TWO-DIMENSIONAL (r,z) MULTIGROUP CALCULATION

FOR AN

INTERMEDIATE SPECTRUM CRITICAL ASSEMBLY

By

P. L. Hofmann

M. L. Storm

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

KNOLLS ATOMIC POWER LABORATORY *

SCHENECTADY, NEW YORK

ABSTRACT

The results of two-dimensional (r,z) multigroup calculations of an experimental low-power, reactor (PPA-19) are described. Comparison is made between calculated results and experimental measurements of reactivity, power distribution, and sodium activation. Generally good agreement is observed between calculation and experiment. This calculation was performed in 1954 and represents the first application of Roe's two-dimensional multigroup formulation to the analysis of an operating critical assembly. The over-all objective of such calculations is to ascertain the accuracy of two-dimensional multigroup methods in order to facilitate their application to future problems in reactor design.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Many important nuclear characteristics required for the design of a large class of non-thermal, non-hydrogenous reactors can be obtained from the solution of the multigroup formulation of the age-diffusion equation. This multigroup formulation has been described in detail by H. Hurwitz, Jr. and R. Ehrlich.⁽¹⁾⁽²⁾⁽³⁾ In addition comparison of theory and experiment, using equivalent, spherical reactor models, are discussed in references (2) and (3). The extension of multigroup methods to cylindrical models (r, z) was carried out by G. M. Roe in 1954⁽⁴⁾ and further extended to ($r-\theta$) coordinates by E. L. Wachspress⁽⁵⁾.

In this article, the first application of Roe's ($r-z$) multigroup formulation to the analysis of an operating critical assembly ("PPA-19") is presented. The calculations were performed on the David Taylor Model Basin UNIVAC under the supervision of Dr. R. H. Stark, E. L. Wachspress and D. V. Steward.

The Preliminary Pile Assembly (PPA) is a flexible critical facility used at KAPL for the investigation of nuclear characteristics of a large number of proposed reactor designs. A description of the structure, control instrumentation and early experimental programs has been given by Snyder⁽⁶⁾. Figure 1 shows a photograph of the Preliminary Pile Assembly with the halves separated to insure subcriticality during modification of the assembly. The matrix of the PPA consists of stainless steel tubes which contain the materials making up the reactor composition.

PPA-19 is one of the series of beryllium moderated and reflected, intermediate spectrum reactors* investigated in this facility

A schematic diagram of PPA-19 listing longitudinal dimensions and molar composition of the core, transition region between core and end reflectors, end reflectors and end blanket are shown in Figure 2. In addition, since PPA-19 simulated a reflector controlled reactor, the side reflector composition is given for the stationary side reflectors and the six rotating control cylinders, exclusive of the reflector poison region. Figure 3 shows a schematic diagram of the PPA-19 equatorial plane. Numerical designation of pertinent matrix tubes is also shown on Figure 3.

A suitable reactor model for an $(r-z)$ calculation is obviously one in which angular variations are either nonexistent or relatively insignificant so that they can be ignored. In the PPA-19 the presence of the control drums, and in particular the presence of the B_4C in the drums, introduces the most marked angular asymmetries. These asymmetries are minimized, as far as the core region is concerned, when the B_4C region is rotated away from the core-reflector interface. This means that the (r,z) formulation is most applicable when the control drums are in the 180° setting. However, it is still possible to apply the (r,z) formulation and derive useful information which can be compared with experimental data obtained when the control drums are in the 106° position (see Figure 3).

* The calculated central median fission energy is about 40 e.v.

The following experimental information is available for comparison with the results of the (r,z) calculation.

1. Reactivity and control data (reactivity as a function of control drum position).
2. Power distribution measurements
3. Measurements of sodium activation

It will be seen that good agreement between theory and experiment can be obtained.

REACTOR MODEL

The longitudinal dimensions and molar compositions given in Figure 2 were used to construct a reactor model of simple geometry which still retained the pertinent nuclear features of the PPA-19. This model is shown in Figure 4 where the plane $z = 0$ is the reactor equatorial plane. The distances $N_j \Delta r$ ($j=1,2,3$) and $M_k \Delta z$ ($k=1,2,3,4$) locate successive interfaces in the r and z directions.

The core region was represented by a cylinder whose volume was obtained from the component molar compositions listed in Figure 2, using room temperature densities. The height of the core was taken to be that of the PPA-19 core, thus determining the core radius of the model. The end reflectors and transition regions for both ends were averaged to obtain one typical top (and bottom) reflector region. The top (and bottom) reflector radii were set equal to the core radius, and the height of the top (and bottom) reflector regions was then determined from the volume of this average region (again found from the given molar composition).

To eliminate angular dependence in the model, it was assumed that the drums are set in the 180° position and that in this setting the influence of the B_4C on the core can be ignored. The result is a side reflector region which combines the non- B_4C region of the drums and the stationary reflectors.

The height of the side reflector was taken to be an average of the drum and stationary reflector heights. The inner radius is the core radius, and the outer radius was again determined from the side reflector volume.

All remaining regions (inclusive of the volume occupied by the B_4C) were included in the blanket region which was then distributed so that the over-all reactor model was of cylindrical form. The thickness of the blanket above the end reflector was fixed at 6.5 in. (Figure 2), thus determining the outer radius of the blanket region.

The results of these calculations are summarized in Tables 1 and 2, in which the dimensions and material volume fractions for each region are listed.

After this model had been constructed, it was necessary to choose mesh spacings Δr and Δz for use in the numerical calculations.

Experience with previous (r,z) calculations at KAPL indicated that results obtained with an intermediate mesh spacing ($\Delta r \sim \Delta z \sim 2$ cm) were satisfactory from the standpoints of accuracy and calculation time expenditure. In addition, it had been found useful to improve the original source estimate by first performing the multigroup calculations with a coarse mesh spacing ($\Delta r \sim \Delta z \sim 6$ cm) before going to the intermediate-size mesh.

Thus both coarse and intermediate mesh spacings were chosen, together with values for N_j and M_k (Figure 4), subject to the restriction that the core dimensions (Table 1) remain unchanged. However, once Δr , Δz , and the core volume were fixed, it was necessary to make minor dimensional adjustments in the reflector and blanket regions so that all interfaces coincided with integral values of N_j and M_k .*

* The UNIVAC coding employed did not permit variable mesh spacing.

Table 3 lists the values of Δr , Δz , N_j and M_k selected for the coarse and intermediate mesh calculations.

Two calculations were made using the coarse mesh, and two were made using the intermediate mesh spacing. An initial source distribution was chosen for the first coarse mesh calculation, and the resulting power distribution (properly normalized) was used as the source for the second coarse mesh calculation. This process was then continued for the intermediate mesh calculations.

Although the PPA-19 core is heterogeneous, the reactor model assumes homogeneity. Thus, some correction should be made for the self-shielding of the heavy absorbers such as fuel and cadmium. To this end, self-shielding factors similar to those discussed by Hurwitz and Roe ⁽⁷⁾⁽⁸⁾ were employed for all core material but cadmium. A separate cadmium self-shielding factor was calculated to account for the fact that the cadmium was inserted in the form of thin strips (cross-sectional area 0.0018 in.²) in the interstitial spaces, and not in intimate contact with fuel. In this calculation the cadmium strips were converted into infinite cylinders, and self-absorption factors, as in reference (8) were computed.

As discussed in reference (1) the average neutron degradation in the i^{th} lethargy group ($\bar{q}$), the degradation into the i^{th} group from the previous group (q^I), and the degradation out of the i^{th} group (q^O) are assumed to be related by the following expression.

$$\bar{q} = \omega_1 q^I + \omega_2 q^O$$

where ω_1 and ω_2 are constants. Two schemes may be employed in choosing these constants. In scheme A, $\omega_1 = 0$ and $\omega_2 = f$, where f is determined by trial and error. In scheme B, $\omega_1 = \omega_2 = 1/2$.

In these calculations scheme B was employed for groups 1 through 10 ($u = 0$ to $u = 17$)*. Scheme A was used in groups 11 through 13 ($u = 17$ to $u = 19.8$). The values chosen for f are listed in Table 4.

This choice of schemes and f 's was based on experience with spherical multigroup calculations for similar reactors. Examination of the results of the cylindrical calculation shows that the above choice was a reasonable one.

All macroscopic input data (including self-shielding corrections) used in these calculations are tabulated in Appendix I. The microscopic data is obtainable from reference (9).

DISCUSSION OF RESULTS

The main purpose of this section is to present in detail the results of the second intermediate mesh calculation and compare them with the results of experiment.

It is of interest, however, first to compare some of the results obtained from all calculations. As mentioned previously, the main objective of a coarse mesh calculation is the improvement of the initial source estimate. To ascertain the effect of mesh size on the flux and power distributions, the following criteria are useful. The error in power** defined as

$$\sum_{\substack{n,m \\ \text{core}}} \left| \begin{array}{c} \text{initial} \\ P_{n,m} \end{array} - \begin{array}{c} \text{final} \\ P_{n,m} \end{array} \right|$$

* Lethargy $\mu = \ln(10^7/E)$, E in electron-volts

** This criterion is employed by Dr. R. H. Stark in the UNIVAC coding.

is divided by the total number of mesh points in the core to yield an error in power per mesh point (E_p). Similarly the error in flux for group i, and j iterations* defined as

$$\sum_{\substack{n,m \\ \text{all regions}}} \left| (\phi_i)_{n,m}^{j-1} - (\phi_i)_{n,m}^j \right|$$

can be averaged over all groups and divided by the total number of mesh points to yield an error in flux per mesh point per group (E_F). These quantities (E_p , E_F) together with the eigenvalue (λ_c) and the approximate calculation time are listed in Table 5 for all coarse and intermediate runs.

Comparison of E_p and E_F for the second coarse mesh and first intermediate mesh calculations indicates that further iteration in the coarse mesh would not result in significant source and flux improvement for use in the first intermediate run. It is also seen that an appreciable reduction in error is achieved between the first coarse and second intermediate calculations.

As can be seen in Table 5, the eigenvalue (λ_c) is a function of mesh size. The value desired is that corresponding to the limiting case of "zero mesh area." An estimate of this value can be obtained by graphically extrapolating the available data to "zero mesh area." Such a procedure indicates that the eigenvalue for the last intermediate mesh calculation might differ from the extrapolated value by as much as 0.25%.

* This criterion is employed by Dr. R. H. Stark in the UNIVAC coding.

(a) Reactivity

The multigroup calculations were formulated for a reactor model with a clean reflector and with a core composition corresponding to an experimental loading that was critical with all control cylinders at the 106° setting. The model reflector configuration represents the 180° drum setting where it is assumed that the reflector B_4C is so far removed from the core that its effect can be neglected. The eigenvalue ν_c of the second intermediate mesh calculation (Table 5) was 2.441, which is equivalent to an excess reactivity of \$.92.*

This excess reactivity represents the reactivity change which would be brought about by a rotation of all drums from the 106° to 180° setting, no change being made in the core composition.

Available experimental information includes a control cylinder calibration curve for PPA-19** extending from the 6° to the 146° settings. The portion of this curve which is applicable here extends from the 106° to the 146° settings. The calibration curve in this range was obtained in two steps in the following manner:

The drums were first rotated from the 106 to the 126° position, sufficient B_4C being added to the core to maintain criticality. Representing the reactivity $R(\theta, Y)$ as a function of drum rotation θ and B_4C core poison content Y , this rotation and addition of poison to maintain criticality can be represented in the (Y, θ) plane (Figure 5) as the curve (AB)***. Similarly the drums were rotated from the 126

* Using $\beta_1 = 0.0085$ in k , and $\nu = 2.46$ neutrons/fission

** The experimental results were obtained by J. A. Bistline

*** This approach has been formulated by H. B. Stewart and L. Tonks.

to 146° settings, and this rotation was compensated by the further addition of B_4C to maintain criticality. This rotation and poison addition is represented by curve (BC) in Figure 5.

To project point B onto the control cylinder calibration curve, an average B_4C reactivity coefficient over the range AB was employed to yield point B' in the (R, θ) plane. A similar procedure was followed for projecting point C into a point C' lying in a plane parallel to the (R, θ) plane. The lines (B'B'') and (C'C'') represent reactivities of 2.2 and 1 dollars, respectively.

The over-all control curve was constructed by piecing together curves AB' and BC' resulting in curve (AB'D). This curve is the portion of the control cylinder calibration curve between the 106° and 146° settings, taking the 106° setting as the origin. The dashed portion DF (Figure 5) represents an extrapolation of the experimental control calibration curve to the 180° setting. This extrapolation yields a value of about \$3.9 for the control margin between the 106° and 180° settings. This value and the calculated value of \$.92 can be compared if it is assumed that at these high angle drum settings the effects of the drum rotations on B_4C reactivity coefficients are not too pronounced. It should be noted that the difference of \$3 between calculated and experimental values amounts to a difference of about 2.5% in k or about 4 Kg in the reactor loading (using experimental fuel reactivity coefficients).

(b) Flux Distributions

In Figures 6 to 19, the relative average flux* in each lethargy group is shown as a function of (r,z). The range in lethargy for each group is indicated on each figure. The relative total flux of all groups is plotted in Figure 20.

*Normalization: 1 source neutron per second per cc of core. Outer Boundary Condition: Flux is zero at edge of blanket.

It is to be noted that a reasonably spherical flux pattern prevails in the core for the first seven lethargy groups (Figures 6 to 12). The effect of the reflector on the core flux pattern becomes apparent in Group 8 (Figure 13). A pronounced minimum flux region appears in Group 10 in the portion of the core furthest removed from the center. This minimum moves toward the center of the core in succeeding groups. At the same time, large flux maxima build up in the side and end reflector regions near the core-reflector interface.

In the (r, θ) formulation of the multigroup procedure, ⁽⁵⁾ the neutron flux is represented by the relation

$$\phi(r, \theta, z) = \phi(r, \theta) \cos \frac{\pi z}{H}$$

in all groups and all regions. H is the height of the equivalent reactor which is unreflected at the ends. Hence, it is of interest to examine the longitudinal flux variations at several radial positions in core, reflector, and blanket as obtained from Figures 6 to 19.

It is found that the longitudinal flux variation can be fitted by a cosine curve over an axial distance from the equatorial plane approximately equal to one-quarter of the core height. This cosine fit is about the same for all energy groups and all regions, substantiating the concept of an equivalent core height. These equivalent core heights have been calculated for radial positions $n = 0, 12, 24$ and $36 \Delta r$. The calculations are based on a cosine fit at $6 \Delta z$, and the results are listed in Table 6.

460-11

(c) Power Distribution

The power distribution is obtained from the summation

$$\sum_1 \phi_1(r, z, u_1) \sigma_{f1} \Delta u_1$$

where ϕ_1 is the relative average flux and σ_{f1} the average macroscopic fission cross section for group 1. The result of this calculation is shown in Figure 21.

Available experimental information* consists of a number of radial and longitudinal power traverses. Although more accurate radial power distributions can be obtained by an (r, θ) formulation of the multigroup method, it is of interest to compare the calculated radial, as well as longitudinal power traverses with experimental data.

Four radial traverses are available. Referring to Figure 3 (a cross section of the PPA at the equatorial plane), it is seen that these radial traverses extend from the central fuel slug (91) to peripheral slugs 172, 151, 138, and 97, respectively. In Figure 22 the power distributions measured along these traverses are plotted against a radial distance scale which has been normalized to unity at the core-reflector interface for each traverse. In addition, the power has been normalized to unity at the core center.

The calculated radial power distribution similarly normalized in the equatorial plane (obtained from Figure 21) is also shown in Figure 22. Referring to Figure 3, it is seen that the best agreement between calculation and experiment can be expected for the 91-97 traverse. Fuel slugs 138 and 151 are comparatively close to the B_4C region. Hence,

*This data was obtained by R. Arthur and G. B. Gavin.

the interface values of the corresponding radial power traverses are lower than the calculated value. On the other hand, slug 172 is located in a corner between two beryllium reflector surfaces remote from the B_4C . Thus the interface value of the 91-172 radial power traverse is higher than the calculated value.

Since this calculation ignores the presence of B_4C in the side reflector, it is reasonable to compare calculated and experimental longitudinal traverses at points farthest removed from the influence of the B_4C . This is certainly the case for the central slug 91. For slugs near the core-reflector interface, slug 171 is best suited for a comparison between calculation and experiment.

Measured values for longitudinal traverses represent the power averaged over a slug cross section. This cross section is circular with a diameter of about two inches. Hence in comparing longitudinal power traverses the multigroup results were appropriately averaged. In Figures 23 and 24, calculated and experimental longitudinal traverses at slugs 91 and 171 are compared. It is seen that the agreement is reasonable.

(d) Sodium Activation

The determination of the gamma activity (1.38 and 2.76 mev) associated with the decay of $Na-24$ is of major importance in shield design. Hence experiments were performed in the PPA-19 to obtain sodium activation data*. Since the measured activity was corrected to the saturated condition, comparable information can be obtained from the results of the multigroup calculation. In particular the saturated activity is proportional to

*This work was done by G. Gavin and E. Chuboff

$$\sum_i \phi_i(n, z, u_1) (\sigma_a)_{Na}^{u_1}$$

The results of this calculation are shown as a function of (r, z) in Figure 25. The conversion factor from relative activation to disintegration /sec/mol/watt is 15.16×10^4 .

The experimental data have been averaged over the various reactor regions and the results are listed in Table 7. Calculated values for comparable regions (obtained by appropriate averaging of data in Figure 25) are listed in the same table.

Comparison of experimental and calculated values indicates excellent agreement for the core region, while the agreement in the other regions is not as good. This reflects the more accurate representation of the PPA core in the multigroup model. However, for shielding purposes, the agreement between calculated and experimental values is very satisfactory.

INPUT DATA FOR MULTIGROUP CALCULATION*

		Group	Δu	$\left(\frac{\Delta u}{\xi \sigma_s}\right) (\text{cm})$	$\left(\frac{\Delta u}{\xi \sigma_s \sigma_{TR}}\right) (\text{cm}^2)$	$\left(\frac{\sigma_a \Delta u}{\xi \sigma_s}\right)$	$\left(\frac{\sigma_f \Delta u}{\xi \sigma_s}\right)$
Core	{	1	0.50	44.33087	195.3506	0.025303	0.0224482
		2	0.50	30.67175	102.8269	0.017540	0.0155315
		3	1.00	42.78642	104.5410	0.024622	0.0216660
		4	1.00	22.86277	32.2570	0.013710	0.0120135
		5	1.00	17.75097	18.0576	0.012411	0.0107136
		6	4.00	53.45805	44.4411	0.073409	0.0575717
		7	3.00	37.21482	27.3559	0.211591	0.1434556
		8	2.75	33.75139	24.1535	0.570484	0.3720549
		9	2.25	27.60783	19.7023	0.518032	0.2506534
		10	1.00	12.26982	8.4201	0.454581	0.2770065
		11	1.00	12.48493	7.8985	1.053361	0.6019580
		12	0.70	9.75892	6.4242	0.987784	0.5913599
		13	1.10	19.31459	13.8662	2.469754	1.5628174
		14	1.00	21.55573	16.5609	3.007650	1.9269890
Top Reflector	{	1	0.50	46.47682	223.7356	0.002163	
		2	0.50	32.39322	118.1677	0.001507	
		3	1.00	44.83296	115.8765	0.002086	
		4	1.00	24.15942	36.2038	0.001124	
		5	1.00	18.29009	18.7857	0.000851	
		6	4.00	56.13148	48.3557	0.002633	
		7	3.00	39.49232	31.4774	0.004016	
		8	2.75	35.85604	28.5233	0.013832	
		9	2.25	29.32982	23.2773	0.039351	
		10	1.00	13.03523	10.3114	0.038011	
		11	1.00	13.26001	10.6008	0.063502	
		12	0.70	10.34862	8.9940	0.076798	
		13	1.10	20.44155	21.3945	0.242059	
		14	1.00	22.77304	27.1798	0.342518	

*These data were prepared by means of a machine code and the number of digits retained is not indicative of the accuracy of the data.

Blanket

Side Reflector

Group	Δu	$\left(\frac{\Delta u}{\xi \sigma_s}\right) (\text{cm})$	$\left(\frac{\Delta u}{\xi \sigma_{s, \text{TR}}}\right) (\text{cm}^2)$	$\left(\frac{\sigma_a \Delta u}{\xi \sigma_s}\right)$	
Blanket	1	0.50	64.57055	138.0529	0.029748
	2	0.50	64.02474	133.6266	0.029496
	3	1.00	137.45886	290.8249	0.063328
	4	1.00	141.12853	273.1437	0.065019
	5	1.00	107.97950	146.2671	0.049747
	6	4.00	321.22014	371.0543	0.148769
	7	3.00	101.01481	39.6733	0.137510
	8	2.75	94.79963	37.9802	0.547748
	9	2.25	77.46653	30.5484	1.565120
	10	1.00	33.02529	12.7354	1.453532
	11	1.00	32.25126	12.0997	2.271737
	12	0.70	24.08094	8.7184	2.680094
	13	1.10	37.84148	12.7991	6.626402
	14	1.00	34.40135	10.8384	7.799285
Side Reflector	1	0.50	34.90389	156.0766	0.000885
	2	0.50	22.20987	66.6044	0.000563
	3	1.00	29.76010	61.6584	0.000755
	4	1.00	14.99581	16.0384	0.000380
	5	1.00	11.78096	9.5457	0.000298
	6	4.00	34.02143	20.8569	0.000874
	7	3.00	23.41671	12.7888	0.001717
	8	2.75	21.18065	11.4401	0.006616
	9	2.25	17.32550	9.3445	0.018892
	10	1.00	7.70015	4.1455	0.018243
	11	1.00	7.84319	4.2909	0.030540
	12	0.70	6.17138	3.7596	0.037264
	13	1.10	12.40853	9.6646	0.119584
	14	1.00	14.06788	13.0901	0.171983

Fission Spectrum χ_F

Group

 χ_F 1
2
3
4
50.023
0.102
0.445
0.300
0.130

Bibliography

- (1) R. Ehrlich and H. Hurwitz, Jr., "Multigroup Methods for Neutron Diffusion Problems" NUCLEONICS, Vol. 12 (February 1954)
- (2) H. Hurwitz, Jr., R. Ehrlich, "Comparison of Theory and Experiment for Intermediate Assemblies", Geneva Paper P/608 (June 1955)
- (3) H. Hurwitz Jr., and R. Ehrlich, "Progress in Nuclear Energy", Series I, Physics and Mathematics, Vol. 1, Chapter 11, Pergamon Press, London (1956)
- (4) KAPL-950 "Adaptation of Multigroup Methods to Cylindrical Geometries" G. M. Roe (September 1954)
- (5) KAPL-1641 "Two-Dimensional r- θ Multigroup Calculations, (Second Edition)" E. L. Wachspress (August 1954)
- (6) T. M. Snyder, "The Critical Assembly - A Nuclear Design Tool" Geneva Paper P/602 (July 1955)
- (7) KAPL-1241 "The Absorption of Neutrons in Doppler Broadened Resonances" G. M. Roe (October 15, 1954)
- (8) KAPL-1262 "Self-Shielding Factors for Infinitely Long, Hollow Cylinders" J. Dwork, P. L. Hofmann, H. Hurwitz, Jr., E. F. Clancy (January 10, 1955)
- (9) KAPL-M-CJH-2 "The Punched-Card Library of Cross Sections at KAPL" C. J. Hibbert

FIGURE 2

A SCHEMATIC DIAGRAM OF FPA-19

STATIONARY HALF Lower				MOVABLE HALF Upper			
27.55"				27.55"			
7.35"	20.20" For 6 drums			18.9"	8.65"		
2.45 9,474.14 12.73%	Be SS Mg Void	35,468.67 1,290.38 289.76 10.62%		Be SS Mg Void	33,139.34 1,218.11 289.76 10.53%	66.107 10,984.23 13.68%	
	.913"			.976"			
6.5"	7.804"	23.72"	13.72"	13.72"	7.741"	6.5"	
		22.437"	22.437"				
End Blanket	End Reflector	Transition	Core	Transition	End Reflector	End Blanket	
8,865.611	8,761.412 580.70 1,956.69	841.92 117.02 235.87	Be SS Mg Al 235 238 Void Cd	30,247.64 3,327.32 3,637.92 3,353.60 192.16 12.62 12.71% 54.71	931.71 117.64 248.31	7,181.10 588.25 2,640.11	3,586.146
11.24%	11.794%	15.93%		15.46%	8.32%	43.18%	
6.55"	21.00" Between Drums			19.70"	7.85"		
4,404.56 11.32%	Be SS Mg Void	19,667.83 693.91 149.88 8.06%		Be SS Mg Void	18,477.79 650.89 149.88 7.81%	38.03 5,187.02 13.08%	
27.55"				27.55"			
Total Side Blanket				SS	133,180.84		
				Void	10.514%		

Above Values in mols

All drums at 106°

FIGURE 3
A SCHEMATIC DIAGRAM OF THE EQUATORIAL PLANE
OF PPA-19

KS-16646

FIGURE 4
REACTOR MODEL

KS-16647

FIGURE 5
(Y, θ , R) SPACE

KS-16648

P.L. HOFMANN
M.L. STORM

FIGURE 6
RELATIVE AVERAGE FLUX IN GROUP I
(U=0 TO U=.5)

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

R. L. HOEMANN
M. L. STORM

FIGURE 7
RELATIVE AVERAGE FLUX IN GROUP 2
($U=0.5$ TO $U=1.0$)

P.L. HOFMANN
M.L. STORM

FIGURE 8
RELATIVE AVERAGE FLUX IN GROUP 3
($U=1.00$ TO $U=2.00$)

FIGURE 9
RELATIVE AVERAGE FLUX IN GROUP 4
(U=2.00 TO U=3.00)

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

FIGURE 10

RELATIVE AVERAGE FLUX IN GROUP 5
(U=3.00 TO U=4.00)

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

FIGURE 11
RELATIVE AVERAGE FLUX IN GROUP 6
(U=4.00 TO U=8.00)

P.L. HOFMANN
M.L. STORM

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

FIGURE 12
 RELATIVE AVERAGE FLUX IN GROUP 7
 ($U=8.00$ TO $U=11.00$)

R.L. HOFMANN
M.L. STORM

FIGURE 13
RELATIVE AVERAGE FLUX IN GROUP 8
($U=11.00$ TO $U=13.75$)

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

P.L. HOFMANN
M.L. STORM

FIGURE 14
RELATIVE AVERAGE FLUX IN GROUP 9
($U=13.75$ TO $U=16.00$)

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

P.L. HOFMANN
M.L. STORM

FIGURE -15
RELATIVE ACTIVE FLUX IN GROUP 10
(U=16.00 TO U=17.00)

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

APRIL 25, 1954

101-6-0-0159

FIGURE 16

RELATIVE AVERAGE FLUX IN GROUP II
(U=17.00 TO U=18.00)

R. L. HOFMANN
M. L. STORM

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

FIGURE - 17
 RELATIVE AVERAGE FLUX IN GROUP 12
 (U=18.0 TO U=18.70)

R.L. HOFMANN
 M.L. STORM

APRIL 22, 1954

REPRODUCED
 FROM THE ORIGINAL

FIGURE 18
 RELATIVE AVERAGE FLUX IN GROUP 13
 (U=18.70 TO U=19.8)

GENERAL ELECTRIC CO.
 KNOX ATOMIC POWER LAB.

DOWNGRADED
 To Confidential - Security Data
 By [Signature]
 10/12/1991

P.L. HOFMANN
M.L. STORM

FIGURE 19
RELATIVE AVERAGE FLUX IN GROUP 14
(THERMAL AT U=19.8)

P.L. HOFMANN
M.L. STORM

FIGURE 20
RELATIVE TOTAL FLUX

GENERAL ELECTRIC CO.
KNOLLS ATOMIC POWER LAB.

P. L. HOFMANN
M. L. STORM

FIGURE 21
RELATIVE POWER DISTRIBUTION

FIGURE 22
EXPERIMENTAL AND CALCULATED RADIAL POWER DISTRIBUTIONS
IN THE EQUATORIAL PLANE

KS-16649

FIGURE 23

EXPERIMENTAL AND CALCULATED AXIAL POWER DISTRIBUTIONS

KS-16650

FIGURE 24

EXPERIMENTAL AND CALCULATED LONGITUDINAL POWER
DISTRIBUTIONS FOR PPA FUEL SLUG 171

KS-16651

170 007
400 041

P. L. HOFMANN
M. L. STORM

FIGURE 25

RELATIVE SODIUM ACTIVATION

NOTE: TO CONVERT TO DISINTEGRATIONS/SEC. / MOLE / WATT, MULTIPLY
BY 15.16×10^4

Table 1. Calculated Dimensions of Reactor Model *

Core radius	36.8205 cm
Outer radius of side reflector	57.5349 cm
Outer radius of blanket	75.3481 cm
Half-height of core	34.8488 cm
Half-height of side reflector	50.6730 cm
Half-height of core plus end reflector	56.8797 cm
Total half-height	73.3897 cm

*Core dimensions were chosen to obtain an accurate core volume and hence fuel load representation. The number of significant figures in the dimensions of the remaining reactor regions were arbitrarily taken to be the same as for the core dimensions.

Table 2. Composition of Reactor Model

Material	Density, g/cm ³	Volume Fraction			
		Core	Top Reflector	Side Reflector	Blanket
Be	1.84	0.4991	0.4624	0.8403	
Steel	7.97	0.0785	0.05241	0.04339	0.8797
Mg	1.74	0.1713	0.3784	0.01975	
Al	2.70	0.1129			
U ₂₃₅	18.7	0.008135			
U ₂₃₈	18.7	0.000541			
Cd	8.648	0.002396			

Table 3. Values for Δr , Δz , N_j , M_k Used in Multigroup Calculations

	Coarse Mesh	Intermediate Mesh		Coarse Mesh	Intermediate Mesh
Δr	5.26009 cm	1.84103 cm			
Δz	5.80812 cm	1.93604 cm			
N_1	7	20	$N_1 \Delta r$	36.8206	36.8206
N_2	11	31	$N_2 \Delta r$	57.8610	57.0719
N_3	14	41	$N_3 \Delta r$	73.6413	75.4822
M_1	6	18	$M_1 \Delta z$	34.8487	34.8487
M_2	9	26	$M_2 \Delta z$	52.2731	50.3370
M_3	10	29	$M_3 \Delta z$	58.0812	56.1452
M_4	13	38	$M_4 \Delta z$	75.5056	73.5695

Table 4. Value Chosen for f in Group 1 (f_1)

f_1	Core	Top and Side Reflector	Blanket
f_{11}	1.70	1.06	1.10
f_{12}	1.40	1.06	1.30
f_{13}	1.30	1.20	1.10

Table 5. Comparison of Coarse and Intermediate Calculations

		E_p	E_F	ν_c	Approximate Calculation Time, hr
Coarse Mesh	Run 1	0.0802	0.00184	2.447	2
Coarse Mesh	Run 2	0.0176	0.000523	2.428	2
Intermediate Mesh	Run 1	0.0363	0.00100	2.435	8
Intermediate Mesh	Run 2	0.00494	0.000187	2.441	8

Table 6. Equivalent Core Heights

<u>Radial Position</u>	<u>Equivalent Core Height, cm</u>
n = 0 Δr (core)	103.4
12 Δr (core)	104.3
24 Δr (reflector)	104.3
36 Δr (blanket)	103.1

Table 7. Average Sodium Activation in PPA-19
(disintegrations/sec/mol/watt)

Region	Experimental	Calculated
Core	2.7×10^5	2.4×10^5
Transition	4.2×10^5	3.0×10^5 (top reflector)
End reflector	4.0×10^5	
Stationary reflector	3.0×10^5	4.0×10^5 (side reflector)
Control drum	3.0×10^5	