \text{ (} k_{\text{eff}} = 0.985 \text{)}$	254.5	with 2 in. Void Center + BF ₃
	- 1.5	for BF ₃
	<hr/>	
	253.0	
	+ 4.9	for 1/2 Full/Water
	<hr/>	
	257.9	
	- 2.6	for Temperature
<hr/>		
	255.3	
	- 3.0	for Center Effect for Sample -
	<hr/>	²³⁵ U, (5.0 g max.), Graphite
$(k_{\text{eff}} = 0.981)$	252.3	

Conditions:

1. Fissile material limit in center equivalent to 5 g ²³⁵U.
2. No correction for outside tube materials
Limit ²³⁵U
Limit graphite and other low absorption scatters
and/or distance
3. Any fuel rod removal must be from the outside
of loaded region.

CONCLUSIONS

1. Outside void radiation tubes have a large effect on reduction of reactivity. They should be flooded or moved away from the assembly.
2. Materials such as graphite (low absorption scatterers) increase reactivity in both the center and outside lattice regions when replacing either water or voids.
3. The reactivity worth of a fuel rod in the outer ring of the lattice, at the operational loading, is 0.00065 in terms of k_{eff} .
4. A temperature increase above 21°C increases reactivity by the equivalent of 2.6 fuel rods maximum.
5. One-third of a fuel element in the central radiation tube increases reactivity by the equivalent of about 4.4 fuel rods.
6. The source mockup with 200 g stainless steel causes a decrease in reactivity equal to 1.9 fuel rods.
7. A large reduction in reactivity was realized by pouring the 2 in. diameter center void tube full of 6 g/liter boric acid solution. This was equivalent to about -29 fuel tubes; however, the reactivity effect would be only about one-half this much in the multiplier, since its central tube is only half-way through the lattice region.
8. The operating loading must account for possible positive reactivity effects. Though attempts were made in design to optimize effects, some still remain. Temperature effects and positive effects from materials placed into the radiation tubes must be considered. The loading of the system must be reduced so that k_{eff} will remain below the maximum allowed. The k_{eff} for the recommended loading would be 0.981, with no samples and at room temperature.

9. Other effects causing reduced reactivity will make the multiplier less efficient; these include outside void tubes near lattice, stainless steel in source construction, and high neutron absorption materials in the irradiation tubes. Sample materials which are fissile or low absorption scatterers could increase reactivity.
10. The thermal neutron flux multiplication (M as defined on page 10) with the multiplier lattice at a k_{eff} of 0.99 was found to vary from 20 (at the source position) to over 50 at a position 15 cm above the source. M is, therefore, subject to some interpretation. It might be more meaningful to consider the average flux increase (due to multiplication) over a finite volume about the source or to consider the average buildup in thermal flux over the region wherein the materials would be irradiated.
11. The flux relationship on the inside near the 3.7 μg Cf source is about 3.2 times that outside the lattice at about 1.8 in. from the outside tube.

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 3 Technical Information Files
 1 Technical Publications