PCTR INTEGRAL MEASUREMENTS
OF ^{239}Pu ALPHA (α)
IN FAST SPECTRA

D. D. Lanning G. J. Busselman

C. L. Bennett D. F. Newman S. R. Bierman

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D. D. Lanning
G. J. Busselman
C. L. Bennett
D. F. Newman
S. R. Bierman

Reactor Physics Department
Battelle-Northwest
Richland, Washington

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INTRODUCTION

Accurate integral values of the capture-to-fission cross section ratio ($\bar{\alpha}$) for ^{239}Pu can be measured in the Physical Constants Test Reactor (PCTR)⁽¹⁾ as a function of the carbon-to-uranium ratio in plutonium oxide-uranium oxide-carbon mixtures. These mixtures are to be prepared with plutonium-to-uranium ratios such that the neutron multiplication factor of an infinite system (k_{∞}) is very close to unity.

The range of carbon-to-uranium (C/U) atom ratios to be studied will vary the neutron spectrum from a mostly fast reactor system, where $\bar{\alpha}$ is believed to be well known, down into the intermediate spectrum where $\bar{\alpha}$ is believed to be more uncertain. Calculations indicate that this can be accomplished by making measurements in the C/U range of zero to about 25. The percent of plutonium in the UO_2 plus PuO_2 mixtures varies from about 8% to approximately 14% over this range of C/U for mixtures with k_{∞} equal to unity.

A total of four to six measurements are required, together with differential perturbation measurements at each point to enhance the accuracy. The expected results include the derived values of $\bar{\alpha}$ and $\bar{\eta}$ for ^{239}Pu , where $\bar{\eta}$ is the integral value of the number of neutrons produced per neutron absorbed in ^{239}Pu . Values of reaction rate ratios (^{238}U absorption, ^{238}U fission, ^{235}U fission, and ^{239}Pu fission) will be measured, together with the standard PCTR measurements of k_{∞} and the breeding ratio. Available techniques for neutron spectral indices such as integral foil measurements will also be used.

These initial PCTR experiments can be extended to greater C/U atom ratios if there is a gap between the present BNL exponential studies on η of ^{239}Pu in D_2O systems and these proposed C/U experiments. The calculated values of the fraction of plutonium in uranium to keep k_∞ equal to unity for the C/U range of 0 to 25 increased with increasing C/U. Obviously, as the system becomes more thermal, this fraction of plutonium will begin to decrease. A study of the region of inflection between mostly fast and mostly thermal systems will add greatly to the reactor physics information and will be of value for theory-experiment correlations. Such systems are a simple extension of the initial experiments described above.

DISCUSSION OF THE PROBLEM

The breeding gain of large fast reactors is dependent upon the number of excess neutrons produced per neutron absorbed in the fissionable isotopes (i.e., $\eta-1$). Some of these excess neutrons are then available for utilization in production of more fissionable isotopes. The value of η thus becomes important as a figure of merit in predicting the breeding ability of a reactor system.

Recent differential experiments by the British⁽²⁾ SM101/41 Karlsruhe Conference, and a new careful analysis of available integral data⁽³⁾ seem to indicate that the value of η for ^{239}Pu is lower than predicted by most cross section sets now being used in the U.S. for reactor designs. In particular, the value of the capture-to-fission cross section ratio $\alpha(E)$ over the energy range from about 10^2 to $4(10)^4$ ev seems to be much higher than the earlier accepted values.

The region between 10^2 and $4(10)^4$ eV presents a special problem for integral measurements related to cross section evaluations. For a thermal reactor this region of the "1/E" tail contains only a small fraction of the total epithermal flux. Also, for most fast reactors, this low energy region again contains only a small fraction of the total flux. To measure integral parameters related to this particular energy region then requires systems with a "tailored" neutron energy distribution. A possible method of obtaining the desired distribution is shown in Figure 1. A pure fissile-carbon system tends to have a peak flux around 1 MeV and addition of more carbon spreads the low energy flux out but does not eliminate the peak, as indicated by the C/ ^{235}U calculations from the GAM-I document (Figure 2). However, the addition of large amounts of ^{238}U with its inelastic scattering does reduce the 1 MeV peak so that the bulk of the neutron flux can be shifted to center about lower energies as shown in Figure 1 with the $^{238}\text{U}/^{239}\text{Pu}$ and carbon cases. (These distributions were calculated by using the 68 group HRG computer code.) The addition of UO_2 not only reduces the average neutron energy, but is also consistent with reactor fuels of interest.

Consideration was given to various possible integral measurements, and it has been concluded that measurements can be made within the accuracy desired if an "equilibrium" spectrum is obtained using plutonium. If the equilibrium spectrum is developed using a ^{235}U fuel, the reactivity coefficient of a small sample of ^{239}Pu fuel depends not only on $\bar{\eta}(\text{Pu})$, but also depends strongly upon the calculated interpretation involving such quantities as the adjoint flux. However, the value of $\bar{\eta}(\text{Pu})$ derived from the standard PCTR technique for materials with k_∞ near unity does not depend on knowledge of

FIGURE 1
Flux Distributions for Various Combinations of ^{239}Pu , ^{238}U , and Carbon

FIGURE 2
Total Lethargy Flux Versus Lethargy for Several C:U²³⁵ Atom Ratios

the adjoint flux and will provide the desired accuracy. Use of this technique will involve more fuel than the usual PCTR experiment since the system to be studied is more closely related to fast reactors where the fuel cross sections are low. For these experiments, the required fuel is estimated to range from 40 to 100 kg of plutonium

The proposed PCTR experiments are quite similar to a recent British experiment in the DIMPLE Reactor, "The Measurement of ^{239}Pu Capture to Fission Ratio in Fast Reactor Lattice," W. N. Fox et al,⁽⁴⁾ in which the techniques described in earlier PCTR proposals to the AEC have been directly applied. This fast-thermal experiment utilizes all of the principle concepts of the PCTR technique. It is not proposed to repeat the British experiment but to extend the integral measurement of α into other energy spectra as suggested by the curves in Figure 1 and the plutonium absorption rate energy distribution curves shown in Figure 13.

The experiments proposed here consist of using 8 to 14% PuO_2 (of low ^{240}Pu) in UO_2 mixed with graphite. The mixture will be loaded into 2 foot long hexagonal tubes (about 1-3/4" across the flats) in as simple a manner as possible. Fabrication estimates indicate that 60% density will be obtained. These fuel elements will be loaded into the Fast Neutron Cavity of the PCTR (see Figure 8). The value of k_{∞} will be very close to unity and the deviation from unity can be determined very accurately. Further, the major reaction rates such as ^{239}Pu , ^{235}U , and ^{238}U fission and ^{238}U capture, can be measured by foil activation techniques that are a standard part of the PCTR measurements.

W. N. Fox et al, in their analysis of the DIMPLe Reactor Experiment, describe the total evaluation of all reaction rates in detail and they conclude that the value of α measured by this technique can be determined to an accuracy of $\pm 10\%$. The use of the PCTR thermal column may, in some cases, help to improve the accuracy of the reaction ratio measurement. An analysis of the PCTR experiments indicates that the value of α^{49} can be derived within an uncertainty of less than ± 0.06 . The derivation of this uncertainty is shown in Table I and Table 2. The value of the uncertainty (i.e., less than ± 0.06) is about constant over the range of experiments.

The accuracy of these experiments is sufficient to distinguish between the earlier values of α used in the U.S. and the more recent British data which indicates higher values of α . Figure 3 shows a calculated plot of the integral value, $\bar{\alpha}(^{239}\text{Pu})$, as a function of the C/U ratio for k_{∞} near to unity. The plot has been made by using the original data for α (see Figure 4). By using the recent British data over a range of 10^2 to $4(10)^4$ ev (see Figure 5) the percent change in α was derived as shown in Figure 6 as a function of the C/U ratio. Also shown in Figure 6 is the change in k_{∞} caused by this change in $\bar{\alpha}$. The values of k_{∞} near unity are usually measured to less than ± 0.003 for PCTR experiments. Hence, the change in k_{∞} caused by the difference in $\bar{\alpha}$ is more than ten times the experimental uncertainties in the measured value of k_{∞} . The additional uncertainty in the measured reaction rates for the evaluation of $\bar{\alpha}$ makes the comparison to the calculated change somewhat smaller but still a factor of about 3 or 4 between the predicted change of about 40% in $\bar{\alpha}$ and the expected uncertainty in the experimental $\bar{\alpha}$ of about $\pm 10\%$ at a C/U of 6.

TABLE I

Reaction Rate Ratios and Expected Uncertainties in the
Evaluation of $\bar{\alpha}$ for ^{239}Pu at a C/U Atom Ratio of 6.6

C/U = 6.643; 0.2% depleted uranium - 10.144 at% Pu fuel ($\text{UO}_2\text{-PuO}_2$)

<u>Quantity</u>					
Measured Terms	Value	Realistic Uncertainty	Uncertainty Contribution in $\bar{\alpha}^{239}$	Conservative Uncertainty	Uncertainty Contribution in $\bar{\alpha}^{239}$
C^8/F^9	1.509	$\pm 1.5\%$	$\pm .023$	$\pm 2.0\%$	$\pm .030$
F^8/F^9	0.147	$\pm 2.5\%$	$\pm .006$	$\pm 3.0\%$	$\pm .007$
F^5/F^9	0.025	$\pm 1.5\%$	$\pm .001$	$\pm 2.0\%$	$\pm .001$
k_{∞}	1.028	$\pm 0.5\%$	$\pm .011$	$\pm 0.07\%$	$\pm .015$
$\frac{\text{C}^{\text{Fe}} + \text{C}^{\text{Ni}} + \text{C}^{\text{Cr}}}{\text{F}^9}$	0.107	$\pm 10\%$	$\pm .011$	$\pm 15\%$	$\pm .016$
<u>Nuclear Data:</u>					
^{239}v	2.900	$\pm 0.5\%$	$\pm .015$	$\pm 1.0\%$	$\pm .020$
^{238}v	2.754	$\pm 1.0\%$	$\pm .004$	$\pm 1.5\%$	$\pm .006$
^{235}v	2.441	$\pm 0.5\%$	$\pm .0003$	$\pm 1.0\%$	$\pm .001$
^{240}v	2.993	$\pm 1.0\%$	$\pm .0003$	$\pm 1.5\%$	$\pm .0004$
<u>Calculated Ratios:</u>					
C^5/F^9	0.010	$\pm 20\%$	$\pm .002$	$\pm 25\%$	$\pm .002$
C^0/F^9 (8% ^{240}Pu)	0.076	$\pm 20\%$	$\pm .015$	$\pm 25\%$	$\pm .019$
$\frac{\text{C}^{\text{carbon}} + \text{C}^{\text{oxygen}}}{\text{F}^9}$	0.006	$\pm 20\%$	$\pm .001$	$\pm 25\%$	$\pm .002$
F^0/F^9	0.009	$\pm 20\%$	$\pm .004$	$\pm 25\%$	$\pm .005$
$\bar{\alpha}^{239} = 0.413$ Total: $\pm .035$ Total: $\pm .052$					

TABLE II

Reaction Rate Ratios and Expected Uncertainties in the
Evaluation of $\bar{\alpha}$ for ^{239}Pu at a C/U Atom Ratio of 2.0

C/U = 2.0 atom ratio; 0.2% depleted uranium - 8.0 at% Pu fuel

<u>Quantity</u>					
Measured Terms	Value	Realistic Uncertainty	Uncertainty Contribution in $\bar{\alpha}^{239}$	Conservative Uncertainty	Uncertainty Contribution in $\bar{\alpha}^{239}$
C^8/F^9	1.963	$\pm 1.5\%$	$\pm .029$	$\pm 2.0\%$	$\pm .039$
F^8/F^9	0.237	$\pm 2.5\%$	$\pm .011$	$\pm 3.0\%$	$\pm .013$
F^5/F^9	0.036	$\pm 1.5\%$	$\pm .001$	$\pm 2.0\%$	$\pm .001$
k_{∞}	0.978	$\pm 0.5\%$	$\pm .009$	$\pm 0.7\%$	$\pm .013$
$\frac{\text{C}^{\text{Fe}} + \text{C}^{\text{Ni}} + \text{C}^{\text{Cr}}}{\text{F}^9}$	0.100	$\pm 10\%$	$\pm .010$	$\pm 15\%$	$\pm .015$
<u>Nuclear Data:</u>					
^{239}v	2.909	$\pm 0.5\%$	$\pm .015$	$\pm 1.0\%$	$\pm .029$
^{238}v	2.765	$\pm 1.0\%$	$\pm .007$	$\pm 1.5\%$	$\pm .010$
^{235}v	2.445	$\pm 0.5\%$	$\pm .0004$	$\pm 1.0\%$	$\pm .001$
^{240}v	2.995	$\pm 1.0\%$	$\pm .0004$	$\pm 1.5\%$	$\pm .001$
<u>Calculated Ratios</u>					
C^5/F^9	0.013	$\pm 20\%$	$\pm .003$	$\pm 25\%$	$\pm .003$
C^0/F^9 (8% ^{240}Pu)	0.054	$\pm 20\%$	$\pm .011$	$\pm 25\%$	$\pm .014$
$\frac{\text{C}^{\text{carbon}} + \text{C}^{\text{oxygen}}}{\text{F}^9}$	0.010	$\pm 20\%$	$\pm .002$	$\pm 25\%$	$\pm .003$
F^0/F^9	0.012	$\pm 20\%$	$\pm .005$	$\pm 25\%$	$\pm .006$
$\bar{\alpha}^{239} = 0.345$ Total: $\pm .040$ Total: $\pm .057$

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FIGURE 3

Integral Value of $\bar{\alpha}$ for ^{239}Pu for k_{∞} of Unity in Pu-U-C Systems as a Function of Carbon to Depleted Uranium Atom Ratio. (See Figure 4 for Differential α^9 Used.)

FIGURE 4
Values of α for ^{239}Pu from 68-Group HRG

FIGURE 5
Differential Values of α for ^{239}Pu as a Function of Neutron Energy

FIGURE 6
Effect of Changing α for ^{239}Pu from Figure 4 to the Values
in Figure 5 for Various C/U Atom Ratios

DESCRIPTION OF WORK

The available Fast Neutron Cavity (FNC) in the PCTR is to be used to provide the proper neutron energy distribution incident upon a central test sample such that accurate values of the PuO_2 to UO_2 ratio can be determined for k_∞ equal to unity at a given carbon-to-uranium ratio. Figure 7 is a schematic of the PCTR and Figure 8 shows the FNC loaded into the core of the PCTR. The FNC outer region contains natural and enriched uranium to convert the incident thermal neutron spectrum into a typical fast spectrum. Inside the FNC the core material is loaded in stainless steel hexagonal cans. These hexagonal cans are 1-3/4" across the flats and 2 feet long. (At a C/U of about 6, each can weighs a little less than 10 pounds.) The system of hexagonal cans prevents neutron streaming and is self-supporting. The central seven can array is removable through a port in the FNC. These central cans are made in three sections such that the center section can be left out for the null reactivity measurement - that is, the comparison of the central can reactivity worth to the worth of a void. If $(k_\infty - 1)$ is zero, the measured reactivity change is zero when the neutron spectrum at the central test cans is in proper equilibrium. Calculations indicate that about 163 cans of this fuel at a C/U of 6, combined with a first attempt at a reasonable adjustment of the outer buffer uranium, is sufficient to provide the proper spectrum within $\pm 2\%$. (163 elements at a C/U of 6 and 60% of theoretical density contains about 40 kg of plutonium.) These 12 group integral transport theory calculations (DTFIV one dimensional computer calculations using group averaged cross sections from HRG and THERMOS) were made on a spherical model for the entire PCTR, including the fast core, the buffer region, the thermal drivers, the control rods, and the graphite reflector. The results are in agreement with the earlier 26 group

FIGURE 7
Schematic View of PCTR

FIGURE 8A

Fast Neutron Cavity Assembled in the PCTR
Showing Removable Uranium Buffer Plug

FIGURE 8B

Cutaway Diagram of Fast Neutron Cavity in the
PCTR showing Installation of the Test Core

calculations⁽⁵⁾ that indicated a satisfactory spectral equilibrium could be obtained on the central test sample. Optimization of the buffer region will reduce the error in the flux match. However, even at a +2% flux match, when combined with the calculated +3 to 4% adjoint flux match, it is estimated that the overall uncertainty in the measured value of k_{∞} will be about +0.003 as discussed above. Detailed foil traverses, for various outer FNC loadings, will be made in the core of the FNC to establish the flux equilibrium conditions. The standard PCTR system of a null activity measurement is then used to establish the proper Pu to U ratio to make k_{∞} equal to unity.

As a check on the calculations an independent series of computations were made by S. R. Bierman to determine the flux distributions in the PCTR. These computations are in agreement with the above results. Appendix A is a description of Mr. Bierman's calculations and results.

In addition to the fundamental results of the composition to make k_{∞} equal to unity, all of the measurable reaction rates involved will be determined by foil activation techniques normalized to a similar foil in the PCTR thermal column. The most important reaction rates that will be measured are:
 (1) ^{238}U absorption; (2) ^{238}U fission; (3) ^{235}U fission; (4) ^{239}Pu fission; and possibly (5) total fission in all Pu isotopes. These measured reaction rates cover most of the important reaction rates that occur in the system. All other reaction rates except ^{239}Pu capture are very small and therefore can be calculated with sufficient accuracy for the analysis of these experiments.

Briefly, the analysis can be viewed as follows:

$$k_{\infty} = \frac{(\nu F)^5 + (\nu F)^8 + (\nu F)^9 + (\nu F)^0 + (\nu F)^1}{A^5 + A^8 + A^9 + A^0 + A^1 + C^{ss}} \quad (1)$$

where superscripts 5, 8, 9, 0, 1, and ss stand for the isotopes ^{235}U , ^{238}U , ^{239}Pu , ^{240}Pu , ^{241}Pu , and stainless steel. The symbols ν , F , A , and C stand for the number of neutrons produced per fission, fission rate, absorption rate, and capture rate. For isotopes with both fission and capture, the absorption rate becomes $A = F + C = (1 + \bar{\alpha})F$, and η is defined as:

$$\eta = \frac{\bar{\nu}}{1 + \bar{\alpha}} \quad (2)$$

Manipulating equation (1) to solve for α of ^{239}Pu gives:

$$\frac{\bar{\alpha}^9}{F^9} = \frac{C^9}{F^9} = \sum_{i \neq 9} \frac{F^i}{F^9} \left(\frac{\nu^i}{k_\infty} - 1 \right) - \sum_{i \neq 9} \frac{C^i}{F^9} + \left(\frac{\nu^9}{k_\infty} - 1 \right) \quad (3)$$

Also, the value of $\bar{\eta}^9 - 1$ becomes:

$$\frac{\bar{\eta}^9 - 1}{\bar{\eta}^9} = \frac{\sum_{i \neq 9} \frac{F^i}{F^9} \left(1 - \frac{\bar{\nu}^i}{k_\infty} \right) + \sum_{i \neq 9} \frac{C^i}{F^9}}{\bar{\nu}^9} + \frac{k_\infty - 1}{k_\infty} \quad (4)$$

The value of k_∞ will be very close to unity and known to within a few milli-k. The value of $\bar{\nu}^9$ is known well within the accuracy that C^8/F^9 and F^8/F^9 will be determined (± 2 to 3%). Thus, the values of the ^{239}Pu capture-to-fission ratio $\bar{\alpha}$ and the fraction of excess neutrons from ^{239}Pu ($\frac{\bar{\eta}^9 - 1}{\bar{\eta}^9}$) can be derived relatively accurately. Analysis of this experiment as shown in Tables 1 and 2, based upon previous experience of measuring the reaction ratios, indicates that the uncertainty in the value of $\bar{\alpha}^9$ will be less than ± 0.06 . Repeated measurements of the reaction rate ratios may reduce these uncertainties by a factor of 1.5 or more.

As noted previously, this accuracy is about $\pm 10\%$ in $\bar{\alpha}$ at a C/U of 6, and this is 3 or 4 times smaller than the difference between the earlier calculations of $\bar{\alpha}$ and the value calculated by using the more recent results, as shown in Figure 6.

In the design of a particular series of PCTR experiments, more detailed calculations have been made for the infinite medium neutron spectra, reaction rates, and k_{∞} values with fuels that have been reviewed for fabrication methods and costs. Further, a set of calculations has been made to determine the operating characteristics of the PCTR with these fuels inserted into the FNC. The results of this latter set of calculations have been used to determine the amount of fuel necessary to bring the neutron flux as a function of energy into the infinite medium equilibrium value at the central test sample.

The results of the infinite media calculations are shown in Figures 9-15. Figure 9 shows the neutron spectra for a C/U atom ratio of about 6 with the addition of oxygen and stainless steel. Comparison with the curves of Figure 1 show the same effect of lowering the energy of the peak flux but additional variations in the spectrum are found due to the resonance interactions in the oxygen and steel.

A parametric study was made to look at the effect on k_{∞} of changing the C/U and the enrichment of ^{239}Pu . These results are shown in Figure 10 and a cross plot is shown in Figure 11 of the calculated values of percent ^{239}Pu in depleted (0.2% ^{235}U) uranium required to keep k_{∞} equal to unity for various C/U ratios. Note also, on Figure 10 are points near k_{∞} of unity, for each C/U ratio, for which the value of α^9 was raised to the British value in the energy range of 100ev to 10^4 ev. The change in k_{∞} at a C/U of 6 is greater

FIGURE 9
 Flux Distribution for $\text{PuO}_2\text{-UO}_2\text{-Carbon}$ System in Stainless Steel Cans
 $\text{C/U} = 6.6$; 10% PuO_2 in $\text{PuO}_2\text{-UO}_2$ Mixtures

FIGURE 10
Parametric Study of k_{∞} for Various PuO_2 Enrichments in UO_2
and Various Carbon-to-Uranium Atom Ratios

FIGURE 11
% PuO₂ to Make $k_{\infty}=1.00$ for Various C/U Ratios (A Cross Plot
of Figure 10)

FIGURE 12
Median ^{239}Pu Fission Energy for Various C/U Atom Ratios at
 $k_{\infty} \approx 1.00$

FIGURE 13
Absorption in ^{239}Pu as a Function of Neutron Energy for Various C/U Atom Ratios
at $k_\infty \approx 1.00$. (Area Under Each Curve is Equal)

FIGURE 14
 Fraction of ^{239}Pu Absorptions in the Neutron Energy Range of 10^2 to $4(10)^4$ ev as a
 Function of the C/U Atom Ratio at $k_\infty \approx 1.0$

FIGURE 15
 Fraction of ^{239}Pu Fissions in the Neutron Energy Range of 10^2 to $4(10)^4$ eV as a
 Function of the C/U Atom Ratio at $k_\infty \approx 1.0$

than 50 milli-k, whereas the value of k_{∞} near unity can be measured to within about ± 5 milli-k.

Figure 12 shows the median fission energy of ^{239}Pu , defined as the energy at which half of the fissions are caused by neutrons of energies greater and half are caused by neutrons of energies less than this value. The variation of this median energy gives an indication of the shift in spectrum as the C/U is increased.

Figure 13 is a plot of the ten group absorption rate in ^{239}Pu for the four different C/U ratios. Indicated by these plots is the shift in the reaction rate energy absorption distribution as the value of the C/U atom ratio is changed. Figure 14 is a plot of the fraction of ^{239}Pu absorptions for neutrons (1) above $4(10)^4\text{ev}$, (2) within the range of 10^2 to $4(10)^4\text{ev}$, and (3) below 10^2ev . Figure 15 is the same type of plot for the fraction of fissions in ^{239}Pu . A maximum calculated value of greater than 60% of the ^{239}Pu absorptions occur within the $(10)^2$ to $4(10)^4\text{ev}$ range at a C/U atom ratio of about 4. Thus, a large fraction of the ^{239}Pu reactions fall in the range of interest.

Tables 1 and 2 give the relative contribution of various terms in the derivation of $\bar{\alpha}$. The uncertainties known from previous experience are included for the measurable quantities and the calculated quantities are assigned conservative uncertainties. These tables show the combined error analysis leading to the uncertainties quoted earlier.

CONCLUSIONS

Integral values of $\bar{\alpha}$ for ^{239}Pu can be measured in the PCTR with an uncertainty less than ± 0.06 . This uncertainty is about $\pm 10\%$ of $\bar{\alpha}$ at a C/U atom ratio of 6.

The present uncertainty in the calculated value of this $\bar{\alpha}$ from differential cross section data is about 40%. Thus, the PCTR experiments over a range of C/U ratio, that give large fractions of the ^{239}Pu reaction rates in the uncertain region of $\bar{\alpha}$, will allow selection of the proper value of the capture-to-fission ratio to be used in Fast Breeder Reactor (FBR) calculations.

Inexpensive (unsintered) homogeneous fuels can be used for these experiments, or, if available, the ZPR platelet fuel could be used.

FURTHER INVESTIGATIONS

If both the homogeneous fuel and the ZPR platelet fuel were used to make a measurement at the same C/U ratio, a valuable comparison could be made. Such an experiment would give data to compare with calculations of the heterogeneity correction for ZPR platelet fuel.

A simple doppler coefficient measurement could be made on the central fuel element in the FNC. This data would give the doppler coefficient in the low temperature region ($\sim <150^\circ\text{C}$) as a function of the carbon-to-uranium ratio for these plutonium fueled systems at k_∞ near unity.

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A P P E N D I X A

Neutron Flux Distribution - Spherical Model of the PCTR
for ²³⁹Pu Alpha Experiments in Fast Spectra

(S. R. Bierman)

APPENDIX A

Neutron Flux Distribution - Spherical Model of the PCTR
for ^{239}Pu Alpha Experiments in Fast Spectra

A series of calculations have been made by using the transport theory code DTF-IV, with one order of anisotropy and an 18-group energy structure, to define in both space and energy the shape of the neutron flux in the following model of the PCTR (referred to as DDLAN-1):

Region	Composition		Outside Radius cm
	Nuclide	at/b cm	
Core	C	0.03760	34.366
	O	0.01260	
	^{238}U	0.00565	
	^{235}U	0.000010	
	^{239}Pu	0.000588	
	^{240}Pu	0.000051	
	Fe	0.00450	
	Cr	0.00117	
	Ni	0.00046	
Void			43.307
Stainless Steel Interface	Fe	0.063542	44.888
	Cr	0.016576	
	Ni	0.006534	
Fast Neutron Cavity	^{234}U	1.9813 (10) ⁻⁶	61.945
	^{235}U	2.3411 (10) ⁻⁴	
	^{238}U	3.2233 (10) ⁻²	
	N	0.48145 (10) ⁻⁵	
	O	0.12887 (10) ⁻⁵	
	Al	4.6887 (10) ⁻²	
Air-Aluminum Interface	Al	5.4224 (10) ⁻²	62.825
	N	1.97 (10) ⁻⁶	
	O	0.52 (10) ⁻⁶	
Graphite Interface	C	0.1282	65.0
Homogenized Pb-UO ₂ and Graphite Driver Ring	^{238}U	1.3487 (10) ⁻⁵	75.0+
	^{235}U	18.6970 (10) ⁻⁵	
	O	39.6190 (10) ⁻⁵	
	Pb	10.3259 (10) ⁻⁴	
	Al	12.0879 (10) ⁻⁴	
	Ni	18.6910 (10) ⁻⁵	
	C	10.6672 (10) ⁻²	
Graphite	C	0.1282	137.845+

Cross sections for these calculations were obtained in the 18-group structure from the GAM-Tech computer code.⁽⁶⁾ Different sized models were considered beyond the scope of this study, since the calculations indicated that the above described model, which is physically acceptable, was adequate neutronically.

Since this spherical model is a simulation of a proposed PCTR loading, calculations were first made for an infinite media of the core material to establish a reference with which to compare the calculations for the model. Flux levels as a function of energy and position are shown in Figure 1A and flux and fission distributions with energy are shown in Table 1A for both the reference infinite model and the spherical model. The important reaction rate ratios (i.e., ^{238}U absorptions, ^{235}U absorption, ^{235}U fission, and ^{239}Pu fission) are also shown in Figure 2A.

To provide some assurance that the capacity of the fast neutron cavity would be sufficient for achieving a matched spectrum in an actual experiment, calculations were made with the cavity only partially loaded. The results of these calculations are referred to as DDLAN-2 in Table 1A. As can be seen, both models, DDLAN-1 and DDLAN-2, have similar flux distributions and both agree quite well with the calculated infinite spectrum over at least the center 20 cm of the core.

A PCTR loading equivalent to the DDLAN-2 model offers the advantage that a greater percentage of the fissions occur in the core region than in a DDLAN-1 type loading. Consequently, the reactor will be more sensitive to any changes made to a given sized test cell.

TABLE 1A

RELATIVE FLUX AND FISSION DISTRIBUTIONSDDLAN Spherical Model of PCTR With Fast Neutron Cavity

Lower E of Group	INFINITE MODEL			COMPUTER EXPERIMENTAL MODELS ⁽¹⁾	
	i	Fission Fraction	$\phi_i(r)$	$\phi_i(10.31 \text{ cm}) / \sum_{i=1}^{18} \phi_i(10.31 \text{ cm})$	
			$\sum_{i=1}^{18} \phi_i(r)$		
			i=1	DDLAN 1	DDLAN 2
7.79 MeV	1	0.0019	0.0005	0.0005	0.0005
6.06 MeV	2	0.0055	0.0017	0.0018	0.0018
4.72 MeV	3	0.0098	0.0046	0.0048	0.0048
3.68 MeV	4	0.0136	0.0070	0.0075	0.0075
2.86 MeV	5	0.0197	0.0103	0.0112	0.0113
1.74 MeV	6	0.0692	0.0386	0.0404	0.0408
1.35 MeV	7	0.0287	0.0237	0.0245	0.0249
183.00 KeV	8	0.1087	0.2400	0.2414	0.2473
24.80 KeV	9	0.1238	0.2560	0.2512	0.2555
3.36 KeV	10	0.1454	0.2226	0.2219	0.2178
454.00 eV	11	0.1930	0.1484	0.1482	0.1428
101.00 eV	12	0.2103	0.0418	0.0418	0.0402
37.30 eV	13	0.0657	0.0042	0.0042	0.0040
13.70 eV	14	0.0039	0.0005	0.0005	0.0005
5.04 eV	15	0.0008	0.0001	0.0001	0.0003
1.85 eV	16				
0.68 eV	17				
0.00 eV	18				
		1.0000	1.0000	1.0000	1.0000
k_{eff}		1.1644	---	1.0002	1.0001
Epi 1.35 MeV		---	0.0864	0.0907	0.0916
101 eV to 1.35 MeV		---	0.9088	0.9045	0.9036
0 eV to 101 eV		---	0.0048	0.0048	0.0048
FISSION DISTRIBUTION					
Core Material		1.000	---	0.0513	0.1103
FNC Fuel		---	---	0.2402	0.0997
PCTR Drivers		---	---	0.7085	0.7900

(1) Models DDLAN 1 and 2 are identical except the fast neutron cavity contains only 3.37 cm of K5N fuel in Model 2 and is fully loaded in 1.

Calculations were not made to assess the reactivity response of either PCTR model to center test cell. With 11.03% of the fissions occurring in the core region of the DDLAN-2 model, adequate sensitivity can probably be obtained with a reasonably sized test cell. Also, with some 70% of the fissions occurring in the driver region, control should not be a problem.

Please note the expanded scales in the reaction rate figures as compared to the scale used in Figure 1A for the flux distribution. In Figure 1A the percentage of the total neutron flux at any point in the given model is plotted for each of the indicated energy groups. This was felt to be a more useful method of presentation than a ratio showing the percent deviation from the corresponding flux in the infinite model.

FIGURE 1A
Neutron Flux Distribution DDLAN Spherical Model of PCTR with
FNC

FIGURE 2A

Cross Section Ratios -- DDLAN Spherical Model of
PCTR with FNC

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