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AEC RESEARCH AND DEVELOPMENT REPORT

LOS ALAMOS SCIENTIFIC LABORATORY OF THE UNIVERSITY OF CALIFORNIA • LOS ALAMOS NEW MEXICO

SOME ROVER NEUTRONICS STUDIES (Title Unclassified)

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SOME ROVER NEUTRONICS STUDIES

(Title Unclassified)

by

C. B. Mills

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ABSTRACT

Neutronic characteristics of the nuclear powered rocket reactor are studied by multigroup numerical methods in this order. First, several simple critical experiments with a large variation in C/U^{235} and BeO/U^{235} show that the neutron cross sections are adequate for reactor studies. Second, a simple gas scattering model for thermal neutron scattering cross sections is shown by calculation to agree with experiment for a variation in reactor temperatures and so to be useful for Rover neutronics. Third, control rod effects on fission density and kinetic constants are computed for two of the Rover reactors using a 2-dimensional (x,y) transport code.

ACKNOWLEDGMENTS

We have been kept in touch with the excellent experimental studies by the critical experiment group at the University of California at Livermore, California, and have received an entire set of their Hot Box Notes. J. J. Griffin of Los Alamos investigated the gas scattering laws for neutrons; and the General Atomics group working on neutron thermalization theory and experiments, and especially on crystal effects,

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has provided us with their thermalization code. Descriptions of the Kiwi B were provided by E. Plassmann, who, also with J. Orndoff and J. Hansen, has been the source of many helpful discussions.

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1. Introduction

This report summarizes the results of some of the work done from 1958 to the present on problems associated with the reactor physics of the graphite moderated nuclear powered rocket (Rover) reactor. The study method is briefly described as follows. Reactor physics associated with higher power reactor development aims first at a detailed knowledge of simple reactors with the same materials and flux spectra. With a fully developed computing scheme and neutron cross sections that show no significant errors in correlation between theory and experiment for simple systems, the more complex high power and temperature systems can be analyzed with the same methods with some confidence. This approach de-emphasizes neutron cross section problems. The largest uncertainty during the period of development of the computing system for the Rover reactor type was connected with thermal flux effects on the multiplier constant of the reactor and the fissioning density as a function of reactor temperature and composition for intermediate* C/U²³⁵ ratios. A simple free-atom model¹ was shown to be sufficiently accurate for the reactor series under study, so only a small supplementary effort was made to include the crystal model research (see, for example, Ref. 2) of Parks, Nelkin, et al. Several previous reports³ and parallel studies show the place of the nuclear powered rocket

*Low C/U²³⁵ ratio reactors have relatively small thermal neutron flux; high C/U²³⁵ ratio reactors have a Maxwellian flux.

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reactor in the whole spectrum of reactor types. The thermal flux spectrum is intermediate, so that numerical multigroup methods are required, and a large variation of composition (with resulting large flux gradients) makes transport (angle dependent) equations necessary.

The work contained here supplements that by Eugene Plassmann,⁴ who is concerned with the complex and general problem of correlating the experimental results and predicting high temperature performance of the very high power Rover reactors.

We will present those background critical experiments and thermalization effects which are of special pertinence to Rover and were not previously studied in LA reports, and then show the results of the use of the computing system on those problems that are not included in Plassmann's work. This latter part will consist of first 1-dimensional (S_4 cylindrical) and then 2-dimensional transport (S_4) (x,y) coordinate representation of the complex 3-dimensional Rover reactor at room temperature. No space will be given to the evolution of nonthermal neutron cross sections, and only the results of some thermal studies will be given, since the theory of thermal neutron scattering is well treated elsewhere.²

2. Analytical Methods

Both diffusion and transport theory of neutron diffusion and slowing down have been used for the intermediate neutron flux spectrum studies necessary for the Rover reactor. These have been reported in

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detail elsewhere⁵ as separate items. In addition a summary of all the reactor physics methods and constants used for fast to thermal systems has been prepared.⁶ The basic approach has been to use the very flexible transport theory codes developed by Carlson⁵ with cross section studies, particularly those by Hansen et al.⁷ and Bell,⁸ to study a very wide variety of simple experiments. The correlations fix the accuracy in a general way so as to minimize the neutron cross section integrations which provide the largest source of problems.

In Section 3, simple reactor criticality experiments appropriate to the Rover program are discussed separately from the reactor temperature dependent experiments to provide a departure point for thermal flux studies. Some of the experiments supporting Section 4, the 1- and 2-dimensional analysis of some of the Rover reactors, have been described previously.³ The Rover reactors, Kiwi B, will be described in a later report by E. Plassmann.

A 2-dimensional (x,y) transport code,⁹ which has been found useful, can represent the Rover reactors only in a limited sense. The method used by Plassmann involves a 1-dimensional cylindrical neutron transport code that simulates control rod effects by an absorbing annulus. Here we use a DB² absorption to make the 1-dimensional cylinder finite consistent with the experimental reactors, and study core neutronics, but the detailed effects of control rod position are obscured. Conversely, the 2-dimensional (x,y) code can represent the reactor as a cell bounded on three sides (see Figure 1) by a zero neutron current

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Fig. 1 - Geometry conversion of reflected reactors from 3-dimensional to 1- and 2-dimensional.

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condition, and with the x-coordinate along a reactor radius. With this, control rod absorption can be studied with some sacrifice of core neutronics. The reactor dimension along x must, of course, have the same neutron flux spectrum as the finite cylinder. This implies at least an equivalent geometrical buckling, which is made difficult to know by a large variation in core composition which is used to flatten the radial temperature profile of the high power reactor.

3. Experiments with BeO and C Showing the Neutron Flux Spectrum Effects

(a) Room Temperature Critical Assemblies

A wide variation in moderator to U^{235} atomic ratios for the two materials -- beryllium oxide¹⁰ and graphite¹¹ -- was studied at the University of California, Livermore, to uncover higher energy neutron flux spectrum-dependent effects at high temperatures. The experiments were simple rectangular parallelepipeds using uranium in the form of foils 1 to 8 mils thick, in which the lower limits of fuel concentration were approached by segmenting large foils into smaller areas. The effects of foil thickness were eliminated by extrapolation techniques for the graphite experiments, but not for the BeO set. Accordingly, the only correction not of a geometrical nature for the graphite experiments involved evaluation of trace impurities (reduced to 5 ppm boron). The BeO experiments had relatively large foil (self-shielding) effects, which were included in the calculations in simple ways.

A summary of the experiments is shown in Table I. For the purpose

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TABLE I
REFERENCE CRITICAL ASSEMBLIES

Atomic ratio BeO/U ²³⁵	Critical radius (cm)	<u>BeO Moderated</u>		$k_{eff}^{(a)}$
		Atomic density ($\times 10^{24}$ atoms/cc)		
		U ²³⁵ (and foil thickness)	BeO	
493	34.45	1.396×10^{-4} (4 mil)	0.06882	0.999
986	35.97	6.97×10^{-5} (2 mil)	0.06807	0.999
1920	38.56	3.58×10^{-5} (1 mil)	0.06885	0.999
3826	43.72	1.80×10^{-5} (1 mil)	0.06879	0.979
7660	53.06	0.899×10^{-5} (1 mil)	0.06885	0.960

<u>Graphite Moderated</u>				
Atomic ratio C/U ²³⁵	Critical radius (cm)	<u>Atomic density ($\times 10^{24}$ atoms/cc)</u>		$k_{eff}^{(a)}$
		U ²³⁵	C	
301	63.48	2.734×10^{-4}	0.0823	0.995
603	64.66	1.368×10^{-4}	0.0825	0.979
1206	66.62	0.684×10^{-4}	0.0825	0.985
2355	69.36	0.350×10^{-4}	0.0825	0.992
9550	83.72	0.0864×10^{-4}	0.0825	0.993

(a) $k_{eff} = \bar{\nu}/\nu_c$, where the mean experimental number of neutrons per fission was assumed to be 2.450, and ν_c is the computed value for each experiment.

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of calculation, approximately cubic experimental geometry was reduced to spherical by an equal geometrical buckling assumption $\left(B_{\text{cube}}^2 = B_{\text{sphere}}^2 \right)$. The computed critical eigenvalue (ν_c), the number of neutrons per fission (ν) for which the neutron conservative equations are satisfied, was reduced to the familiar effective neutron multiplication (k_{eff}) with a value of 2.450 for ν . (Note: recent experiments by Diven and Hopkins¹² show $\nu(\text{thermal}) = 2.427$ for U^{235} .) This set of calculations demonstrates the adequacy of the C and U^{235} cross sections for criticality studies of a wide variety of neutron flux spectra - from intermediate to thermal.

(b) High Temperature Critical Assemblies

These Livermore experiments¹³ involved construction of an apparatus for high temperature measurements of criticality. There is a disagreement between critical size of the simple reference experiments and the relatively complicated assemblies that were heated. In terms of k_{eff} , this amounts to 1 to 2%. Accordingly, rather than the experimental value of ν , the base for the calculations was $\bar{\nu}$, which was the mean experimental value. Temperature effects are, then, relative. This does not affect the evaluation of temperature coefficients of reactivity.

The experiments, performed between 90° and 1200°F, were in an assembly called Hot Box. Uranium was protected by stainless steel (347) foil, which had only a small effect on reactivity (k_{eff}). There were, however, large effects due to complex geometry. The BeO experi-

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ments¹³ (HB-22)* have roughly the appearance of a cross (+) of BeO, with a thick fuel section in each quadrant, the whole then enclosed in BeO. Smaller effects were due to a central cross of graphite and a box enclosure of graphite, both of which may be eliminated from the calculation. The graphite experiment¹⁴ suffered from effects of large U²³⁵ foil separation in one experiment (HB-14) and much smaller similar effects in a second set (HB-18).

(c) Beryllium Oxide

Table II describes the extremes of the high temperature BeO-U²³⁵ experiments. The relatively complex cruciform shape was simply homogenized to an atomic ratio BeO/U²³⁵ of 565 for one trial calculation. Next, the first approximation to the experimental geometry was retained by representing the reactor as a symmetric slab (in the order: BeO, fuel, BeO, fuel, BeO), with the actual geometry for one coordinate, and a neutron escape term ($B^2/3\Sigma_{tr}$) that provides an appropriately spectrum-dependent neutron loss for the other two coordinates. One of these coordinates was used to vary the reactor dimension for temperature-dependent reactivity control. Note that in Table II both approximations show a small error in k_{eff} , which may be ascribable to temperature.

The temperature coefficient of reactivity of critical assemblies may be separated into geometrical (expansion) and nuclear (spectrum

*HB-22 refers to Hot Box Note 22, which reported experimental details for internal UCRL-L distribution. Other HB numbers refer to similar notes.

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TABLE II

BERYLLIUM OXIDE ASSEMBLIES (HB-22)

Tempera- ture (ev)	Atomic ratio BeO/U ²³⁵	Critical core dimensions (cm)	Atomic density (× 10 ²⁴ atoms/cc)			k _{eff} ^(a)
			U ²³⁵	BeO	Fe	
<u>Sphere Radius</u>						
0.0252	565	40.79	1.155 × 10 ⁻⁴	0.06527	6.47 × 10 ⁻⁴	1.000
0.0676	565	41.52	1.148 × 10 ⁻⁴	0.06488	6.43 × 10 ⁻⁴	1.006
<u>Slab Boundaries</u> ^(b)						
0.0252	∞,373.2,∞	4.744,26.84,32.43	1.787 × 10 ⁻⁴	0.0667	10.31 × 10 ⁻⁴	1.000
0.0676	∞,373.2,∞	4.744,26.89,32.43	1.787 × 10 ⁻⁴	0.0667	10.31 × 10 ⁻⁴	1.001

(a) $k_{\text{eff}} = \frac{\nu(T \text{ 0.0252})}{\nu(T \text{ 0.0676})}$, where ν is the critical number of neutrons per fission.

(b) Reactor computed as a symmetric slab, with U²³⁵ in the center section of each reactor half. Reactor criticality is experimentally determined by thickness of another dimension for which $B^2(0.0252 \text{ ev}) = 0.05392$ and $B^2(0.0676 \text{ ev}) = 0.05205$. (B^2 is geometrical buckling determined for length of side of reactor assembly varied with temperature.)

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shift) effects. The nuclear effect alone, between 0.056 and 0.0676 ev, for the two BeO geometries, was computed to be:

<u>Geometry</u>	<u>$\Delta k / \Delta T (\text{ev})$</u>	<u>$\Delta k (\text{cents}) / \Delta T (100^\circ \text{F})$</u>
Homogeneous sphere	0.81	49
Heterogeneous slab	0.47	28

or, in particular, $\Delta k = 0.0094$ and 0.0054 for a change from 0.0676 to 0.056 ev for the sphere and slab, respectively. This difference illustrates the importance of a detailed calculation of an experiment. The experimental values were 29 to 30¢/100°F.

(d) Graphite

Table III summarizes results of both sets of high temperature graphite U^{235} experiments. Both are neutronically complex but geometrically simple. Table IIIA (HB-14) describes an approximately cubic experiment with wide U^{235} fuel foil spacing as equivalent cells, with a neutron escape term (DB^2) in the form of an equivalent flux spectrum-dependent absorption. The symmetric cell for k_{eff} calculations contains U^{235} , stainless steel, and graphite parallel slabs in the same way the experiment contains them, but with the thickness of the graphite layer determined by the C/U^{235} ratio in the experiment. Both C/U^{235} dependence for a constant geometry and constant C/U^{235} ratio for a varying geometry are shown.

Table IIIB (HB-18) describes approximate cubes, with thick graphite reflectors on one coordinate, as equivalent spheres with the reflector

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TABLE IIIA

GRAPHITE ASSEMBLIES (HB-14)

Graphite temperature (ev)	Atomic ratio C/U ²³⁵	Geometrical buckling B ²	Atomic density (× 10 ²⁴ atoms/cc)		Graphite slab thickness (cm)	k _{eff} ^(a)
			N(C)	N(U ²³⁵)		
0.0252	8950	9.35 × 10 ⁻⁴	0.07322	0.0448	13.91	0.996
0.0560	7476	9.35 × 10 ⁻⁴	0.07322	0.0448	11.62	1.005
0.0788	6743	9.35 × 10 ⁻⁴	0.07322	0.0448	10.48	1.009
0.0244	8700	9.58 × 10 ⁻⁴	0.07322	0.0448	13.51	0.993
0.0642	8700	8.41 × 10 ⁻⁴	0.07322	0.0448	13.51	0.996

(a) Cell calculations, diffusion approximation; $k = \bar{\nu}/\nu_c$, where $\bar{\nu}$ is the average number of neutrons per fission for each experiment to be critical, and ν_c is the computed specific value.

TABLE IIIB

GRAPHITE ASSEMBLIES (HB-18)

Graphite temperature (ev)	Atomic ratio C/U ²³⁵	Critical sphere radius (cm)	Atomic density (× 10 ²⁴ atoms/cc)		k _{eff} ^(a)
			N(C)	N(U ²³⁵)	
0.0252	3140	82.72	0.07347	2.34 × 10 ⁻⁵	1.001
0.039	2832	82.82	0.07335	2.59 × 10 ⁻⁵	1.000
0.056	2514	82.88	0.07317	2.91 × 10 ⁻⁵	1.001
0.0788	2288	82.95	0.0730	3.19 × 10 ⁻⁵	0.997

(a) Homogeneous bare reactor calculations; 2-mil foil and 4-mil stainless steel; $k_{eff} = \bar{\nu}/\nu_c$ as above. Diffusion approximation used with $\epsilon = 10^{-4}$.

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thickness converted to core material in the familiar "reflector savings" manner. Much smaller foil spacing than in the earlier experiment (HB-14) permits the use of homogeneous reactor calculational methods, with the 2-mil-thick (0.00508 cm) foil implicit in corrected U^{235} cross sections. This experiment is relatively simple to compute because of the geometry and the relatively small effects of foil thickness and spacing.

(e) Remarks on the Study of These Experiments

These high temperature (90° to 1200°F) BeO and graphite-moderated experiments, although very well reported, are difficult to analyse. As an example of the difficulties, Figure 2 shows the fissioning density along one coordinate across the fuel of the experiment. This should be a simple cosine function if thermal effects from the BeO layers are small. Clearly there are large effects outside of the fuel region.

Figure 3 shows the distribution of 0.0252 ev neutrons across a cell in a graphite experiment. The absorption by impurities in the cell, because of the $1/\text{velocity}$ dependence of absorption, would cause an important temperature-dependent error if homogeneity were assumed. The method of calculation requires the use of reasonably homogeneous and simple geometries because of limitations in the amount of data that can be handled. The usual homogenization process, in which neutron cross sections are modified to accommodate flux distribution effects, required the use of geometry-determined reduction in impurity (largely boron) absorption cross section, which in turn requires an average over space and energy. Instead, a non-temperature-sensitive neutron escape term was used with the experiment, represented by a cell of an infinite lattice.

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Fig. 2 - Fission density on one coordinate of the BeO high temperature experiment.

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Fig. 3 - Neutron flux vs. cell dimension for the graphite high temperature experiment.

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This changed the reference eigenvalue (2.45 neutrons per fission) by several per cent, but conserved purely thermal trends in reactivity as a function of temperature. The second graphite experimental set gave no trouble except for a difference in ν_c of 3% between the reference critical assembly and the Hot Box critical assembly.

The effect of the particular model used, whether one thermal group with neutron cross sections averaged over a (free atom) Maxwellian flux distribution, a "differentiated" representation of thermal neutron cross sections and flux (11 groups $0 < E < 1$ ev), or a crystal model, is shown on Table IV. These trends as a function of temperature for each model are within the experimental uncertainties. The comparison of trends for the different models is significant.

A simple free-atom model gives acceptable results for reactor criticality as a function of temperature for BeO and C in the 90° to 1200°F range. A comparison with the more physically exact crystal model results shows that temperature coefficients, if important, should be computed with the crystal-model-derived cross sections.

4. Kiwi B Studies

Kiwi B is a graphite moderated reactor designed to give an almost flat fissioning density distribution at very high temperature. An illustration of the small difference between experiment and 1-dimensional calculation is shown on Figures 4 and 5. Here the experimental determination of fission density vs. position and the calculation show close correspondence.

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TABLE IV

EFFECT OF MODEL ON k_{eff} FOR GRAPHITE ASSEMBLIES
(HB-18)

Graphite temperature (ev)	k_{eff} (a)			
	one thermal group	gas model	crystal	gas/crystal ratio
0.0252	1.0016	0.9702	0.9770	0.9930
0.039	0.9855	0.9807	0.9887	0.9919
0.056	0.9897	0.9900	0.9992	0.9907
0.0788	0.9847	0.9950	1.0050	0.9900

(a) k_{eff} as computed from the experimental description of Hot Box 18.

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Fig. 4 - Fissioning density distribution for ZEPH-B, computed and experimental.

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Kiwi B-2 and B-4, discussed in some detail here, are different by a factor of 2 in U^{235} concentration, and B-4 has a much larger radial variation in core composition. Because of this, the fissioning density, neutron lifetime, delayed neutron value, and control rod value should be different, and this difference provides a good theory to experiment comparison.

(a) 1-Dimensional Studies

A transition between the generally familiar 1-dimensional and homogeneous reactor analyses and the 2-d (x,y) results described below is helpful. The use of Plassmann's description⁴ of Kiwi B-2A and -4A gives results (in Table V) for reactor constants (18 group neutron cross sections; 1 thermal group):

TABLE V

1-DIMENSIONAL KIWI B CONSTANTS

Effect	Kiwi B-2A	Kiwi B-4A
(1) Δk for removal of boron curtain	0.0209	0.0105
(2) Δk for $\rho(\text{mod.}) \times 1.072$ in reflector	0.0229	0.01797
(3) value of delayed neutrons (Δk)	0.00749	0.00792
(4) neutron lifetime (sec.)	43×10^{-6}	15.8×10^{-6}
(5) k_{eff} computed (approx. critical)	0.9937	1.0021
(6) per cent thermal fissions	10	5.5

These numbers result from 1-dimensional S_4 calculations of Kiwi B with

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certain assumptions:

(a) that the control rod bank (essentially 2-dimensional) can be simulated by a simple curtain of B in the reflector and that the effect is the same for 1- and 2-dimensional neutronics if the B thickness (or $\sum_a d$) is adjusted to give the same mean radial fissioning density as that in the just critical (control rods at 90° for $0 < \theta < 180^\circ$) reactor;

(b) the 1-dimensional computed results can be interpreted quantitatively in the light of reactor experiments and 2-dimensional (x,y) results; and

(c) the 1-dimensional cylinder computing code can include neutron transport effects of finite reactor length implicitly by adding a DB^2 term for the axial escape. We assume further that the 2-d (x,y) approximation gives true values for control rod effects. Then, since item (1) in Table V must be adjusted to 0.0433 and 0.0260 for B-2A and -4A (see Table VI), we have (5) for the absolute reference for the just critical reactor. This shows an error in the effective value of the boron curtain as compared to the actual localized and neutronically "black" control rods. Accordingly, reflector effects on neutron life-time and control rod value must be evaluated using a 2- or more-dimensional method. It was seen, however, by a study (item 2, Table V) of the effect of the Be reflector density on control rod sensitivity for Kiwi B-4A that certain reflector effects can be reasonably approximated. The calculation showed 9¢, the experiment 11¢ per degree¹⁵

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TABLE VI

CONTROL ROD EFFECT ON k_{eff}

	Kiwi B-2	Kiwi B-4
0°	0.9621	0.9969
90°	1.0028	1.0295
180°	1.0473	1.0630

for the difference in rod value. It is also easy to see that evaluation of prompt neutron lifetime requires a multi-dimensional approach since a part of the neutron life is spent in the thermalizing reflector.

Compare the experimental¹⁵ values of 43 ± 2 microseconds for neutron lifetime in the Kiwi B reactors with the reactor dependent values given above in Table VI and below in Table VII. These values, experimental and computed, show that 1-dimensional analysis does not suffice for some values and that at least 2-dimensional work should be done.

TABLE VII

REACTOR CONSTANTS VS. CONTROL ROD POSITION

	Kiwi B-2A	Kiwi B-4A
Total control rod value ($0-180^\circ$), Δk	0.0852	0.0661
Total control rod value (average)	\$11.4	\$8.34
Delayed neutron value (Table V)	0.00749	0.00792
Neutron lifetime (sec.), 0° rod	35.3×10^{-6}	12.8×10^{-6}
Neutron lifetime (sec.), 180° rod	62.2×10^{-6}	25.8×10^{-6}

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(b) 2-Dimensional Studies

2-dimensional (x,y) coordinate calculations must de-emphasize reactor core in favor of reflector effects on neutron transport (see Figure 1). The internal structure of one of these calculations is indicated on Figure 6, which shows a few of the integrated fission density values as a function of position. We will emphasize "azimuthal" variation of fission density since this cannot be done with the 1-dimensional methods. Note, however, that some sacrifice of absolute accuracy is occasioned by the rectangular cell calculation of a cylinder. This appears most clearly in the difference in k_{eff} for the 2A and 4A reactors. The 3% difference is due to the "reflector savings" being changed by 2 cm with no change in the core radius (along x) actually used.

(1) Fission Density

Figures 7 and 8 show azimuthal and the two extremes of radial variation of fissioning density per gram of U^{235} in the reactor core. The actual reduction of U^{235} concentration changes this to a nearly flat (constant radially) distribution of fissioning density per unit of core volume (adjusted to be flat at high reactor temperature). Figure 8 also shows a comparison of foil activator measurement of fission density around the core surface with that averaged in the nearest 0.5 cm of core volume. This comparison is also favorable in relating theory and experiment for fission density.

(2) Reactor Constants

The 2-d (x,y) transport calculation of the reactor as a cell assumes

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Fig. 6 - 2-dimensional (x,y) coordinate fission density for several x-coordinate lines.

Fig. 7 - 2-dimensional (x,y) code results for Kiwi B-2A.

Fig. 8 - 2-dimensional (x,y) code results for Kiwi B-4A.

that the whole reactor acts in the same way as the cell segment representation. A change in control rod position, as computed, effectively changes the reactor as though all control rods acted together. The results of k_{eff} as a function of rod angle from "in" (0°) to "out" (180°) are given in Table VI.

Because of the difference (a factor of 2) in U^{235} concentration, the two reactors have different neutronic constants. These are tabulated as a function of control rod position in Table VII. A sketch of the control rod geometry as required to fit an (x,y) coordinate system is shown on Figure 9. Note that representation of a curve demands a fine structure mesh in the space intervals required for numerical calculations.

A study of neutron lifetime was made by J. Orndoff¹⁵ for a reactor very like Kiwi B-2A. The measured value, necessarily for control rod position near 90° , was 43×10^{-6} sec. The value he measured for the delayed neutrons is $\Delta k = 0.0078$, somewhat higher than the computed values above. Both of these are close to the computed values in Tables V and VII.

A comparison of experimental and computed control rod value for Kiwi B reactors is given on Figures 10 and 11. The reason for the discrepancy shown on Figure 10 from Kiwi B-2A experiments is not clear, since the reactor experiment is similar in description to the one computed. A reduction in the discrepancy of 22% is possible by measuring the motion of all the rods together (Orndoff's experiment used 1 rod).

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Fig. 9 - 2-dimensional (x,y) control rod geometry.

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Fig. 10 - Control rod value vs. position for Kiwi B-1 and -2.

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Fig. 11 - Control rod value vs. position for Kiwi B-4A.

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Mortenson¹⁸ found an uncertainty here near 6% in differential rod value.

(c) Summary

This section was directed largely at the use of a 2-dimensional (x,y) transport calculation of typical Kiwi B reactors, as a supplement to Plassmann's studies. It emphasizes control rod effects on reactor neutronic constants. From the large effects on neutron lifetime and control rod value, it is clear that an operating (and completely unapproachable) high power reactor can be studied with profit by multi-dimensional reactor transport codes. With the type of reactor studied, control rod effects on neutron flux spectrum can change the fission density distribution significantly and neutron lifetime by a factor of 2.

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