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For The Atomic Energy Commission

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Static Physics of Rover Reactors

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Static Physics of Rover Reactors

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by

C. B. Mills

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ABSTRACT

The current series of nuclear-powered rocket reactors are small, graphite-moderated, beryllium-reflected cylinders loaded with U^{235} in such a way that the power distribution is roughly flat at high temperatures. Niobium protects the graphite during high-temperature operation. Boron is the shim and control material. Inasmuch as the temperature varies from very low in the reflector ($200^{\circ}R$) to very high in the core ($5000^{\circ}R$), the reactor assembly has relatively neutronicallly complex energy and space functions. The median energy for neutron absorption is 100 ev; thus, fast neutron transport, resonance absorption, and - because of the beryllium reflector - thermal neutron transport problems are all important.

The data needed for high-temperature reactor design are critical U^{235} fuel distribution and kinetic constants. Room-temperature experiments resulted in detailed structural data for experiments having several degrees of geometrical complexity. In turn, these data yielded criticality and structural material value coefficients that were compared with calculated values. These values included U^{235} , graphite, niobium, and hydrogen as functions of position and fission density distribution.

The variety of multigroup reactor physics techniques used to calculate static physics of the critical assemblies included numerical neutron transport, resonance integral, and thermalization codes for group-averaged cross sections and fast transport cross sections. The Rover reactor studies followed extensive analyses of critical assemblies to establish the overall effects of moderators and absorbers; and, thus, the integral results are expected to be relatively accurate.

The results of Kiwi-B studies were partially repeated to compare the computed and experimental fission density distributions. The description of the Phoebus reactor and the results of reactivity distribution studies show that k_{eff} is approximately equal to 1.00 ± 0.005 ; that the computed fission density distribution is within the experimental uncertainty; and that some errors still remain in resonance integrals, as is shown by the comparison of computed and experimental material reactivity values.

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STATIC PHYSICS OF ROVER REACTORS

C. B. Mills

Nuclear power for space flight propulsion, the Rover project, utilizes simple heat transfer to liquid—then gaseous—hydrogen, resulting in an engineering problem of some complexity, and providing a situation of interest in the field of reactor physics. A brief review follows of some of the work on reactor statics, to indicate methods used and results obtained, but without detailed inspections of nuclear physics or numerical transport theory, which have been described adequately elsewhere. Because all of the reactor integrals use neutron flux weighting, that aspect is the basis of the discussion below.

Rover Reactors

The reactors developed in this early stage of the nuclear propulsion program have been small cylinders (about 4 ft square), graphite-moderated (for maximum heat-transfer surface temperature and minimum critical mass), and beryllium-reflected (for optimum absorber control and, again, minimum critical mass). Various materials have been used as carbides to

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protect the core structure from evaporation and corrosion. Although the critical mass can vary between 10 and 1000 kg for geometries of interest, because of the protective material required as part of the reactor composition, the actual values are near 500 kg. The critical mass of earlier geometries without fuel element protection was as low as 16 kg, while adequately protected fuel elements (the heat-transfer material) utilizing even the highest melting-point metal carbides may require critical masses of well over 1000 kg. As one result of neutron absorption by the protective materials, only a small proportion of neutrons are thermalized. (From 5% to 10% of the fissions are due to "thermal" neutrons.) Consequently, fission absorption as a function of energy is relatively constant in the energy coordinate. Accordingly, resonance absorption integrals must be carefully done. The few thermal fissions that do occur result from neutron diffusion into the core from the strongly moderating beryllium reflector (no thermal neutrons are present in a large part of the reactor core), so that neutron transport effects also must be evaluated with care. In addition, because maximum total heat flux implies maximum and uniform temperature, the fission density (fissions/cc) must be reasonably constant, requiring a wide variation in U^{235} density. Last, as the propellant enters the core at a low temperature and leaves it at a very high temperature, the flux spectrum of the almost-thermal neutrons that do exist (moving from reflector into core) must change by a large degree axially.

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In summary, the Rover reactors are geometrically simple but neutronically complex, with large transport, resonance, and thermalization effects. The reactor flux is non-thermal in a large part of the core and near the strongly absorbing control rods in the reflector, but has a thermal spectrum that varies axially and azimuthally at the core periphery and in the reflector.

The measurements reported in experimental studies of Rover reactors are critical mass, reactivity value and distribution for component materials, and fission density distribution. The rectangular dimensions of the experimental assembly pieces vary from three inches in earlier stages to one inch for more refined stages. Within one inch, the thermal neutron flux can be attenuated by more than one order of magnitude very near the core boundary. Impurities are generally present in only trace amounts, and are relatively unimportant, except for water vapor, which can be troublesome for temperature-sensitive studies. The simplest experiments, called Honeycomb because of an aluminum support structure, are comprised of three-inch-rectangular beryllium and graphite parallelepipeds, 2-mil- (0.002-in.-) thick uranium, and 5-mil-thick niobium foils. Foil interaction effects are significant in calculations of integrals for the resonance region; consequently, the problem of describing the experiment in terms of a computable set of data is severe. Even in advanced experiments having homogeneous regions and simple surfaces, inherently more complex "averaged" geometries cause the descriptions to be complicated. The transformation of experiment to equivalent idealized (computable) geometry is discussed below. Also, inasmuch as reactivity studies compare the effect on reactor

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period of gross material removal, with simple density changes in the reactor, caution must be observed in evaluations of such procedures.

The integral effect of reactor materials on neutron multiplication, computed usually as Δk_{eff} , depends in absolute magnitude on a large number of approximations. It is necessary, therefore, to establish a reference point, usually the basic one of a series of critical experiment descriptions, and move from there. Fast computing codes for the solution of transport problems make these studies highly self-consistent and provide satisfactory solutions. This is especially clear in the case of neutron cross section effects, for which the results are frequently more exact than the basic numbers.

The results of experimental and theoretical studies must include a knowledge of critical mass, fission density distribution, value of control assemblies and structural parts, kinetic constants, and the dependence of these factors on reactor temperatures ranging from 0° to 5000°R (0.0 to 0.3 ev).

Reactor Static Physics Methods

Reactor physics computations utilize numerical codes to solve neutron transport equations and to generate resonance absorption integrals and thermal neutron scattering matrices. All cross sections are not yet entirely computable; however, the neutron cross section studies by Hansen¹ for $0.1 < E < 14$ Mev energy groups, the resonance studies for $1.0 \text{ ev} < E < 0.1 \text{ Mev}$ by Bell² and by Nordheim,³ and the thermalization studies by

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Beyster and associates⁴ for $E < 1.0$ ev provide the elements of a complete set of multigroup neutron cross sections (fission, absorption, transport, and inter-group transfer). These data are effectively independent of geometry (the space part of the transport equation). Then, fast and flexible transport codes - one-dimensional slab, cylinder, sphere, and two-dimensional (r,z) , (r,θ) , and (x,y) geometries - with microscopic neutron cross section constants appropriate to the reactor details, complete the reactor physics study system.

The weighting function for all neutron absorption and transport processes is, of course, the neutron flux. A few typical functions of space for different energy groups are illustrated in Fig. 1. To evaluate neutron cross sections and fluxes in various energy vicinities, using the integral methods of reactor physics, it is necessary to choose experiments for each material of interest with the flux spectrum maximized in each vicinity. (Inspect Fig. 2 for $\phi(r,u)$ at constant radius, r , and for the neutron cross sections.) The selection procedure is tedious, requiring close attention to experimental reactor assemblies for a wide variety of materials and geometries; but it has successfully produced both useful integral results and confidence in those specific to the Rover reactors. In the following discussion, the neutron transport codes, neutron cross sections, and experiments will be described separately, although actual experimental analysis requires an inter-related approach.

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Fig. 1 Neutron flux $\phi(r)$ in Rover reactors.

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LINEAR SCALE

Fig. 2 Neutron flux $\phi(E)$ and cross sections $\Sigma_a(E)$ in Rover reactors.

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Neutron Transport Codes

The numerical solution of the equation for neutron flow has been highly developed by B. Carlson and associates in the past few years.⁵⁻⁸ The numerical codes solve for $N(r, \mu, E)$ where N is the neutron flux in energy interval ΔE at E (or in lethargy interval Δu at u) as a function of position in space r and in angle interval $\Delta \mu$ at $\mu (\mu \equiv \cos \theta)$. In particular, Floco (DTK) or Fortran (DTF) versions of the numerical transport codes are used for one-dimensional slab, cylinder and sphere geometries; and codes DDK and DDF, for (r, z) , (r, θ) , and (x, y) two-dimensional geometries. The energy variable, E , is divided into a number of narrow intervals to cover the whole energy range of interest, $0 < E < 14$ Mev. The separation of space and energy variables is made possible by defining inter-group neutron transfer coefficients $\sigma_{g \rightarrow g'}$. These recent transport codes have excellent neutron conservation properties, and are fast and flexible. Published descriptions of the codes are not quite finished; however, several works are available.⁵

The DTK code, used in S_2 or higher approximations, has the following characteristics: the number of space intervals to be used for reactor description is determined more by desired fidelity to the experiment than by internal numerical code approximations. For example, the Rover reactor described below was computed with three sets of radial intervals, preserving only the interval at the core boundary as 0.1 cm. The results in terms of k_{eff} (relative to unity) and fission density ($F_{\text{boundary}}/F_{\text{center}}$) were:

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	k_{eff}	F_b/F_c
Half size Δr	1.0004	2.61
Reference	1.00	2.62
Double size Δr	0.9991	2.63

However, too crude a mesh in neutron flux, a cross-section rather than a code change, can result in much larger differences. For example, changing from 25 to 15 g (preserving intact the first five groups and using a simple constant flux weighting function) changed k_{eff} by -4%, a relatively large error (about 30% in critical mass equivalent) and F_b/F_c by 15%.

The completion of a problem is determined by iterative comparison of neutron fluxes and integrals. Clearly, the smaller the difference between terms in a convergent series of calculations, the more accurate the sum; and the longer the time required. The DTK code, Carlson's most advanced, has errors in k_{eff} , as compared to asymptotic values of the same magnitude or smaller than the convergence index, so that perturbation (reactivity) calculations can be made relatively quickly. For example, in complex thermal problems, 25 minutes are required to move from a convergence index of 10^{-3} to 10^{-5} , while only 10 minutes are needed to reach 10^{-4} . Inasmuch as the experiments are made with $\Delta k \approx 10^{-4}$, the shorter time seems adequate. As the problem converges toward tighter convergence criteria, k_{eff} also changes slightly. For example, $10^{-3} \leq \epsilon \leq 10^{-5}$ gave $1.00118 \leq k \leq 1.00000$. This change affects the reactivity coefficient calculations discussed below.

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Approximations and Uncertainties

The Rover reactor series fits very easily into the computing scheme discussed here, which was designed to evaluate a wide variety of reactors. The largest approximations, and consequent errors, are the geometrical transformations from the heterogeneous experimental arrangements to the regionally homogeneous theoretical systems consistent with reasonable computing times.

The second largest group of uncertainties in the case of the Rover reactors is in the areas of thermalization and unresolved resonance. The latter deserves more attention than it has received, because the effective magnitude of unresolved resonances on k_{eff} is approximately 10%. This value is much larger than the effect on k_{eff} of moving from room temperature to higher temperatures. The quality of the effect is that of removing neutrons and of changing the capture-to-fission ratio (Σ_c/Σ_f) as the neutron flux moves from high- to low-energy values. Inasmuch as only 10% of the neutrons escape (escape being spread relatively uniformly over the energy spectrum) the competition Σ_c/Σ_f , as a function of energy, determines both k_{eff} and the neutronic temperature coefficients of reactivity.

Neutron thermalization problems are relatively small because the thermal flux in the reactor core is extremely small. The thermal and epithermal fissions result almost entirely from neutrons diffusing into the core from the reflector; thus, the rate of energy change at the core boundary is the more important quantity.

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However, large errors can be tolerated in only one place: the absorption cross section of the control rod. This is due to the fact that the control rods are "black", so that increased absorption has little effect. Neutrons moving radially into the core form a "thermal spike" in the fission density that is two to three times greater than the value at the core center. Localized control rod absorption in the reactor can be simulated by using a thin annular "curtain" of absorber in the reflector, so that relatively fast one-dimensional calculations can be used. The curtain (a thin shell of boron at the weighted mean center of the beryllium reflector) reproduces the mean peripheral fission density value for the actual two-dimensional reactor with its localized control rods. The absorption thickness ($\Sigma_a d$) is dependent on the thermalization model for graphite.⁹

In summary, neutron cross sections and transport codes can be used very satisfactorily for the evaluation of Rover reactors. Simple arithmetic errors, and their propagation, are the largest source of uncertainties. The many numerical and physical approximations and their small effects require the use of a reference absolute value of k_{eff} (no normalization) near unity, representing the barely critical reactor (k equaling 1.0000, and a constant flux level).

Examples of the effects of changes in neutron cross sections are meaningless outside of context and will not be discussed here. However, it has been found that for the Rover reactors the thermalization model for the reflector material does not affect k_{eff} to second order; nor does the model used for neutron thermalization in the reactor core, for reasons mentioned above. The resonance integral model is much less important

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than the microscopic capture-to-fission ratio, and any of several codes can be used, e.g. GAM, GAM-W, ARES, and ZUT.¹⁰ The code developed by Bell² and Devaney¹¹ for U^{235} and U^{238} resonance integrals was used primarily because it is very direct and agrees at least as well with experiment as any other. The ZUT code for isolated resonances seems to agree with experiment a little better than the others, and has the advantage of being very well described in the literature.³ The unresolved resonances must be studied further, because they must account for much of the difference between the experimental and computed changes in k_{eff} with reactor temperature found in high-power reactor experiments.

Neutron Cross Section Codes

Fast neutron multigroup cross sections are not generally produced by numerical codes. (Active work in this area is connected with a Master Data Tape being constructed as a single depository for the great volume of experimental data.¹²) Instead, the quasi-theoretical studies of G. Hansen and others provide tabulations of useful group-averaged cross section data for the higher energy region $E > 0.1$ Mev.^{13,14} Well known elements such as hydrogen and deuterium have been integrated directly into the system of calculations, using the tabulated values of $\sigma(E, \mu)$ and a simple flux weighting function specific to the reactor experiment. Expansion of the scattering probability into the first three terms of the Legendre polynomials and use of these terms in the DTF code has given good, essentially analytic results.¹⁵ Further work is continuing toward completely numerical production of energy-group-averaged neutron cross

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sections. Intermediate-energy neutron codes are largely directed toward the effect of nuclear resonance structures on reactor absorption and scattering. The basic ideas will be briefly reviewed here.

The reaction rate

$$R = \phi(E) \Sigma(E)$$

is proportional to the neutron flux, $\phi(E)$, and to the neutron macroscopic cross section, $\Sigma = N(\text{atom/cc}) \times \sigma(\text{cm}^2/\text{atom})$. The microscopic absorption cross section, σ_a , (to a good approximation for isolated resonances) is:

$$\sigma_a(E) = \pi \lambda^2 g \frac{\Gamma_n \Gamma_j}{\Gamma^2/4 + (E - E_0)^2}$$

where λ is the neutron wave length, and Γ is the resonance width centered at energy E_0 . The other terms are familiar. Inasmuch as the neutron flux at E comes from higher energies, the total reaction rate is:

$$R_t = \Sigma_t(E) \phi(E) = \int_E^{E/\alpha} \Sigma_s(E') \frac{dE'}{(1 - \alpha_m)E'}$$

for neutrons scattered from the moderator, m , with a similar term for other scattering sources. In the above expression, $\alpha = (A - 1)^2/(A + 1)^2$, and A is atomic weight. The absorption and scattering rate at E may be computed if $\phi(E)$ is known. Doppler motion of the scattering or absorbing nucleus broadens the resonance, so that

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$$\sigma(E) = \int_{-\infty}^{\infty} \frac{\sigma_{\max} dE'}{1 + \left(\frac{E' - E_0}{\Gamma/2} \right)^2} \left(\frac{A}{4\pi E k T} \right)^{\frac{1}{2}} \exp \left(- \frac{A(E' - E)^2}{4 E k T} \right)$$

where $\sigma_{\max} = 4\pi\lambda^2 g \Gamma_n / \Gamma$, and $\Gamma = \Gamma_n + \Gamma_\gamma$. The increase in resonance width, $\delta\Gamma$, is

$$\delta\Gamma = \left(\frac{4 E k T}{A} \right)^{\frac{1}{2}}$$

due to thermal motion of the nuclei, where these forms presume the nuclei to have a Maxwellian velocity distribution. A mixture of moderator and absorbers considerably complicates the analysis. For an excellent and detailed study, see Bell.² Isolated resonances are treated with the method developed by Nordheim.³

There are several thermal neutron-transfer matrix codes for the free atom model, but only one for the crystal model. General Atomic Codes SMT and THM use the Yoshimoi and Kitano frequency spectrum of graphite and the general time-dependent formulation of Glauber to calculate the scattering of slow neutrons by a system of chemically bound atoms.¹⁶ Neutron velocity distribution in the lowest energy groups of the transport equations is, to a first approximation, Maxwellian. Early studies based on this assumption gave excellent agreement for the integral value of k_{eff} for the Rover reactors. The effects of the actually very non-Maxwellian velocity distribution of scattering centers in the crystal lattice has been studied by Nelkin and associates.^{17,18} They find a significant effect

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on the temperature coefficient of reactivity resulting from the somewhat smaller number of low-energy neutrons in the crystal-model calculations, as compared to free-atom-model calculations, of thermal neutron flux. This results from the fact that the neutron may absorb and transmit significantly more energy from quantum states in the atomic lattice than the elastic scattering processes permit. The derivation of the energy-transfer rate processes starts with the Fermi pseudopotential approximation, which replaces the short-range potentials between neutrons and nuclei with point interactions. The interaction potential between a neutron at $\vec{r}_n$ and an atom at $\vec{r}'$ is:

$$Y(\vec{r}_n, \vec{r}') = (2\pi a/m) \delta(\vec{r}_n - \vec{r}')$$

where a is the essentially experimental bound-scattering length. On traversing a molecule the sum of the contributions to the potential is:

$$V(\vec{r}_n) = \sum_v Y(\vec{r}_n, \vec{r}_v) = \left(\frac{2\pi}{m}\right) \sum_v a_v \delta(\vec{r}_n - \vec{r}_v)$$

A collision that changes neutron momentum from k_0 to k , and a molecular transition from ψ_i to ψ_f (energies E_i and E_f) results in a differential cross section, using the first Born approximation, given by :

$$\sigma_{i,f}(\theta) = \left(\frac{m}{2\pi}\right)^2 \frac{k}{k_0} \left[\left\langle \psi_f \int \exp(-i(\vec{k} - \vec{k}_0) \cdot \vec{r}_n) \left| V(\vec{r}_n) d\tau_n \right| \psi_i \right\rangle \right]^2$$

that sums the transitions over all admissible final states of the molecule.

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Nelkin¹⁷ has reduced this equation to one that is tractable, and Parks⁴ has continued into the mechanics of neutron scattering in graphite. The differential scattering cross section for scattering from momentum $\vec{k}_0$ to $\vec{k}$ is:

$$\frac{d^2\sigma}{d\Omega dE} = \frac{k}{k_0} \frac{a^2}{2\pi} \int_{-\infty}^{\infty} dt e^{i\epsilon t} \chi(\vec{k}, t)$$

where a is the bound-atom scattering length; $\epsilon = \frac{1}{2}(k^2 - k_0^2)$ is the energy transfer; and $\vec{k} = \vec{k} - \vec{k}_0$ is the momentum transfer. The scattering dependence on the dynamics of atomic motions is contained in $\chi(k, t)$. The model used could be important for the Rover reactors, which need a proper rate of energy change for the thermal neutron entering the core from the reflector. For this reason, although the gas model has given results that are generally within the range of experimental errors, the General Atomic codes have been consistently used in recent studies.¹⁶

Another code specific to recent generalizations in the transport approximation, by Lathrop, computes the neutron flux due to anisotropic scattering for all groups.¹⁵ These effects are generally large for fast neutrons, and are significant at all energies for the lightest moderators, hydrogen and deuterium. Lathrop retains anisotropy by expanding $\sigma(E, \mu)$ in a series of Legendre polynomials, with DTF code acceptance of $l = 0, 1$ and 2 in P_l . There has not yet been sufficient experience to evaluate the anisotropic scattering model, in comparison to the previous isotropic scattering assumption corrected by $\bar{\mu} \sigma_s$ for the "transport" approximation.

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It is expected that differential experimental data, $\sigma(E, \mu)$, will eventually be used to obtain group-averaged cross sections.

Experiments

Critical experiment assemblies provide, in a sense, normalization points for any general computing scheme. In a "barely critical" reactor the ratio of the neutron inventory from one neutron generation to the next is equal to unity: $k_{\text{eff}} = 1.0000$. As was discussed above, k_{eff} must be computed for a wide variety of critical reactor assemblies and, consequently, for a wide variety of neutron flux spectra. It was assumed that if computed $k_{\text{eff}} = 1.000 \pm 0.005$ for the discrete assemblies, then both transport and cross section codes are, for theoretical and numerical approximations and for basic data, almost error-free; and, at the very least, are consistent. Clearly, many small physical effects and uncertainties may cancel. (And it is a favorite sport of reactor theorists to evaluate these small effects for magnitude.) A limited tabulation of critical experiments was published early in the Rover reactor program.^{19,20} A second tabulation, covering those experiments and a number of others, was made recently by Paxton.²¹ The first table was utilized in the systematic development of a "best" set of multigroup neutron cross sections.

The first of the Rover reactors, a graphite-moderated, beryllium-reflected assembly, was subjected to detailed study by Hoogterp and associates, who measured not only k_{eff} and fission density (versus position) but also cadmium fractions for indium and gold.²² These were shown to agree closely

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with one- and two-dimensional calculations, thereby providing a partly differential inspection of the neutron flux spectrum, $\varphi(E,r)$.²³ In addition, Hoogterp measured the peripheral fission density on the surface of the very roughly cylindrical surface of the critical experiment (actually fabricated of 3-in.-square building blocks). A comparison of two-dimensional calculations and experimental numbers showed that k_{eff} changed when converted, for computing purposes, from the "cylindrical" experiment to an area-weighted "equivalent cylinder", because of variation in fission peaking at the core periphery.

The next-stage Rover reactor had an axially oriented D_2O island for power flattening and control. The study of this system showed, again, that neutron cross sections consistent with the physical characteristics of simple D_2O -, beryllium-, or graphite-moderated experiments could be used for the quantitative analysis of a complex reactor having all three moderators present in the form of concentric cylinders.²⁴ Also, in addition to excellent fidelity in the case of room-temperature experiments, the calculations of high-temperature, high-power conditions predicted very well the radially flat power density that was found experimentally.²⁵

A third-stage study has been made, including a number of relatively computable, temperature-dependent critical assemblies of graphite and BeO , in order to study power distribution and control rod effects.^{26,27} This study depended only partially on the tabulations mentioned previously, because a wide variety of graphite and BeO experiments have been made with k_{eff} as a function of temperature.^{28,29,30} The primary effects of interest,

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such as neutron lifetime and azimuthal power density distribution (roughly $1 + 0.1 \sin \theta$), were those related to control rod position.

One further test is of interest, in connection with a comparison of computed and experimental fission densities: This experiment was performed to determine the effect on measured fission density of the 1- or 2-mil-thick foils used for activation comparison. Figure 3 shows the computed values (normalized at the reactor core center) and two sets of data, one with a fission counter and one with a 2-mil foil. The curious discrepancies at positions several mean free paths from the core boundary are significantly reduced when foil thickness is eliminated as a factor. Note, however, a significant discrepancy remaining at the core boundary that is not apparent in one-dimensional studies. This error illustrates resolution loss due to the time- and space-point interval limitations consistent with computer time and memory.

Later developments in the third-stage study of moderate reactor power (to 1000 megawatts) were followed closely by E. Plassmann,⁹ and, some of that work is included here in order to show fission density distributions and the nuclear part of the temperature coefficient of reactivity. Power distributions and temperature effects on k_{eff} can be computed from Plassmann's description of Kiwi-B4D and -E.^{31,32} The calculated Kiwi-B4D fission density distribution was essentially identical with the experimental results (see Fig. 4). The same quantitative result was found to be true for the B4E experiment (see Fig. 5). In addition, an effect not noticed experimentally was that neutron lifetime was affected

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Fig. 3 Fission density near the core boundary.

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Fig. 4 Fission density in Kiwi-B4D.

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Fig. 5 Fission density in Kiwi-B4E.

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somewhat by control rod position, as shown in Fig. 6, if the time absorption effect is ignored. In B4E, the nuclear effect on reactivity going from room temperature to a high-core-, low-reflector-temperature was a Δk equal to $-\$1.00$ (with the reactor having a constant geometry; only neutron cross sections changing; an initial temperature of 0.0255 ev overall; and final temperatures of 0.2 ev and 0.008 ev in the core and the reflector, respectively). This value is very similar to that found for B4D: calculations for B4D with an overall temperature of 0.2 ev in the reflector gave a change in Δk equal to $-\$1.15$.

All neutron cross sections used for reactor analyses to this point had been developed from the sets of simple experiments tabulated above. The excellent agreement between computed values and both room-temperature and high-temperature "flat power" experiments justifies the approach described in the preceding section.^{*,9}

Phoebus Studies

Experiments conducted on the next, higher power Rover reactor - designated "Phoebus"- were sufficiently detailed to warrant close study. Emphasis was placed on material value reactivity (ϕ/kg) as a function of position, and on the effects of a large variation in the amount of the metallic material (niobium) used to protect the core from corrosion. The

* Reports on a later series of Kiwi reactors show that the temperature effects on power distribution and criticality control for the high-power experiments agreed very well with computed predictions.

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Fig. 6 Neutron lifetime versus control rod position.

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experimental assemblies were very similar to those mentioned above, but had a somewhat larger radial dimension and a greater void fraction in the core.

The Phoebus experiments, carried out in a low-density aluminum lattice named Honeycomb, have been described in some detail in other reports.^{53,54} Impurities in the graphite were equivalent to less than 1 ppm boron; the microscopic absorption cross section of the aluminum (No. 1100) was increased 6.0% by 1/v impurities; the uranium (in the form of 0.002-in.-thick foils) contained 6.6% U^{238} ; and the niobium (0.005-in.-thick foils) was chemically pure. Two relatively large effects were not evaluated: fuel- and absorber-foil interaction, and the consequence on neutron flux, $N(r,u)$, of building blocks in the reactor lattice construction. (The core, core annulus, and reflector geometries were irregular cylindrical configurations made of 3-in.-rectangular parallelepipeds.)

The first set of experiments, described in Table IA, was directed toward determining the material value of the protective element, niobium. A second experiment described the actual reactor prototype (see Table IB). The resonance integral parameter, σ_p , and foil volume fraction may be found from the volumes and weights shown on the table. The irregular reactor volume was transformed into a 3-region cylinder by area-weighting. This technique is equivalent to moderator-volume-weighting and is a more simple and accurate way to account for geometry effects than most methods. This approach (rather than an "equivalent geometrical buckling" approximation) was validated by the results of one- and two-dimensional calculations. Axially, the assembly was uniform and geometrically simple.

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TABLE I
EXPERIMENTAL REACTOR COMPOSITIONS

A: Honeycomb Phoebus, HC-1

Atomic Density (10^{24} atoms/cc)	Core	Annulus	Reflector
U-235	variable		
U-238	variable		
C	0.05869	0.07333	
Al	0.00368	0.00368	0.00368
Nb	variable		
Be			0.117
Outer radius: (cm)	69.85	74.88	91.40
Length: 132 cm			

Variable Composition of Core							
Experimentation No.	1	2	3	4	5	6	HC-2
Oralloy (kg)	34.4	38.17	42.34	53.50	88.70	189.74	294.
Niobium (kg)	0.0	27.76	54.72	109.66	219.79	444.56	361.1
U-235 (per cc $\times 10^{19}$)	4.06	4.50	5.00	6.32	10.47	22.39	34.7
σ_p with foil interaction:							
U-235	1515	1462	1413	1294	1016	629	593
Nb	0	1255	887	559	321	160.7	146

B: Honeycomb Phoebus, HC-2

Atomic Density (10^{24} atoms/cc)	Core	Annulus	Reflector
U-235	0.000347		
U-238	0.0000245		
C	0.05868		
Al	0.0	0.03616	0.00363
Nb	0.001158		
Be			0.1117
Outer radius: (cm)	70.31	77.86	96.89
Length: 132 cm			

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Analysis

The analysis of a critical assembly having a fixed geometry is concerned with variables in the "method of approximation", rather than with the effects of radii and densities, factors that are studied before an experimental assembly is constructed. In the cases of interest here, five analytical procedures were carried out:

First, it was necessary to estimate the foil resonance integrals (including the large foil-thickness and interaction effects) and the dimensions and atomic densities of the assemblies (by using reported weights and measurements) and then to evaluate various theoretical assumptions made in the computation of neutron cross sections. The transport code is well known from earlier studies.⁸

Second, reactivity effects, in terms of k_{eff} vs $N(\text{Nb})$, were computed. Results were evaluated in the light of resonance-parameter and neutron-cross-section-model uncertainties in the codes used for the scattering matrix.

Third, the effects of geometry on the absolute value of k_{eff} were computed, using the two-dimensional (x,y) DDK code.

Fourth, the spatial distribution of reactivity was computed in order to help rationalize errors.

Fifth, the computed fission density distribution was compared with the experimental value.

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Reactivity as a Function of Composition and Approximations

Absolute values of k_{eff} for the graphite/niobium ratio experiments, outlined in Table IA, are shown in Table II below. Note that the table shows a small but significant systematic error. The two- to one-dimensional geometry transformation effects are evaluated briefly in the following discussions.

TABLE II

VALUES OF k_{eff} VS NIOBIUM IN THE PHOEBUS REACTOR CORE

<u>Experiment No.</u>	<u>Niobium (kg)</u>	<u>k_{eff}</u> [*]
HC-1-1	0	1.015
-2	28	1.011
-3	55	1.012
-4	110	1.007
-5	220	1.002
-6	445	1.007
HC-2	361	1.014

* Computed; no correction for internal corners of assemblies.

The second set of experiments, listed in Table IB, measured value (in terms of ϕ/kg) as a function of the position of the three major structural materials: graphite, niobium, and Orallo (93.4% U^{235}). The experimental measurements were made in terms of reactor period, so that the natural unit of delayed neutron value (ϕ) was equal to 100 times the value of the delayed neutrons; the determinations consisted of measuring the effects

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of removing and adding small amounts of material in relatively localized positions. On the other hand, the calculations compared values of k_{eff} during small, uniform changes in reactor material over a radial interval of the cylindrical geometry. Calculations of Δk_{eff} gave results directly comparable to the experiment:

$$\text{Reactivity } (\$/\text{kg}) = \frac{100 \times \Delta k_{\text{calc}}}{\Delta k_{\text{del}}} / \Delta M$$

where Δk_{del} is the delayed neutron group, and ΔM is kilograms of material removed.

The value of the delayed neutrons (in terms of Δk_{del}) is a function of reactor design. A reactor with a very small fast neutron escape has $\Delta k_{\text{del}} \approx 0.0065$; whereas the Rover reactors have a relatively large fast neutron escape, so that the more favorably placed delayed neutrons may have Δk_{del} values ranging from 0.007 to 0.008. Tables III and IV, on pages 36 and 37, respectively, summarize the results of computed and experimental reactivity values. Differences shown are not simple errors, because of the relatively simple method used for the calculations. To faithfully reproduce the experiment, a two-dimensional (x,y) calculation would be required.

The effect of one- versus two-dimensional geometry, for the same radial intervals, cross sections, and numerical approximations in the computing DTF and DDF codes, was to decrease k_{eff} by about 2% for the first Honeycomb experiments. This is the largest single correction. For the second set (HC-2) the Δk resulting from this change was 0%, due to much greater experimental symmetry.

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TABLE III
MATERIAL REACTIVITY IN THE PHOEBUS REACTOR CORE

<u>Experiment</u>	<u>Element</u>	<u>Value (#/kg)</u>		
		<u>Calculated</u>	<u>Experimental</u>	<u>Error(%)</u>
HC-1-1	C	4.0*	4.9	-20
	Nb	-7.7	-5.9	+30
	U	42.	39.3	+6
HC-1-6	C	2.6	2.9	-12
	Nb	-3.0*	-2.5	+20
	U	5.9	6.3	-6
HC-2	C	2.7	3.0	-8
	Nb	-2.74	-2.55	+7
	U	6.00*	7.34	-18
	Al	0.57	0.73	-20
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* This error is consistent with an experimental doubling of local niobium concentration and halving of uranium concentration.

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TABLE IV

REACTIVITY VERSUS PHOEBUS CORE RADIUS

<u>Element</u>	<u>Radial Interval (in.)</u>	<u>Material Value (ϕ/kg)</u>	
		<u>Calculated</u>	<u>Experimental</u>
C	0-7	4.5	4.9
U	0-7	10.0	11.3
C	12-15	4.0	3.9
U	12-15	8.0	8.5
C*	27-27.7	0.91	1.35
U*	27-27.7	10.2	6.5

*The reference experiment shows a very large effect on reactivity due to uranium concentration near the core boundary. The calculated values are consistent with the smaller uranium concentrations.

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The effects of a close approximation to reactor composition when computing the thermalization effects on graphite cross sections ranged from -0.0093 for the high graphite/uranium configuration (HC-1-1) to -0.0005 for the low ratio (HC-1-6), as compared to an infinite clean system.

The effects of changing the niobium foil thickness from 0.0025 to 0.0050 in. were 0.00035 for the high graphite/uranium ratio (HC-1-2) and 0.0007 for the low (HC-1-6), using the Nordheim resonance integral method. The total cost of niobium for the HC-2 experiment was $\Delta k = 0.0735$ or about \$9.73, consistent with a change in critical mass of a factor of 5.5.

The cost of $1/v$ absorber in core and graphite annulus was less than \$1 for 3 ppm boron equivalent. No correction was made for impurities because the maximum value was 1 ppm, and it should be even less.

The effects on the k_{eff} of the Phoebus prototype, HC-2, of different methods for determining the resonance integrals of U^{235} and Nb are as follows: The GAM-W code for U^{235} gave a k_{eff} less by 0.023 than the k_{eff} value calculated with the code written by Bell and Devaney. The GAM-W code for niobium gave a value of k_{eff} greater by 0.012 than that computed by the Nordheim code, ZUT. Inasmuch as the values of the resonance integrals for constant flux weighting were nearly equal, the differences in the experimental values were due to the structure of the neutron flux spectrum (see Fig. 2). The effect of the unresolved resonances of the aluminum structural shell in HC-2 was to decrease k_{eff} by 0.012, a significant change.

Two effects resulted from the use of a minimal number of energy (or lethargy) groups:

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1) A change in delayed neutron value from $\Delta k = 0.00710$, for all delayed neutrons in the interval 0.4 to 0.9 Mev, to $\Delta k = 0.00731$ when a more exact correspondence to energy values of the delayed neutrons was used; and

2) A change in fission density (per g U) at the core boundary of 4% with reference to the reactor core center, consistent with a change in k_{eff} by $\Delta k = +1\%$, when the total number of groups below 0.1 Mev was halved. This result, compared to that on page 16, illustrates the relative simplicity that can be achieved by careful flux weighting of the neutron cross sections.

The errors in calculated reactivity, as compared to the experimental data, reflect: 1) an effective lack of identity between computed symmetrization and experiment, 2) incorrect resonance integrals, 3) localized changes in resonance self-shielding not effectively shown by the homogenized calculation, and 4) theoretical and experimental errors connected with overall descriptions of the experiments. The most probable causes of the above errors will be briefly evaluated:

Table II, k_{eff} versus niobium in the Phoebus reactor core, strongly indicates that items 2 and 4 above do not contribute large errors. The integral reactivity is very nearly unity, and only small systematic errors in k_{eff} appear for large niobium and U^{235} concentrations.

Table III, material reactivity in the Phoebus core, shows large absolute errors; but errors for HC-1-1 and HC-1-6 are of nearly the same magnitude, independent of large changes in critical mass (a factor of

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5.5) and in the quantity of niobium (none to 444 kg). Also, Table IV, reactivity versus Phoebus core radius, does not show central sensitivity to position; the error is roughly constant until the core periphery is approached. Accordingly, items 2, 3, and 4 could not cause the errors.

However, Table IV does show a dramatic increase in error as the core boundary is approached. This situation has been reported in the experimental studies, which showed a change in U^{235} reactivity by a factor of more than two with a change in Orallo concentration at the core boundary.³⁵ Accordingly, the errors shown in Table III must be due to item 1, lack of identity between experimental and theoretical descriptions.

These considerations are made less objective by uncertainties in experimental data for unresolved absorption resonances. Earlier niobium data resulted in a computed reactivity value 30% too low (rather than 30% too high) for HC-1-1, 11% too high (rather than 20% too high) for HC-1-6, and 5% too low (rather than 7% too high) for HC-2. Again the same effects were found for uranium, with changes from 2% too low for HC-1-1 (rather than 6% too high), 0% for HC-1-6 (rather than 6% too low), and 15% too low for HC-2 (rather than 18% too low). From these changes we see that the average correlation change is 18% for niobium and 6% for uranium. This is a measure of the uncertainty in cross sections, rather than an indication of the uncertainty of the description of the experiment for computation.

A further study of the experiments was made to evaluate the integral reactivity value k_{eff} , and to determine the temperature coefficients,

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which integrate resonance, transport, and thermal expansion effects. As k_{eff} is the integral of the fission density (per unit fission neutron), this may be inspected via the power distribution. Fig. 7 compares the experimental and computed fission density. No obvious discrepancy appears on this figure to explain the above reactivity errors. The effect of reduced fission density resulting from a varied fuel loading is shown in Fig. 8. This figure also indicates a reasonably satisfactory correspondence between theory and experiment, although not as close as that in Fig. 7, partly because of data scatter. It might be asserted that the graphite is less effective at scattering thermal neutrons (the computed fission density being to the left of the experimental values in Fig. 8); and the reactivity values in Table IV also show a too-small total effect. However, k_{eff} should then be seriously in error.

The general conclusions might be, then, that the experimental measurements and the calculations are both too integral in effect, so that the specific cause of the differences, other than σ_p changes with the experimental measurement, are too well concealed; and that there is no large error in theoretical methods. If other studies of the Rover reactors are given some consideration, in which the calculated integral value k_{eff} , power density, and temperature coefficients agreed very well with experimental data, it must be concluded that either a more heterogeneous (three-dimensional) calculation or a more homogeneous experimental perturbation is necessary to explore the causes of the differences shown above. A review of the more recent calculations of the complex Kiwi and

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Fig. 7 Fission density in HC-2.

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Fig. 8 Fission density in HC-2, revised.

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Phoebus critical assemblies illustrates the point that reactivity measurements are not sufficient to demonstrate errors in cross section data or in their reduction to group-averaged integrals. The values of k_{eff} for the assemblies are: B4D, 0.9995; B4E, 0.9946; HC-1-1 through -6, 1.009 ± 0.006 ; HC-2, 1.014; and ZEPO (Phoebus) 1.001. The HC errors are certainly connected with internal details of the experiment, so an indication of the average absolute error obtained from the other assemblies would be the value $\Delta k \simeq \pm 0.0015$. This may be compared to a niobium cost of $\Delta k \simeq 0.07$. In general, the agreement is considered to be within the experimental and computational uncertainties; therefore, a closer study, using two-dimensional computing methods, was not made. The assemblies are large and complex; in order to attain more quantitative reactivity studies, simpler and more homogeneous assemblies are necessary.

Prompt Neutron Lifetime

The DIK code provides a very simple and direct method of computing the reactor kinetic constant α , in the expression $N = N_0 \exp(\alpha t)$. Because α , as computed, relates the neutron flux, N , to the prompt neutron excess per generation, Δk_{ex} , the prompt neutron lifetime, l , is equal to $\Delta k_{\text{ex}}/\alpha$. The prompt neutron lifetime for the HC-2 assembly described above was 40.2 μsec . No experimental measurements have been made of this value in the Phoebus reactor assembly. However, the Kiwi-B4E reactor had a measured value of 30 μsec , in good agreement with the computed value of 31 μsec .

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High-Temperature Studies

There is generally good agreement between experimental and computed values of fission density distribution and temperature coefficients for very high-temperature reactor operations. Kiwi-A fission density distribution was flat at high temperature,²⁵ as was that in the Kiwi-B experiments.⁹ Also, the temperature coefficient of reactivity at high power and temperature, an experimental number very hard to separate into its component parts, was at least near the small negative value computed.⁹ Experiments specifically directed toward high-temperature neutronics are now under study (see Kiwi-TNT reports, LASL³⁶).

Summary

The theoretical analysis of Rover reactors, using neutron transport numerical codes and the best neutron cross section data, continues to be troublesome, because of uncertainties in neutron cross section data, as well as because of some fluctuation in experimental data. The absolute values of the integral fission density and the effective neutron multiplication constant (k_{eff}) agree with experiment, within the experimental error; and the prediction of high-temperature reactor performance agrees very well with experimental results. As a result, the differences found in reactivity values, experimental and theoretical, are considered to be due to both experimental and theoretical (largely reactor experiment simplification) errors, and will be the subject of continuing detailed investigations.

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