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ATOMICS INTERNATIONAL <small>A Division of North American Aviation, Inc.</small>		NAA-SR TDR NO 5673	APPROVALS <i>C. H. Allen</i>
TECHNICAL DATA RECORD		PAGE 1 OF 73	
AUTHOR O. R. Hillig	DEPT & GROUP NO 791-25	DATE 12-19-60	
		GO NO 7508	
TITLE Thermal Utilization Measurements with U-10 Mo fuel in the Hallam Exponential Lattices.		S/A NO 7950	RECOMMENDED FOR OUTSIDE DISTRIBUTION ☒ YES ☐ NO <i>C. H. Allen</i> SIGNATURE
		TWR 23872	
PROGRAM HNPF	SUBACCOUNT TITLE U-Alloy Physics Studies and Experiments		
DISTRIBUTION USE AN ASTERISK (*) TO INDICATE THOSE WHO ARE TO RECEIVE COMPLETE COPIES * R. C. Allen * K. E. Buttrey * E. R. Cohen * F. L. Fillmore * R. C. Gerber * C. L. Larson * J. E. Mahlmeister * W. T. Morgan * J. L. Watts Project File * * File (3) * Exp. Physics (25)	STATEMENT OF PROBLEM <p>Some experimental data on a lattice similar to the HNPF reactor was required to reduce the uncertainty in making calculations of the lattice parameters. Also, information regarding the detailed thermal neutron flux distributions within a lattice are useful in helping to solve some of the heat transfer problems.</p>		
	ABSTRACT: <p>This report covers the measurements required to determine the values of thermal utilization for the U-10 Mo fuel elements in hexagonal lattices of three different spacings. The experiments and the results which lead to the values of thermal utilization are discussed. This includes a description of the lattice cells studied, the results, in graphical form, of the detailed neutron flux distributions in the various materials in the cells, and the values obtained.</p> <p>Epi-cadmium neutron flux distributions and cadmium ratios for each of the lattice cells studied are also included in this report.</p>		

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I INTRODUCTION

For the first fuel loading of the sodium-cooled, graphite-moderated Hallam reactor ⁽¹⁾ a uranium-molybdenum alloy is to be used.

Because the Hallam reactor core design differs substantially from other sodium graphite reactors for which experimental details are available, exponential experiments were performed to provide some firm data on which to base calculations for the reactor core. The design of the reactor includes multi-rod fuel elements loaded at the vertices of regular hexagons (hexagonal lattice). Because of the difficulty in making calculations for this type geometry, the reactor core was mocked-up as closely as practical in an exponential lattice, and the lattice parameters were measured. Measurements to determine thermal utilization were also made in assemblies where the lattice spacing was different from that of the Hallam reactor and in one lattice with fuel element of lower enrichment than that of the full fuel loading. In this report the intracell measurements required to determine thermal utilization are to be discussed. In addition to contributing to the determination of thermal utilization, these measurements can be used to aid in solving some heat transfer problems pertinent to the Hallam reactor. Additional internal reports are being issued to cover the other lattice parameters measured. ^{2,3}

II SUMMARY OF RESULTS

Three lattices were studied to determine the values of thermal utilization. The lattices were formed by fuel elements at the

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vertices of regular hexagons, the distance across the flats of the hexagons being 13-inches, 16-inches, and 19-inches. For these three lattice spacings a unit cell containing a fuel element of 19 rods, $3\frac{1}{2}$ w/o enriched U-10 w/o Mo as shown in Figure (1) was studied. Using the 16-inch hexagonal lattice, an 18 rod fuel element of the same fuel composition and a 19 rod, 2.78 w/o enriched U-10 w/o Mo fuel element were studied in a unit cell. Also, the 19 rod $3\frac{1}{2}$ w/o enriched U-10 w/o Mo fuel element was studied in the 16-inch hexagonal lattice where it was surrounded by a zirconium tube to mock-up the process tube and stainless steel to mock-up the moderator cladding present in the actual reactor. See Figure 2 for a description of this cell.

The thermal neutron flux distributions were measured within a representative unit cell of each lattice. Then by means of graphical integration, the relative average thermal neutron fluxes were determined for each fuel rod and each material within the unit cell. The values of these obtained for the fuel rods were used to determine the power disadvantage factors for each fuel element. The power disadvantage factor is defined as the average thermal neutron flux in the fuel, where the relative thermal neutron fluxes have been normalized so the average flux in a fuel rod of the outer ring of rods is equal to one. By using the relative average thermal neutron flux values obtained for each material of a cell with their corresponding macroscopic thermal neutron cross sections and volumes, the value of thermal utilization for each lattice was calculated. The values obtained for

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thermal utilization and the power disadvantage factor for each lattice studied are listed in Table I. The comparison of the power disadvantage factors for the 19-rod and the 18-rod fuel elements aided in the decision to change the design of the Hallam fuel elements from 19 rods per element to 18 rods.

TABLE I

THERMAL UTILIZATION AND POWER DISADVANTAGE FACTOR

<u>Fuel Element, Lattice</u>	<u>Thermal Utilization</u>	<u>Power Disadvantage Factor</u>
19-rod, $3\frac{1}{2}$ w/o e U-10 w/o Mo, 13-inch hexagonal	$0.919 \pm .013$	$0.850 \pm .006$
19-rod, $3\frac{1}{2}$ w/o e U-10 w/o Mo, 16-inch hexagonal	$0.885 \pm .009$	$0.840 \pm .004$
18-rod, $3\frac{1}{2}$ w/o e U-10 w/o Mo, 16-inch hexagonal	$0.884 \pm .011$	$0.868 \pm .005$
19-rod, 2.78 w/o e U-10 w/o Mo, 16-inch hexagonal	$0.876 \pm .010$	$0.856 \pm .005$
19-rod, $3\frac{1}{2}$ w/o e U-10 w/o Mo, 16-inch hexagonal with stainless steel and zirconium	$0.859 \pm .010$	$0.848 \pm .005$
19-rod, $3\frac{1}{2}$ w/o e U-10 w/o Mo, 19-inch hexagonal	$0.832 \pm .012$	$0.821 \pm .006$

III EXPERIMENTAL MEASUREMENTS

A. Description of Apparatus

A sketch of the AE-6 reactor with an exponential assembly on the reactor thermal column is shown in Figure 3. The AE-6 water boiler reactor and the vertical thermal column ⁽⁴⁾ supply the thermal neutrons to the exponential assembly. The thermal column is five feet in diameter. The thermal neutron flux at the top of

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the thermal column is about 10^5 neutrons/cm²/sec/watt. The reactor can be operated at a maximum steady state power level of two kilowatts. The exponential assemblies are contained by a cadmium sheet which serves as definite boundaries on the four sides for thermal neutrons. Graphite blocks and fuel clusters can be loaded into or removed from the assembly by use of the overhead crane.

The graphite moderator blocks in the assembly were 69-inches high. The graphite was machined to provide channels for the fuel clusters, these holes being 4.75-inches in diameter. The graphite dimensions were designed to allow stacking a basic assembly where fuel channels are at the vertices of regular hexagons, the distance across the flats of the hexagons being 13-inches. By adding spacer blocks, the basic assembly could be expanded to allow for fuel channels to be at the vertices of hexagons either 16-inches or 19-inches across the flats. The 16-inch spacing is the same as that proposed for the Hallam reactor. Figure 4 shows the top view of the 16-inch hexagonal lattice. For brevity the different lattices are to be referred to as the 13-inch, 16-inch, and 19-inch lattices throughout the remainder of this report.

In the actual Hallam reactor core ⁽¹⁾ the fuel slugs are to be contained in stainless steel tubes, and sodium coolant will flow through the fuel clusters. Also, zirconium and stainless steel are to be used for process tubes and moderator cladding respectively. Because it was not practical to mock-up these structural

materials for the exponential experiments, they were omitted from the measurements except in one case mentioned in Part II of this report. The sodium around the fuel elements was mocked-up using aluminum because the two materials have similar macroscopic thermal nuclear properties, and aluminum is a convenient metal with which to work. The fuel holders consist of 1100F aluminum cylinders, $4\frac{1}{4}$ -inches in diameter by 66-inches long with 19 holes drilled parallel to the axis of the cylinders. For details of the fuel element, see Figure 5.

The $3\frac{1}{2}$ w/o enriched U-10 w/o Mo as cast slugs were made in 12-inch lengths and 6-inch lengths. The slugs were cast to a $0.590 \pm .005$ -inch diameter specification. The average diameter of the slugs was checked as 0.592-inches. The fuel slugs were loaded into the holes of the aluminum holders to give a fuel height of $66 \pm \frac{1}{4}$ -inches and a fuel area of about 5.2 square inches. The average composition of the slugs is 10.15 w/o molybdenum and 89.85 w/o uranium with an enrichment of 3.448 w/o U^{235} . A detailed description of the fuel slugs can be obtained from Reference 5.

B. Measuring Techniques

The intracell thermal neutron flux distributions were determined by activating small detector foils. The foils are one-centimeter by two millimeters by 0.005-inch thick. Their composition is 15% by weight dysprosium oxide dispersed in aluminum. These foils were inter-calibrated by exposing them on a rotating disc to flux from the AE-6 reactor and then measuring their beta

activities. The reproducibility of the calibration of a foil was found to be within 0.15%. The foils were exposed in the graphite, aluminum and fuel in which these foils were covered with either aluminum or cadmium boxes. This was done to obtain relative total (bare or aluminum covered) and relative epithermal (cadmium covered) neutron fluxes. The thickness of the cadmium boxes was 0.020-inch.

After exposing the detector foils in the various assemblies, their relative activities were determined by means of gross beta counting with scintillation counters. These counters consist of photomultiplier tubes with 2 mm thick anthracene crystals and the other necessary instrumentation. Three counting channels were in operation where 0.1% counting statistics were obtainable.

C. Thermal Neutron Flux Distribution Measurements Within a Lattice Cell

For each of the lattices studied, thermal neutron flux distribution measurements were made within a lattice cell. These measurements were used to determine the average relative thermal neutron flux in each material. The lattice cell was taken as an equilateral triangle containing the fuel element as shown in Figure 1. Provisions were made in a lattice cell to allow for the placement of the small dysprosium oxide detector foils. Figure 6 shows a horizontal cross section of a cell with the locations provided for foil measurements. Measurements were also made in neighboring cells as a check on reproducibility. Detector foils were placed in the graphite, aluminum and fuel where the foils were exposed both bare and cadmium covered to allow for the

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determination of the relative thermal neutron flux distributions. The fuel element of interest was placed in the geometric center of the lattice in the 16-inch and 13-inch lattices. In the 19-inch lattice, the center of a hexagon formed by the fuel elements was at the geometric center of the lattice.

All measurements were made at a height of 30 inches from the base of the assembly where experiment revealed that higher harmonics were not present. The observed data were divided by a factor $\cos \nu_x X \cos \nu_y Y$, where x and y are the horizontal coordinates of the data point with reference to the lattice axis, and where ν_x and ν_y are the horizontal lattice bucklings. This correction transforms the observed flux to that which would exist in a cell situated in a lattice of infinite extent. All the data listed in this report have been corrected in this way. Graphs are included to show the results of each situation studied. The different arrangements studied were:

1. 3.448 w/o e U-10 Mo, 19 rod fuel element in the 13-inch lattice.
2. 3.448 w/o e U-10 Mo, 19 rod fuel element in the 16-inch lattice.
3. 3.448 w/o e U-10 Mo, 18 rod * fuel element in the 16-inch lattice.

* The 18-rod element is the same as the 19-rod element except that the central fuel rod has been removed, and a void is left in its place.

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4. 3.448 w/o e U-10 Mo, 19 rod fuel element in the 16-inch lattice with stainless steel and zirconium.**
5. 2.78 w/o e U-10 w/o Mo, 19 rod fuel element in the 16-inch lattice.
6. 3.448 w/o e U-10 w/o Mo, 19 rod fuel element in the 19-inch lattice.

For all the arrangements listed above, the remainder of the lattice was made up of the 19 rod, 3.448 w/o e U-10 w/o Mo type of fuel elements.

In the fuel, measurements were made across the diameters of the fuel rods. Figure 7 shows two slotted fuel slugs with inserts for cadmium covered and bare foil measurements. The bare foil measurements were made with 0.0003 inch aluminum wrappings to eliminate any possible fission products which might contribute to the counting rate. The cadmium covered foils were contained in 0.020 inch thick cadmium boxes. For the central rod where the distribution was symmetrical about the rod axis within the accuracy of the measurements, flux measurements along one fuel rod diameter sufficed. For the inner and outer rings of fuel rods where the distributions were not axially symmetrical, a two-dimensional survey of the flux in these rods was obtained by making diametrical flux traverses at angles of 0°, 45°, and 90° to a line through the center of the fuel rod to the center of the

** Figure 2 gives a detailed description of the lattice cell with stainless steel and zirconium present.

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cell. The locations of these measurements can be seen in Figure 6. By studying the four fuel rods marked 1, 2, 8, and 9 in the Figure, the average fluxes for the 19 rods of the element are obtained by consideration of the symmetry present. From the data shown in Figures 8 to 29, the average relative thermal neutron flux in the various fuel rods for the different situations studied was determined by graphical integrations.

Several of the aluminum fuel holders were designed so they could be split at the 21-inch and 30-inch levels to allow for the removal of a 6-inch section which contained the detector foils for the neutron distribution measurements. This 6-inch section of aluminum fuel holder can be seen in Figures 7 and 30. Figure 30 shows the aluminum fuel holder with the provision for determining the neutron flux distribution within the aluminum. Measurements were also made at the surfaces between the aluminum and the fuel rods. The holes shown in the Figure were either filled with aluminum plugs containing detector foils or with solid aluminum plugs. The solid plugs were used when measurements were not required in that material. By measuring the thermal neutron flux distributions in a 30° section of the aluminum fuel holder, the thermal neutron flux in the entire aluminum holder was represented through symmetry. The measured value of the relative thermal neutron flux was plotted for each measured point in the aluminum to make a contour mapping of the relative fluxes in the aluminum. Figures 31 to 35 show the results of this work. By graphically integrating the small areas, using a compensating polar planimeter, and finding the average

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relative fluxes, the average relative thermal neutron flux in the aluminum was determined.

To obtain the relative average thermal neutron flux in the graphite, neutron flux distribution measurements were made in the locations shown in Figure 6. Holes, 5/8-inch in diameter, were drilled in the graphite in the lattice cell being studied to accept the graphite foil holders. The graphite foil holders had provisions for foils to be placed at one-centimeter intervals. Some of the many graphite foil holders used for the measurements can be seen in Figure 36. The equilateral triangle which makes up a lattice cell can be divided into 6 symmetrical 30° - 60° - 90° triangles with the 60° vertex in the center of the cell. By determining the average relative thermal neutron flux in one of these right triangles, the average flux in the graphite is known. To determine the average flux in the 30° - 60° - 90° triangle, flux plots were made at 15° intervals through the 60° angle. See Figure 6. Graphs showing these plots for each of the situations studied can be seen in Figures 37 to 41 inclusive. From these plots the average relative thermal neutron flux in the graphite was determined by graphical integration.

For the one case where the fuel element was surrounded by stainless steel and zirconium, Figure 2, the relative average thermal neutron fluxes for these materials were determined by placing detector foils along their surfaces and averaging the measured values obtained.

Table II gives a summary of the relative average thermal neutron flux determined for each material of each cell studied. For the case of the 18-rod fuel element in the 16-inch lattice, a minimum number of measurements were performed in the aluminum and the graphite. A few were made, and it was found that the values obtained agreed with the 19-rod, 16-inch lattice work so the results from that series of measurements were used for the 18-rod work. Also, because in the other measurements, the average relative thermal neutron flux for rods 8 and 9 were equal within experimental uncertainty. Measurements were made mostly in rod 8 for the 18-rod measurements. The results from rod 8 were used for rod 9.

IV CALCULATION OF THERMAL UTILIZATION

In addition to the relative average thermal neutron flux for each material, the volumes and macroscopic thermal neutron absorption cross sections are required for each material to determine thermal utilization from the equation:

$$f \text{ (thermal utilization)} = \frac{\bar{\phi}_{\text{fuel}} V_{\text{fuel}} \Sigma_{a,\text{fuel}}}{\sum_i \bar{\phi}_i V_i \Sigma_{a,i}}$$

where $\bar{\phi}$ is the average flux, V is the volume fraction, and Σ_a is the macroscopic absorption cross section. The average fluxes are listed in Table II. Table III gives the fuel composition.

TABLE II
RELATIVE AVERAGE THERMAL NEUTRON FLUX

CELL STUDIED	FUEL					GRAPHITE	ALUMINUM	SS	Zr
	Rod 1	Rod 2	Rod 8	Rod 9	Avg				
3.448 w/o e U-10 w/o Mo, 19 rod fuel element in the 13-inch lattice	0.510 ±.003	0.606 ±.016	0.991 ±.012	1.009 ±.009	0.850 ±.006	3.596 ±.052	1.467 ±.012		
3.448 w/o e U-10 w/o Mo, 19 rod fuel element in the 16-inch lattice	0.464 ±.003	0.584 ±.012	1.000 ±.008	1.000 ±.007	0.840 ±.004	4.670 ±.060	1.516 ±.009		
3.448 w/o e U-10 w/o Mo, 18 rod fuel element in the 16-inch lattice	Void ±	0.605 ±.009	1.000 ±.010	1.000 ±.010	0.868 ±.005	4.609 ±.060	1.496 ±.012		
3.448 w/o e U-10 w/o Mo, 19 rod fuel element in the 16-inch lattice with stainless steel & zirconium	0.475 ±.004	0.608 ±.006	1.000 ±.011	1.000 ±.011	0.848 ±.005	4.833 ±.072	1.399 ±.012	3.381 ±.039	2.265 ±.014
2.78 w/o e U-10 w/o Mo, 19 rod fuel element in the 16-inch lattice	0.522 ±.004	0.625 ±.010	0.994 ±.009	1.006 ±.010	0.856 ±.005	4.113 ±.052	1.438 ±.010		
3.448 w/o e U-10 w/o Mo, 19 rod fuel element in the 19-inch lattice	0.441 ±.005	0.525 ±.012	1.000 ±.012	1.000 ±.013	0.821 ±.006	5.653 ±.076	1.651 ±.011		

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TABLE III
FUEL COMPOSITION

<u>FUEL</u>	<u>w/o U²³⁸</u>	<u>w/o U²³⁵</u>	<u>w/o Mo</u>	<u>ρ density gm/cm³</u>
3½ w/o	86.75	3.10	10.15	17.12
2.78 w/o	87.35	2.50	10.15	17.12

The aluminum density is taken as 2.67 gm/cm³. The densities and macroscopic thermal neutron absorption cross sections were determined for the graphite in each assembly by weighing and from thermal neutron diffusion length measurements respectively. These results are discussed in another report, Reference 2. The volumes and macroscopic cross sections used for the different materials in each assembly are listed in Table 4. The results of the calculation of f are listed in Table 1.

In order to determine the relative average thermal neutron flux distributions, relative total and epi-cadmium flux distributions were measured. Plots of the relative epi-cadmium neutron flux and the cadmium ratios along the cell diagonals are also included in this report. See Figures 42 - 47. Figures 48 - 53 show the relative thermal neutron flux distributions along the cell diagonal. These are the results using dysprosium detector foils.

These results are being used as a basis for checking calculational models used to predict lattice parameters. They were useful in establishing a fuel element design for the Hallam reactor. The

TABLE IV

6 VOLUMES AND MACROSCOPIC THERMAL NEUTRON CROSS SECTIONS

CELL STUDIED	VOLUME (cm ³)					Σ_A (cm ⁻¹)				
	FUEL	GRAPHITE	ALUMINUM	SS	Zr	FUEL	GRAPHITE	ALUMINUM	SS	Zr
3.448 w/o E U-10 w/o Mo, 19 ROD FUEL ELEMENT IN THE 13-INCH LATTICE	33.74 ± 0.13	357.80 ± 0.29	78.49 ± 0.60			.9245 $\pm .0065$	.5127x10 ⁻³ $\pm .0302 \times 10^{-3}$	0.01464 ± 0.00035		
3.448 w/o E U-10 w/o Mo, 19 ROD FUEL ELEMENT IN THE 16-INCH LATTICE	33.74 ± 0.13	600.84 ± 0.29	78.49 ± 0.06			.9245 $\pm .0065$	0.5997x10 ⁻³ $\pm .0299 \times 10^{-3}$	0.01461 ± 0.00035		
3.448 w/o E U-10 w/o Mo, 18 ROD FUEL ELEMENT IN THE 16-INCH LATTICE	31.96 ± 0.12	600.84 ± 0.29	78.49 ± 0.60			.9245 $\pm .0065$	0.5997x10 ⁻³ $\pm .0299 \times 10^{-3}$	0.01461 ± 0.00035		
3.448 w/o E U-10 w/o Mo, 19 ROD FUEL ELEMENT IN THE 16-INCH LATTICE WITH STAINLESS STEEL AND ZIRCONIUM	33.74 ± 0.13	600.84 ± 0.29	67.25 ± 0.56	1.390 ± 0.20	9.404 $\pm .094$	.9245 $\pm .0065$	0.5997x10 ⁻³ $\pm .0299 \times 10^{-3}$	0.01461 ± 0.00035	.230 $\pm .006$	8.154x10 ⁻³ $\pm .179 \times 10^{-3}$
2.78 w/o E U-10 w/o Mo, 19 ROD FUEL ELEMENT IN THE 16-INCH LATTICE	33.74 $\pm .013$	600.84 ± 0.29	78.49 ± 0.60			0.7679 $\pm .0056$	0.5997x10 ⁻³ $\pm .0299 \times 10^{-3}$	0.01461 ± 0.00035		
3.448 w/o E U-10 w/o Mo, 19 ROD FUEL ELEMENT IN THE 19-INCH LATTICE	33.74 ± 0.13	894.21 ± 0.29	78.49 ± 0.60			.9245 $\pm .0065$	0.6494x10 ⁻³ $\pm .0377 \times 10^{-3}$	0.01461 ± 0.00035		

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18 rod fuel element was chosen in preference to a 19 rod fuel element after examining the results of these experiments. It is felt that these results may be of further use in future calculations concerning sodium graphite reactors.

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- II. Relative Average Thermal Neutron Flux
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Figure 1 Unit Cell in the Hexagonal Lattices

Figure 2 Unit Cell in the 16-inch Hexagonal Lattice with Stainless Steel and Zirconium Present

FIGURE 3. CUTAWAY VIEW OF EXPONENTIAL ASSEMBLY
AND WATER BOILER REACTOR

Locations marked A and B
show positions of voided channels

FIGURE 4. LOCATION OF VOIDED FUEL CHANNELS

Fuel Composition	86.75 w/o U ²³⁸ 3.10 w/o U ²³⁵ 10.15 w/o Mo	a	fuel rod - 0.592 inch diameter
Fuel Density	17.12 gm/cm ³	b	aluminum hole - 0.606 inch diameter
Aluminum Density	2.67 gm/cm ³	c	inner ring location - 1.654 inches diameter
Graphite Density	1.685 gm/cm ³	d	outer ring location - 3.320 inches diameter
		e	aluminum cluster - 4.742 inches diameter
		f	graphite moderator fuel channel hole - 4.755 inches diameter

Figure 5. Cross Sectional View of Fuel Cluster with Dimensions

Figure 6 Foil Locations Available within a Unit Cell

Figure 7 Fuel Slugs with Inserts for Detector Foils and 6-inch Aluminum Fuel Holder Section

KE 10 X 10 TO THE CM. 359-14
KEUFEL & ESSER CO. WA, P. M. S. A.

Relative Thermal Neutron Flux

Figure 8 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 13-inch Lattice

KE 10 X 10 TO THE CM. 359-14
 KEUFFEL & ESSER CO. MADE IN U.S.A.
 Relative Thermal Neutron Flux

Figure 9 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 13-inch Lattice

K&E 10X10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

Figure 10 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 13-inch Lattice

K&E
10 X 10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

Figure 11 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 13-inch Lattice

K&E 10X10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

Figure 12 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

K-E 10 X 10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADISON, U.S.A.

Figure 13 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

K&E
10X10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

Figure 14 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

KE 10 X 10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

11.8
11.4
11.0
10.6
10.2
9.8
9.4
9.0
8.6
8.2
7.8
7.4
7.0
6.6
6.2
5.8
5.4
5.0
4.6
4.2
3.8
3.4
3.0

Figure 15 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

Figure 16 Thermal Neutron Flux Distributions in Fuel Rods For the 16 rod 3½ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

10 X 10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

12.0
11.0
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0

-0.8 -0.7 -0.6 -0.5 -0.4 -0.3 -0.2 -0.1 0 .1 .2 .3 .4 .5 .6 .7 .8
Distance From Center of Fuel Rod (cm.)

Figure 17 Thermal Neutron Flux Distributions in Fuel Rods for the 18 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

8x — ○
8y — x
8z — △

Aluminum

Fuel

Aluminum

10 X 10 TO THE CM. 359.14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

Figure 18 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod 3/2 w/o Enriched 0-10 w/o Mo Fuel Element in the 16-inch Lattice with Stainless Steel and Zirconium Present

10 X 10 TO THE CM. 359-14
KUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

Figure 19 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice with Stainless Steel and Zirconium Present

K-E 10 X 10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

Figure 20 Thermal Neutron Flux Distributions in Fuel Rods for the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice with Stainless Steel and Zirconium Present

10 X 10 TO THE CM. 359.14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

Figure 21 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice with Stainless Steel and Zirconium Present

Figure 22 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod 2.78 w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

K-E 10X10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

6.0
5.6
5.2
4.8
4.4
4.0
3.6
3.2

Figure 23 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod
2.78 w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

K&E 10 X 10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

12.5
11.5
10.5
9.5
8.5
7.5
6.5
5.5
4.5

Distance From Center of Fuel Rod (cm.)

Figure 24 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod
2.78 w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

K&E 10X10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

12.5

11.5

10.5

9.5

8.5

7.5

6.5

5.5

4.5

-0.8 -0.7 -0.6 -0.5 -0.4 -0.3 -0.2 -0.1 0 .1 .2 .3 .4 .5 .6 .7 .8
Distance From Center of Fuel Rod (cm.)

Figure 25 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod
2.78 w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

K&E 10X10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Relative Thermal Neutron Flux

4.8
4.6
4.4
4.2
4.0
3.8
3.6
3.4
3.2

Distance From Center of Fuel Rod (cm.)

Figure 26 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod
3 1/2 w/o Enriched U-10 w/o Mo Fuel Element in the 19-inch Lattice

K&E 10X10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Figure 27 Thermal Neutron Flux Distributions in Fuel Rods for the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 19-inch Lattice

K&E 10 X 10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.

Figure 28 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 19-inch Lattice

K&E 10 X 10 TO THE CM. 359-14
KEUFFEL & ESSER CO. MADE IN U.S.A.
Relative Thermal Neutron Flux

Figure 29 Thermal Neutron Flux Distributions in Fuel Rods For the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 19-inch Lattice

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Figure 30 6-inch Section of Aluminum Fuel Holder with Provisions for Measurements in the Aluminum

Figure 31 Section of Aluminum Fuel Holder Showing Relative Average Thermal Neutron Fluxes
19 Rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 13-inch Hexagonal Lattice

Figure 32 Section of Aluminum Fuel Holder Showing Relative Average Thermal Neutron Fluxes
19 Rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Hexagonal Lattice

Figure 33 Section of Aluminum Fuel Holder Showing Relative Average Thermal Neutron Fluxes
19 Rod 3½ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Hexagonal Lattice with Stainless Steel and Zirconium

Figure 35 Section of Aluminum Fuel Holder Showing Relative Average Thermal Neutron Fluxes
19 Rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 19 inch Hexagonal Lattice

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Figure 36 Graphite Foil Holders

Relative Thermal Neutron Flux

499

82.5

Figure 37 Thermal Neutron Flux Distributions in the Graphite for the 19-rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 13-inch Lattice

Figure 38 Thermal Neutron Flux Distributions in the Graphite for the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

09 499

Figure 39 Thermal Neutron Flux Distributions in the Graphite for the 19-rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel element in the 16-inch Lattice with Stainless and Zirconium

Figure 40 Thermal Neutron Flux Distributions in the Graphite for the 19-rod 2.78 w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Lattice

Relative Thermal Neutron Flux

62 499 664

Figure 41 Thermal Neutron Flux Distributions in the Graphite for the 19 rod $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 19-inch Lattice

Figure 42 Epithermal Neutron Flux Distribution and Cadmium Ratio For The 19 rod, 3 1/2 w/o Enriched U-10 w/o Mo Fuel Element in the 13-inch Hexagonal Lattice

Figure 43 Epithermal Neutron Flux Distribution and Cadmium Ratio For The 19 rod, 5/2 w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Hexagonal lattice

Figure 44 Epithermal Neutron Flux Distribution and Cadmium Ratio For The 18 rod, $3\frac{1}{2}$ w/o Enriched U-10 w/o

Mo Fuel Element in the 16-inch Hexagonal Lattice

Figure 45 Epithermal Neutron Flux and Cadmium Ratio For the 19 rod, $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo fuel Element in the 16-inch Hexagonal Lattice with Stainless and Zirconium

664
499
67

Figure 46 Epithermal Neutron Flux and Cadmium Ratio For The 19 rod, 2.78 w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Hexagonal Lattice

Figure 47 Epithermal Neutron Flux Distribution and Cadmium Ratio For The 19-inch, 5/8 w/o Enriched U-10 w/o

Mo Fuel Element in the 19-inch Hexagonal Lattice

Figure 48

Thermal Neutron Flux Distribution For The 19 rod, 3 1/2 w/o Enriched U-10 w/o Mo Fuel Element in the 13 inch Hexagonal Lattice

Relative Thermal Neutron Flux

664
70

Figure 49 Thermal Neutron Flux Distribution For The 19 Rod, 3 1/2 w/o Enriched U-10 w/o Mo Fuel Element in the 16-inch Hexagonal Lattice

64
71
499

Figure 50

Thermal Neutron Flux Distribution For The 18 rod, $3\frac{1}{2}$ w/o Enriched U-10 w/o Mo Fuel Element in the 16 inch Hexagonal Lattice

Relative Thermal Neutron Flux

Figure 51

Thermal Neutron Flux Distribution For The 19 rod, 3 1/2 w/o Enriched
U-10 w/o Mo Fuel Element in the 16 inch Hexagonal Lattice with Stainless Steel & Zirconium

Relative Thermal Neutron Flux

699
74
73

Figure 52 Thermal Neutron Flux Distribution For The 19 rod, 2.778 w/o Enriched U-10 w/o Mo Fuel Element in the 16 inch Hexagonal Lattice