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MEASUREMENT AND ANALYTICAL
CORRELATION OF NEUTRONIC
PARAMETERS FOR A WATER MODERATED
LATTICE CONTAINING UO_2 - PuO_2
PARTICULATE FUELS

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MEASUREMENT AND ANALYTICAL CORRELATION
OF NEUTRONIC PARAMETERS FOR A WATER MODERATED
LATTICE CONTAINING UO_2 - PuO_2 PARTICULATE FUELS

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ABSTRACT

The neutron multiplication factor, k_{∞} , and relative reaction rates have been measured at room temperature in the Physical Constants Testing Reactor for a water moderated lattice fueled with zircalloy clad UO_2 - PuO_2 fuel rods. The plutonium content in different sets of fuel rods ranged from 0.9 to 2.5 wt% PuO_2 in UO_2 . These experiments provide neutronics data for uniform lattices of mixed oxide fuels which contain plutonium oxide distributed throughout the uranium oxide as finite-sized particles. The PuO_2 particle sizes in different sets of fuel rods ranged between zero and 328 microns in diameter. Knowledge of the effect that plutonium particle size has on neutron multiplication is important for the economic utilization of plutonium. An experiment-theory correlation is presented which provides a basis for assessing the adequacy of calculational techniques.

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INTRODUCTION

Fuel element development for plutonium recycle has led to fuels composed of the mixed oxides of plutonium and uranium. These fuel elements are presently fabricated by distributing the plutonium throughout the uranium as particles. Due to self shielding properties of the particle, the reactivity of the fuel is a function of particle size. Since a change in particle size is probably either in the event of a reactor transient or during normal fuel irradiation and burn up, knowledge of the effect that plutonium particle size has on the reactivity of a reactor system is important for the economic.

As part of the Plutonium Utilization Program of the USAEC, conducted at Pacific Northwest Laboratories by Battelle Memorial Institute, a series of experiments has been performed in the Physical Constants Testing Reactor (PCTR) with a 1.0-inch square pitch light water moderated lattice containing UO_2 - PuO_2 fuel rods. The fuel rods were fabricated by distributing plutonium oxide, in the form of spherical particles, throughout the uranium oxide diluent. These experiments provide neutron multiplication and relative reaction rate data as a function of PuO_2 particle size in a water-moderated lattice. The results provide a base for assessing the adequacy of calculational techniques used in reactor design to predict neutronic characteristics of light water moderated UO_2 - PuO_2 lattices.

DESCRIPTION OF EXPERIMENT

Since the Physical Constants Testing Reactor (PCTR), shown in Figure 1, first went critical in October of 1955, a large number of neutronics experiments using many different types of reactor lattices have been performed. A light water lattice tank was installed in the PCTR central test cavity, shown in Figure 2, during 1967. Several lattice neutronics experiments using light water moderated enriched uranium⁽¹⁾ and UO_2 - PuO_2 fuels⁽²⁾ have been conducted in the PCTR water tank. Most of the water moderated UO_2 - PuO_2 fueled experiments used 0.565-inch outer diameter zircalloy clad rods in a 1.0-inch square lattice pitch. Results presented in this report are from those mixed oxide experiments.

Test Lattice in PCTR Water Tank

The PCTR Water Tank is a cylindrical vessel, 17.5-inches inside diameter and 37-inches long. An aluminum drip pan beneath the tank centers it in the PCTR test cavity which is 41.25 x 41.25 x 37.5 inches. The inner region of the tank, 11.5-inches inside diameter, contained 108 fuel rods of either UO_2 - 0.9 wt% PuO_2 or UO_2 - 2.0 wt% PuO_2 (8% ^{240}Pu). The outer tank annulus was empty during all UO_2 - PuO_2 fueled experiments. The central lattice assembly, shown in Figure 3, was inserted in the inner most region of the water tank. A 20-inch long center cell, consisting of a set of 4 rods, could be removed from the central lattice assembly, as shown in Figures 3 and 4, and replaced with an aluminum void can. The central region end buffers were shortened duplicates of the center cell fuel such that the total length of the assembly was less than the 37.5-inch length of the test cavity. Fuel specifications for the UO_2 - 2 wt% PuO_2 and the UO_2 - 0.9 wt% PuO_2 are given in Tables I and II, respectively.

FIGURE 1: THE PHYSICAL CONSTANTS TESTING REACTOR (PCTR)

FIGURE 2. WATER MODERATED UO_2 - PuO_2 LATTICE INSTALLED
IN THE PCTR

FIGURE 3: CENTRAL LATTICE ASSEMBLY WITH REMOVABLE CENTER CELL AND ALUMINUM VOID CAN

FIGURE 4: REMOVAL AND INSERTION OF CENTRAL LATTICE ASSEMBLY

TABLE I

FUEL SPECIFICATIONS FOR UO_2 -2 wt% PuO_2 FUEL RODS

Rod Dimensions:

Cladding is Zircalloy-2 tubing with plugs seal welded at both ends.

Total weight of loaded fuel rod is 1340 gm (average).

FUEL LOADINGS

The average fuel composition consists of 22.56 gm of PuO_2 mixed with 1105.44 gm of natural UO_2 , vibrationally compacted to a density of 9.54 gm/cm^3 . The PuO_2 particle size is nominally 25 microns diameter. The isotopic composition on January 1965 was:

<u>Plutonium - Atom %</u>	<u>Uranium - Atom %</u>
91.615 ^{239}Pu	0.006 ^{234}U
7.654 ^{240}Pu	0.72 ^{235}U
0.701 ^{241}Pu	99.27 ^{238}U
0.031 ^{242}Pu	

TABLE II

FUEL SPECIFICATIONS FOR UO_2 -0.9 wt% PuO_2

FUEL RODS

Rod Dimensions:

Cladding is Zircalloy-2 tubing with plugs seal welded at both ends.

Total weight of loaded fuel rods is 1316 gms (average).

FUEL LOADINGS

The average fuel composition consists of 9.61 gm of PuO_2 mixed with 1097.39 gm of depleted UO_2 , vibrationally compacted to a density of 9.38 gm/cm^3 . The PuO_2 particle size is nominally 25 microns diameter. The isotopic composition on November 24, 1964 was:

<u>Plutonium - Atom %</u>	<u>Uranium - Atom %</u>
92.139 ± 0.050 ^{239}Pu	0.005 ± 0.001 ^{234}U
7.241 ± 0.055 ^{240}Pu	0.234 ± 0.002 ^{235}U
0.595 ± 0.005 ^{241}Pu	0.006 ± 0.0005 ^{236}U
0.025 ± 0.001 ^{242}Pu	99.755 ± 0.002 ^{238}U

UO₂ - PuO₂ Particulate Fuel

Eleven different sets of four center cell fuel rods with known PuO₂ particle diameters were used in the PCTR water tank experiment. These fuel rod sets can be categorized into five different PuO₂ particle size diameters: 0, 52, 107, 195, and 328 microns. The zero micron PuO₂ was made by crushing and ball milling the plutonium oxide to an extremely fine powder, cold pressed into pellets, and sintered to a solid solution. The uniformity of the finished pellets is illustrated by the α - radiograph of a UO₂ - 2 wt% PuO₂ pellet ($\sim$ zero micron PuO₂ particle diameter) shown in Figure 5. Finite sized PuO₂ particles were prepared by two different methods. The first method involved crushing PuO₂ pellets to produce granules, sieving to separate desired size fractions, jet milling to round the pellets, sintering to desired density, and double screening for final particle size separation. Roundness of particles was assured by rolling tests and microscopic examination. The second method involved the same steps for crushing PuO₂ pellets to produce granules and sieving to separate desired size fractions, but the particles were spheroidized and densified by fusing in a plasma torch. Spheroidicity of fused PuO₂ particles (nominally 195 microns in diameter) produced in a plasma torch is illustrated by the photomacrograph in Figure 6. Finite PuO₂ particles with diameters of 52, 107, and 195 microns were produced in the plasma torch. The time required to fuse a granule, falling through the plasma torch, limited the maximum size of finished PuO₂ particles to about 200 microns diameter. The jet milling method produced PuO₂ particles with diameters: 107, 195, and 328 microns. The spheroidicity of these particles depends largely upon the amount of jet milling received by the individual particle. Fabrication of UO₂ - PuO₂

FIGURE 5: ALPHA PARTICLE AUTORADIOGRAPH OF UO₂ - 2 WT% PuO₂
PELLET ($\sim$ 0 MICRON PuO₂ PARTICLE DIAMETER)

FIGURE 6: PHOTOMACROGRAPH OF 195 MICRON DIAMETER PuO_2 PARTICLES

fuel of a given enrichment would require 250 times as many 52 micron PuO_2 particles than would be required if 328 microns PuO_2 particles were used. Consequently, no 52 micron PuO_2 particles were jet milled because the number of particles required was prohibitive.

Procedures for fabrication of UO_2 - PuO_2 pellets using finite sized PuO_2 particles were the same as that using zero micron PuO_2 , except that the cold pressed pellets were not sintered. The glycerine and water binder used in the pellet pressing operation was removed by firing at 200°C in a vacuum. The distribution of the PuO_2 particles in the finished pellets is illustrated by the α -radiographs of fuel containing 52, 107, and 195 micron diameter PuO_2 particles in Figures 7, 8, and 9, respectively. The diffused white spots on the α -radiographs are due to film overexposure caused by plutonium contacting the film. If the film does not contact the pellet, the PuO_2 particle image resolution is poor, as shown by the α -radiograph of the axial surface of a fuel pellet containing 52 micron diameter PuO_2 particles in Figure 10.

The binder free pellets were loaded into zircalloy tubes and zircalloy plugs were seal welded at both ends. The loaded fuel was densified by swaging in the cladding tube to approximately 80% of the theoretical density. The final outer diameters of the fuel and cladding after swaging were 0.505-inches and 0.565-inches, respectively. Compositions of the UO_2 - PuO_2 particle size fuel sets are listed in Table III.

The relative plutonium enrichment in each type fuel was determined from extensive gamma scan measurements with a high resolution Ge-Li Detector.⁽³⁾ Corrections were made for the difference in gamma ray attenuation due to lumping the plutonium in different diameter spherical particles.⁽⁴⁾ Pellets from one rod containing zero micron PuO_2 particles and one rod containing 328 micron PuO_2 particles were dissolved and chemically analyzed.⁽⁵⁾

FIGURE 7: ALPHA PARTICLE AUTORADIOGRAPH OF RADIAL SURFACE OF
UO₂ - PuO₂ PELLET CONTAINING 52 MICRON DIAMETER PuO₂
PARTICLES

FIGURE 8: ALPHA PARTICLE AUTORADIOGRAPH OF RADIAL SURFACE OF
 $\text{UO}_2 - \text{PuO}_2$ PELLET CONTAINING 107 MICRON DIAMETER PuO_2
PARTICLES

FIGURE 9: ALPHA PARTICLE AUTORADIOGRAPH OF RADIAL SURFACE OF
 $\text{UO}_2 - \text{PuO}_2$ PELLET CONTAINING 195 MICRON DIAMETER PuO_2
PARTICLES

FIGURE 10: ALPHA PARTICLE AUTORADIOGRAPH OF AXIAL SURFACE OF UO_2 - PuO_2
PELLET CONTAINING 52 MICRON DIAMETER PuO_2 PARTICLES

The relative enrichment results were normalized to the chemical analysis data to obtain the percent by weight of PuO_2 in $\text{UO}_2 + \text{PuO}_2$ for each set of rods listed in Table III.

TABLE III
COMPOSITION OF UO_2 - PuO_2 PARTICULATE FUELS

Fuel Set Identification	PuO_2 Particle Diameter (microns)	Fuel Density (gm/cm^3)	wt% PuO_2 in UO_2 - PuO_2 Mixture	PuO_2 Particle Fabrication Method
8BMB	0	8.571	1.98	Ball Mill
8SC	0	8.704	2.08	Ball Mill
8BME	0	8.474	2.10	Ball Mill
8P50	52	8.601	2.29	Plasma Torch
8B100	107	8.739	2.48	Jet Mill
8P100	107	8.638	2.47	Plasma Torch
8LD100	107	7.630	2.45	Jet Mill
8B200	195	8.718	2.35	Jet Mill
8P200	195	8.657	2.41	Plasma Torch
8B350	328	8.691	2.42	Jet Mill
8RA350	328	8.798	2.47	Jet Mill

Isotopic Composition on March 25, 1970:

<u>Plutonium Atom %</u>		<u>Uranium Atom %</u>	
90.88	^{239}Pu	0.006	^{234}U
7.99	^{240}Pu	0.72	^{235}U
1.07	^{241}Pu	99.27	^{238}U
0.06	^{242}Pu		

Rod Dimensions:

Fuel: 1.283 cm diameter x 50.80 cm length.

Clad: 1.435 cm outer diameter x 0.076 cm wall Zircalloy - 2.

Overall Clad Length: 51.92 cm

Total Weight of Loaded Fuel Rods: 678 grams (average)

DATA COLLECTION

Null Reactivity Measurements

The measurement in the PCTR consists of simultaneously determining the matched spectrum incident on the central cell and the proper central cell constituents such that the change in the PCTR reactivity is zero when the central cell is removed leaving a void. The k_{∞} of the void as seen by the PCTR is unity. With no change in neutron distribution or reactivity upon reinsertion of the central cell, the k_{∞} of the central cell material is thus implied to be unity.⁽⁶⁾

The change in reactivity of the PCTR due to the substitution of a void for each of eleven sets of $UO_2 - PuO_2$ particle size fueled center cells was measured with natural boron dissolved in the water moderator at concentrations of 1200, 1300, and 1400 weight - parts - per - million (wppm). Similar measurements were made with the same concentrations of boron in water using $UO_2 - 2.0 \text{ wt\% } PuO_2$ (nominally 25 micron diameter PuO_2) vibrationally compacted center cell rods. Spectral buffering regions surrounding the center cell, for all these measurements, contained $UO_2 - 2.0 \text{ wt\% } PuO_2$ vibrationally compacted fuel (listed in Table I). The fundamental mode neutron spectrum of the lattice with this fuel very nearly matched that of the particle size fueled lattice.

Additional measurements were made in the lattice completely fueled with $UO_2 - 0.9 \text{ wt\% } PuO_2$ rods (listed in Table II). The change in reactivity of the PCTR, due to substitution of a void for a $UO_2 - 0.9 \text{ wt\% } PuO_2$ fueled center cell, was measured with natural boron in the water at concentrations of 65, 250, and 280 wppm. Experimentally determined corrections to the measured reactivity worths, for the presence of aluminum walls and nitrogen in the void, were very small in all cases. The reactivity worth of additional

layers of aluminum wall added to the void was measured at each concentration of boron in water. The reactivity worth of the 0.125-inch thick aluminum wall of the void can was determined from a regression analysis of the resulting data, at each boron concentration. Determination of the reactivity worth of nitrogen (in air), which fills the void formed by the removal of the center cell, was inferred from measurement of the reactivity worth of a thin strip of copper inserted in the void can.⁽⁷⁾ The boron concentration corresponding to the null reactivity condition, for each center cell fuel composition, was obtained by interpolation of the center cell reactivity worth versus boron concentration data, as shown in Figures 11 and 12 for the UO_2 - 2.0 wt% PuO_2 fueled lattice and the UO_2 - 0.9 wt% PuO_2 fueled lattice, respectively. For the particle size fuel, a two-dimensional (R-Z) calculational model of the PCTR with the water moderated lattice installed in the test cavity, was used to evaluate small corrections to the null reactivity boron concentrations due to neutron spectrum mismatch in the center cell.⁽⁸⁾ These refinements, calculated using the 2DB⁽⁹⁾ code, have resulted in small changes in the measured boron concentrations at null reactivity. The corrected values of the concentration of natural boron dissolved in the water corresponding to $k_\infty \equiv 1.000$ in the matched spectrum of the center cell are listed in Table IV for each center cell fuel composition.

Neutron Spectrum Matching

The measurement of k_∞ requires that the neutron energy distribution in the center test cell duplicates the neutron spectrum found in the center of a bare, just critical, array of identical cells. This is accomplished in the PCTR by surrounding the test cell with a layer of medium, the buffer, which is identical to the test sample and which brings the spectrum into equilibrium, i.e., alters the flux incident upon the test cell such that

FIGURE 11: CONCENTRATION OF BORON IN WATER TO ACHIEVE $K_{\infty} = 1.0$ FOR THE UO_2 - 2.0 WT% PuO_2 FUELED LATTICE

FIGURE 12: CONCENTRATION OF BORON IN WATER TO ACHIEVE $K_{\infty} = 1.0$ FOR THE $\text{UO}_2 - 0.9 \text{ WT\% PuO}_2$ FUELED LATTICE

TABLE IV

ANALYSIS OF PuO₂ PARTICULATE EXPERIMENTS WITH THE LATTICE POISONED TO K_∞ OF UNITY

Particulate Fuel Type	PuO ₂ Particle Diameter (microns)	Fuel Density (gm/cm ³)	wt% PuO ₂ in UO ₂ + PuO ₂	Experimental Boron Concentration in H ₂ O for k _∞ = 1.000 (wppm)	Calculated k _∞ at k _∞ = 1.000 Boron Concentration from Experiment
8BMB	0	8.571	1.98	1213 ± 4	0.990
8SC	0	8.704	2.08	1286 ± 4	0.991
8BME	0	8.474	2.10	1274 ± 4	0.992
8P50	52	8.601	2.29	1350 ± 4	0.991
8B100	107	8.739	2.48	1378 ± 4	0.997
8P100	107	8.638	2.47	1332 ± 4	0.999
8LD100	107	7.630	2.45	1227 ± 4	0.995
8B200	195	8.718	2.35	1275 ± 4	0.985
8P200	195	8.657	2.41	1261 ± 4	0.992
8B350	328	8.691	2.42	1163 ± 4	0.986
8RA350	328	8.798	2.47	1229 ± 4	0.984

Vibrationally
Compacted
Fuels

Nominal PuO₂
Particle Diameter
(microns)

Average Value = 0.991

Root - Mean - Square Deviation = ± 0.004

UO ₂ -2 wt% PuO ₂	25	9.54	2.00	1353 ± 3	0.995
UO ₂ -0.9 wt% PuO ₂	25	9.38	0.87	276 ± 3	0.989

the epithermal-to-thermal flux ratio remains constant across the center test cell. The true value of $k_{\infty} - 1$ for the test lattice is measured only if the neutron spectrum incident on the center cell is identical to the fundamental mode spectrum of the test lattice.^(6, 10) Determination of the condition of the spectral equilibrium was made by measuring spatial variations in a suitable integral spectral index. Measurements of the cadmium ratio of 0.05-cm diameter gold pins, placed on the fuel cladding surface throughout the test cell and buffer region were sufficiently sensitive for this purpose. Results of the radial traverse data, shown in Figure 13, indicate that the test lattice volume was more than adequate to achieve an equilibrium neutron spectrum in the vicinity of the center cell for all lattice compositions.

Relative Reaction Rates

Foil activation techniques were utilized in the determination of the relative reaction rates in all central cell constituents and the standard absorber (either copper or boron) inserted in the center of the PCTR. Both bare and cadmium covered foils were used to measure epithermal - to - thermal reaction rates in the fuel, moderator, and cladding for all central cell constituents. The PCTR graphite thermal column was used to normalize activities induced by thermal neutrons in different foil materials. Activation rates from foils irradiated in the center cell were measured relative to activation rates from the same type foils irradiated simultaneously in the thermal column. Foils inserted into a separable center cell fuel rod, shown in Figure 3, were 1.27 cm in diameter. The foil thicknesses and compositions were: 0.0127 cm copper; 0.0254 cm Al - 1.8 wt% ^{239}Pu ; and 0.0102 cm Al - 7 wt% ^{235}U . Another separable center cell fuel rod contained foils of 20-fold depleted uranium (0.036% ^{235}U) and natural uranium metal which were 1.27 cm in diameter and 0.025 cm thick. Separable center

FIGURE 13: NEUTRON SPECTRUM EQUILIBRATION IN THE WATER MODERATED UO_2 - PuO_2 LATTICE

cell fuel rods containing zero micron PuO_2 particles in UO_2 - 2 wt% PuO_2 (Type 8SC) and 328 micron diameter PuO_2 particles (Type 8RA350) were used to measure relative reaction rates in their respective particulate fueled center cells. A special type foil, 1.27 cm in diameter and 0.050 cm thick, containing 48 PuO_2 particles 328 microns in diameter dispersed in an aluminum matrix, was used to measure the relative ^{239}Pu fission rate in the 328 micron PuO_2 particle sized separable center cell fuel rod. A similar set of separable center cell rods for the vibrationally compacted UO_2 - 0.9 wt% PuO_2 fueled lattice were used to measure relative reaction rates in the fuel for that lattice. Copper sector foils, 0.0127 cm thick, were inserted in the water moderator. Bare foils and foils covered by 0.1 cm thick cadmium were irradiated separately in the center cell. The relative reaction rates in all center cell constituents were measured in both the lattice poisoned to unit multiplication and the unpoisoned lattice, for each fuel composition. The resulting relative reaction rates are listed in Table V for each lattice condition.

Relative activation rates of irradiated foils were measured using a NaI(Tl) gamma ray scintillation counting system.⁽¹¹⁾ The counting data was analyzed using computer code APDAC⁽¹²⁾ which corrects the counting data for activity decay, room background, foil mass variations, residual activity, counter drift, and dead time. The excess neutron production in the unpoisoned lattice can be determined from a neutron balance in the center cell, using the measured relative reaction rates after corrections due to foil perturbation effects⁽¹³⁾ have been applied to the data.

TABLE V
RELATIVE REACTION RATES MEASURED IN AN H₂O
MODERATED UO₂ - PuO₂ FUELED LATTICE

UO ₂ - PuO ₂ Lattice (Zero Micron PuO ₂ Particulate Fuel) Type 8SC				
Relative Reaction Rate	Unpoisoned Lattice		Poisoned Lattice	
	Bare	Cd Covered	Bare	Cd Covered
$\frac{(\sigma_a \phi)_{H_2O}^{Cu}}{(\sigma_a \phi)_{Thermal Column}^{Cu}}$	12.75(± 0.63%)	1.136(± 0.42%)	3.428(± 0.42%)	0.412(± 0.46%)
$\frac{(\sigma_a \phi)_{Fuel}^{Cu}}{(\sigma_a \phi)_{Thermal Column}^{Cu}}$	10.15(± 0.47%)	1.105(± 0.43%)	2.735(± 0.31%)	0.401(± 0.46%)
$\frac{(\sigma_f \phi)_{Fuel}^{235}}{(\sigma_f \phi)_{Thermal Column}^{235}}$	9.43(± 0.71%)	0.615(± 0.97%)	2.484(± 0.86%)	0.224(± 0.90%)
$\frac{(\sigma_f \phi)_{Fuel}^{239}}{(\sigma_f \phi)_{Thermal Column}^{239}}$	11.36(± 0.71%)	0.623(± 1.14%)	3.056(± 0.94%)	0.225(± 1.00%)
$\frac{(\sigma_c \phi)_{Fuel}^{238}}{(\sigma_c \phi)_{Thermal Column}^{238}}$	17.58(± 1.70%)		5.762(± 1.20%)	
UO ₂ - PuO ₂ Lattice (328 Micron Diameter PuO ₂ Particulate Fuel) Type 8RA350				
Relative Reaction Rate	Unpoisoned Lattice		Poisoned Lattice	
	Bare	Cd Covered	Bare	Cd Covered
$\frac{(\sigma_a \phi)_{H_2O}^{Cu}}{(\sigma_a \phi)_{Thermal Column}^{Cu}}$	13.06(± 0.29%)	1.142(± 0.50%)	3.631(± 0.28%)	0.421(± 0.46%)
$\frac{(\sigma_a \phi)_{Fuel}^{Cu}}{(\sigma_a \phi)_{Thermal Column}^{Cu}}$	10.34(± 0.28%)	1.101(± 0.50%)	2.901(± 0.27%)	0.410(± 0.46%)
$\frac{(\sigma_f \phi)_{Fuel}^{235}}{(\sigma_f \phi)_{Thermal Column}^{235}}$	9.70(± 0.73%)	0.612(± 1.17%)	2.653(± 0.82%)	0.231(± 0.63%)
$\frac{(\sigma_f \phi)_{Fuel}^{239}}{(\sigma_f \phi)_{Thermal Column}^{239}}$	8.10(± 0.37%)	0.459(± 0.77%)	2.342(± 0.84%)	0.163(± 0.75%)
UO ₂ - 0.9 wt% PuO ₂ Vibrationally Compacted Fuel (See Table II)				
Relative Reaction Rate	Unpoisoned Lattice		Poisoned Lattice	
	Bare	Cd Covered	Bare	Cd Covered
$\frac{(\sigma_a \phi)_{H_2O}^{Cu}}{(\sigma_a \phi)_{Thermal Column}^{Cu}}$	9.053(± 0.60%)	0.491(± 0.55%)	5.501(± 0.65%)	0.341(± 0.59%)
$\frac{(\sigma_a \phi)_{Fuel}^{Cu}}{(\sigma_a \phi)_{Thermal Column}^{Cu}}$	7.630(± 0.41%)	0.481(± 0.56%)	4.710(± 0.43%)	0.337(± 0.59%)
$\frac{(\sigma_f \phi)_{Fuel}^{235}}{(\sigma_f \phi)_{Thermal Column}^{235}}$	7.339(± 0.72%)	0.283(± 1.06%)	4.465(± 0.76%)	0.198(± 1.35%)
$\frac{(\sigma_f \phi)_{Fuel}^{239}}{(\sigma_f \phi)_{Thermal Column}^{239}}$	8.406(± 1.15%)	0.254(± 1.42%)	5.318(± 1.11%)	0.170(± 1.99%)
$\frac{(\sigma_c \phi)_{Fuel}^{238}}{(\sigma_c \phi)_{Thermal Column}^{238}}$	12.204(± 2.16%)	4.791(± 3.67%)		

THEORY OF EXPERIMENT INTERPRETATION

PCTR Experiments

PCTR experiments utilize a small central test lattice region in which the neutron spectrum can be adjusted to duplicate the equilibrium spectrum in the fundamental mode (a just critical bare sphere of the test material). These experiments permit neutronic parameters of various reactor lattices to be measured using relatively small volumes of those lattices.⁽¹⁰⁾ A removable central cell is surrounded both radially and axially by test lattice medium which is identical to the central cell and which brings the neutron spectrum into equilibrium. When the test lattice region is of sufficient volume, the neutron spectrum in the vicinity of the central cell is identical to the fundamental mode spectrum of the test lattice.⁽⁶⁾

The PCTR technique for measuring the infinite neutron multiplication factor, k_{∞} , of a multiplying medium is based upon the determination of the excess neutron production of a test sample.⁽¹⁴⁾ This excess neutron production is deduced from the measured concentration of neutron absorber that must be added to the central test lattice region such that the change in the PCTR reactivity is zero when the central cell is removed leaving a void.⁽¹⁵⁾ The k_{∞} of the void as seen by the PCTR is unity. Since there would be no change in reactivity, neutron density, or neutron energy distribution upon reinsertion of the central cell, the k_{∞} of the central cell material is implied to be unity.⁽⁶⁾

Theoretical Model

The k_{∞}^* measured in the PCTR poisoned technique is consistent with the usual two group critical equation⁽¹⁶⁾

$$k_{\infty}^* = (1 + B^2 \tau)(1 + B^2 L^2), \quad (1)$$

where τ and L^2 are defined as

$$\tau = \frac{D_1}{\Sigma_1^a + \Sigma_{1 \rightarrow 2}} \quad \text{and} \quad L^2 = \frac{D_2}{\Sigma_2^a},$$

where Σ_1^a and Σ_2^a are the absorption cross sections for epithermal and thermal neutrons, and $\Sigma_{1 \rightarrow 2}$ is the cross section for removal of neutrons from the epithermal group by scattering to the thermal group. Equation (1) is obtained from the usual condition that the two group flux equations,

$$-D_1 B^2 \phi_1 - \Sigma_1^a \phi_1 - \Sigma_{1 \rightarrow 2} \phi_1 + \frac{k_\infty^*}{p} \Sigma_2^a \phi_2 = 0 \quad (2)$$

$$-D_2 B^2 \phi_2 - \Sigma_2^a \phi_2 + \Sigma_{1 \rightarrow 2} \phi_1 = 0 \quad (3)$$

have a nontrivial solution, i.e., the determinant of the coefficients shall vanish. One particular point needing emphasis is that ϕ_1 and ϕ_2 are the equilibrium fluxes in the test region in the fundamental mode (just critical) environment.

For Equation (2) to explicitly account for all neutron production, epithermal as well as thermal fissions, k_∞^* must be defined as

$$k_\infty^* = \frac{\nu_1 \Sigma_1^f \phi_1 + \nu_2 \Sigma_2^f \phi_2}{\Sigma_2^a \phi_2} \quad (p) \quad (4)$$

where

$$p = \frac{\Sigma_{1 \rightarrow 2}}{\Sigma_{1 \rightarrow 2} + \Sigma_1^a} = \frac{(\Sigma_2^a + D_2 B^2) \phi_2}{(\Sigma_2^a + D_2 B^2) \phi_2 + \Sigma_1^a \phi_1} \quad (5)$$

The notation in Equation (4) can be simplified using the symbols A_i to refer to absorption of neutrons in energy group i , and P_i to refer to production of neutrons into energy group 1 due to fission caused by neutrons in group i .

$$k_{\infty}^* = \left[\frac{P_1 + P_2}{A_2} \right] p = \left(\frac{TP}{A_2} \right) p, \quad (6)$$

where $TP = P_1 + P_2$.

A similar expression can be written for the infinite neutron multiplication factor of the lattice poisoned to null reactivity.

$$k_{\infty}' = 1 = \left[\frac{P_1' + P_2'}{A_2' + A_{2a}'} \right] p' = \left[\frac{TP'}{A_2' + A_{2a}'} \right] p' \quad (7)$$

where the primed symbols refer to quantities in the poisoned lattice, the symbols, A_{2a}' and A_{1a}' refer to the subcadmium and epicadmium absorption in the poison, and the expression for p' is given by

$$p' = \frac{A_2' + A_{2a}'}{A_2' + A_{2a}' + A_1' + A_{1a}'} \quad (8)$$

The unpoisoned lattice k_{∞}^* can be written as Equation (6) divided by Equation (7)

$$k_{\infty}^* = \left(\frac{TP}{A_2} \right) \left[\frac{A_2' + A_{2a}'}{TP'} \right] \left(\frac{p}{p'} \right). \quad (9)$$

The infinite neutron multiplication factor calculated by various theoretical models is defined as

$$k_{\infty} = \frac{\text{Total neutron production}}{\text{Total neutron absorption}} = \frac{\nu_1 \Sigma_1^f \phi_1 + \nu_2 \Sigma_2^f \phi_2}{\Sigma_1^a \phi_1 + \Sigma_2^a \phi_2}$$

where ϕ_1 and ϕ_2 are equilibrium fluxes in the fundamental mode problem (a just critical sphere of the test material). This definition leads to a different expression for Equation (2), i.e.,

$$-D_1 B^2 \phi_1 - \Sigma_1^a \phi_1 - \Sigma_{1 \rightarrow 2} \phi_1 + k_\infty (\Sigma_1^a \phi_1 + \Sigma_2^a \phi_2) = 0 \quad (10)$$

The critical equation obtained from the coefficients of Equation (10) and Equation (3) is

$$k_\infty = (1 + B^2 L^2)(1 + B^2 \tau) / [1 + (1 - p)B^2 L^2] \quad (11)$$

Thus, the relationships between the infinite neutron multiplication factor measured in the PCTR (k_∞^*) and that usually quoted from theory (k_∞) is

$$k_\infty^* = k_\infty [1 + (1 - p)B^2 L^2] = \frac{k_\infty}{p} \left[1 + \frac{\Sigma_1^a \phi_1}{\Sigma_2^a \phi_2} \right]^{-1} \quad (12)$$

It is also readily apparent that if the thermal leakage is negligible ($D_2 B^2 \phi_2 \rightarrow 0$), then the escape - to - thermal probability becomes

$$p = \left[1 + \frac{\Sigma_1^a \phi_1}{\Sigma_2^a \phi_2} \right]^{-1} \quad (13)$$

and k_∞^* becomes identical with k_∞ . The advantage of the PCTR technique is that it measures $k_\infty^* - 1$.⁽¹⁰⁾ Therefore, it is appropriate to look at the difference between k_∞ and k_∞^* and compare this difference with $k_\infty - 1$. Rewriting Equation (11),

$$k_\infty = (1 + L^2 B^2)(1 + B^2 \tau) - \frac{(1 + \tau B^2)(1 + L^2 B^2)(1 - p)L^2 B^2}{1 + (1 - p)L^2 B^2} \quad (14)$$

and thus

$$\begin{aligned} k_\infty &= k_\infty^* - \frac{(1 + \tau B^2)(1 + L^2 B^2)(1 - p)L^2 B^2}{1 + (1 - p)L^2 B^2} \\ &\approx k_\infty^* - (1 - p)L^2 B^2. \end{aligned} \quad (15)$$

It may be seen that in cases with thermal leakage and p significantly less than 1, this difference can be a significant correction to $k_{\infty} - 1$.

Physically, the difference between k_{∞} and k_{∞}^* arises because k_{∞}^* does not take into account the fact that neutrons absorbed before being thermalized do not undergo thermal leakage.⁽¹⁴⁾ The value of $k_{\infty} - 1$ represents the true number of neutrons that leak out of a bare critical reactor for each neutron (of any energy) absorbed in the reactor.

The values for k_{∞} measured in the PCTR were interpreted in a manner consistent with the theory used for fundamental mode calculations. The expression used to evaluate the experimental k_{∞} values for the unpoisoned lattice was obtained by substitution of Equation (9) into Equation (15).

The result is:

$$k_{\infty} = \left(\frac{TP}{A_2}\right) \left(\frac{A_2' + A_{2a}'}{TP'}\right) \left(\frac{p}{p'}\right) - \frac{(1 + \tau B^2)(1 + L^2 B^2)(1 - p)L^2 B^2}{1 + (1 - p)L^2 B^2} . \quad (16)$$

EXPERIMENT ANALYSIS

The theory of PCTR measurements of k_{∞} has been described in detail in the preceding section. The presence of plutonium, which has a thermal neutron cross section with a significantly non- $1/v$ energy dependence, required a two group expression for analysis of experimental data. The original expression derived for determination of k_{∞} was based on the assumption that all materials in the center cell had a $1/v$ cross section energy dependence in the thermal range.⁽¹⁰⁾

The first term in Equation (16) is the dominant term. Relative reaction rates measured in the unpoisoned lattice were used to evaluate the quantity (TP/A_2) . Similarly, the measured concentration of boron in the water corresponding to k_{∞} of unity and relative reaction rates measured in the poisoned lattice were used to evaluate the quantity $(\frac{A_2' + A_{2a}'}{TP'})$. The addition of boron absorber in the water moderator did not change the slowing properties of the lattice significantly. The ratio of escape - to - thermal probabilities (p/p') of 1.0125 for the UO_2 - 2.0 wt% PuO_2 lattice and 1.0022 for the UO_2 - 0.9 wt% PuO_2 lattice were calculated using the HRG⁽¹⁷⁾ code. The second term in Equation (16) is a small correction term to account for the fact that neutrons absorbed before being thermalized do not undergo thermal leakage. This term was evaluated as 0.0054 for the UO_2 - 2.0 wt% PuO_2 lattice and 0.0016 for the UO_2 - 0.9 wt% PuO_2 lattice, using thermal group parameters calculated using Battelle-Revised-THERMOS⁽¹⁸⁾ and epithermal group parameters calculated using HRG. The fundamental mode buckling, B^2 , was evaluated using the calculated two group parameters for the unpoisoned lattice

$$B^2 = \frac{\eta_1 f_1 (1 - p) + \eta_2 f_2 p - 1}{\tau + L^2 [1 - \eta_1 f_1 (1 - p)]} \quad (17)$$

where

$$\eta_1 f_1 = \frac{v_1 \Sigma f_1}{\Sigma a_1}, \quad \eta_2 f_2 = \frac{v_2 \Sigma f_2}{\Sigma a_2},$$

$$\tau = \frac{D_1}{\Sigma a_1 + \Sigma_{1 \rightarrow 2}}, \quad L^2 = \frac{D_2}{\Sigma a_2}, \text{ and}$$

$$p = \frac{\Sigma_{1 \rightarrow 2}}{\Sigma a_1 + \Sigma_{1 \rightarrow 2}}.$$

Similar quantities were calculated for the lattice with each of the UO_2 - PuO_2 particulate fuels. Thermal group parameters were calculated using GRANIT⁽¹⁹⁾ and epithermal group parameters were calculated using EGGNIT.⁽²⁰⁾ These computer codes permit calculation of doubly heterogeneous systems, such as the UO_2 - PuO_2 particulate fueled lattices.

Thermal spectrum-averaged cross sections for the i_{th} reaction in the PCTR graphite thermal column, calculated using THERMOS, were used to normalize measured activity ratios,⁽²¹⁾

$$\frac{(\overline{\sigma\phi})_{\text{center cell}}^i}{\phi_{\text{Thermal Column}}} = \left[\frac{(\overline{\sigma\phi})_{\text{center cell}}^i}{(\overline{\sigma\phi})_{\text{Thermal Column}}^i} \right]_{\text{Exp.}} \left(\bar{\sigma}_{\text{THERMOS}} \right)_i^{\text{Thermal Column.}} \quad (18)$$

Total production, TP, was evaluated from relative fission rates measured for ^{239}Pu and ^{235}U

$$TP = \sum_i TP_i = \sum_i N_i V_i \left[\bar{v}_2^i \frac{(\overline{\sigma_f\phi})_2^i}{\phi_{\text{Thermal Column}}} + \bar{v}_1^i \frac{(\overline{\sigma_f\phi})_1^i}{\phi_{\text{Thermal Column}}} \right] \quad (19)$$

using calculated values for $\bar{v}$. Total production by ^{241}Pu was inferred from measured two group ^{239}Pu fission rates using calculated microscopic fission cross section ratios

$$TP^{241} = (NV)^{241} \left[\frac{(\overline{\sigma_f\phi})_2^{239}}{\phi_{\text{Thermal Column}}} \left(\frac{\sigma_f^{241}}{\sigma_f^{239}} \right)_2 \bar{v}_2^{241} + \frac{(\overline{\sigma_f\phi})_1^{239}}{\phi_{\text{Thermal Column}}} \left(\frac{\sigma_f^{241}}{\sigma_f^{239}} \right)_1 \bar{v}_1^{241} \right]. \quad (20)$$

Total production by ^{240}Pu and ^{242}Pu were inferred from the epithermal ^{239}Pu fission rate and calculated epithermal microscopic fission cross section ratios.

$$TP^{240} = (NV)^{240} \bar{\nu}_1^{240} \left[\frac{(\overline{\sigma_f \phi})_1^{239}}{\phi_{\text{Thermal Column}}} \left(\frac{\bar{\sigma}_f^{240}}{\bar{\sigma}_f^{239}} \right)_1 \right] \quad (21)$$

Similarly, total production by ^{238}U and ^{234}U were inferred from the epithermal ^{235}U fission rate and calculated epithermal microscopic fission cross section ratios.

$$TP^{238} = (NV)^{238} \bar{\nu}_1^{238} \left[\frac{(\overline{\sigma_f \phi})_1^{235}}{\phi_{\text{Thermal Column}}} \left(\frac{\bar{\sigma}_f^{238}}{\bar{\sigma}_f^{235}} \right)_1 \right] \quad (22)$$

Values of TP^{238} , TP^{240} , TP^{241} , and TP^{242} are dependent on calculations, but they contribute less than 5% of the total neutron production, TP , for all cases. Errors in these quantities tend to cancel in the determination of k_∞ using Equation (16), since they occur in both numerator and denominator in nearly equal proportion.

Thermal absorption, A_2 , was evaluated from relative thermal fission rates measured for ^{239}Pu and ^{235}U , and thermal absorption rates measured for ^{63}Cu . Values of the capture - to - fission ratio, $\bar{\alpha}$, for ^{239}Pu and ^{235}U , averaged over the thermal neutron spectrum in the fuel, were calculated using GRANIT.

$$A_2^{239} = (NV)^{239} \left[\frac{(\overline{\sigma_f \phi})_2^{239}}{\phi_{\text{thermal column}}} \right] (1 + \bar{\alpha})_2^{239} \quad (23)$$

and

$$A_2^{235} = (NV)^{235} \left[\frac{(\overline{\sigma_f \phi})_2^{235}}{\phi_{\text{thermal column}}} \right] (1 + \bar{\alpha})_2^{235} \quad (24)$$

Thermal absorption rates by all other isotopes in the fuel were evaluated from relative thermal absorption rates in ^{63}Cu in the fuel

$$A_2^i = (NV)^i \left[\frac{(\overline{\sigma_a \phi})_2^{63}}{\phi_{\text{thermal column}}} \left(\frac{\overline{\sigma_a}^i}{\overline{\sigma_a}^{63}} \right)_2 \right]. \quad (25)$$

A similar expression was used to infer thermal absorption rates for materials in the moderator and cladding, using relative thermal absorption rates for ^{63}Cu measured in those lattice components. Total thermal absorption rate in the center cell, A_2 , is the sum of all thermal absorption rates of the center cell constituents. The value of the flux in the thermal column, $\phi_{\text{thermal column}}$, is an arbitrary normalization factor which cancels in the evaluation of either the quantity (TP/A_2) for the unpoisoned lattice, or $(\frac{A_2 + A_{2a}}{TP})$ for the poisoned lattice. Resulting experimental values for k_∞ in the unpoisoned lattice for various fuel types are listed in Table VI. Quantities used in the evaluation of these experimentally - derived values of k_∞ are also listed in Table VI.

TABLE VI

ANALYSIS OF PuO_2 PARTICULATE EXPERIMENTS TO DETERMINE k_∞

Particulate Fuel Type	PuO_2 Particles Diameter (microns)	Fuel Density gm/cm^3	wt% PuO_2 in $\text{UO}_2 + \text{PuO}_2$	$\frac{\text{TP}}{A_2}$	$\frac{\text{TP}}{A_2 + A_{2a}}$	$\left(\frac{p}{p_s}\right)$	k_∞^*	k_∞
8BMB	0	8.571	1.98	1.685	1.292	1.011	1.319	1.314 ± 0.005
8SC	0	8.704	2.08	1.699	1.296	1.012	1.327	1.322 ± 0.005
8BME	0	8.474	2.10	1.699	1.296	1.012	1.326	1.321 ± 0.005
8P50	52	8.601	2.29	1.707	1.296	1.013	1.333	1.328 ± 0.005
8B100	107	8.739	2.48	1.707	1.299	1.013	1.331	1.326 ± 0.005
8P100	107	8.638	2.47	1.704	1.305	1.012	1.322	1.317 ± 0.005
8LD100	107	7.630	2.45	1.674	1.294	1.012	1.309	1.304 ± 0.005
8B200	195	8.718	2.35	1.667	1.275	1.012	1.323	1.318 ± 0.005
8P200	195	8.657	2.41	1.668	1.279	1.012	1.319	1.314 ± 0.005
8B350	328	8.691	2.42	1.619	1.255	1.011	1.304	1.299 ± 0.005
8RA350	328	8.798	2.47	1.630	1.254	1.011	1.315	1.310 ± 0.005
Vibrationally Compacted Fuels	Nominal PuO_2 Particle Diameter (microns)							
$\text{UO}_2 - 2.0 \text{ wt\% PuO}_2$	25	9.54	2.00	1.717	1.307	1.013	1.331	1.325 ± 0.005
$\text{UO}_2 - 0.9 \text{ wt\% PuO}_2$	25	9.38	0.87	1.315	1.192	1.002	1.106	1.104 ± 0.004

CALCULATIONAL METHODS

PuO₂ Particle Size Lattice Constants

A two group calculational model was used for analysis of the experiment and the experiment - theory correlation. The thermal group constants were calculated over the energy range 0 to 0.683 eV by the integral transport method with 30 neutron energy groups and 11 radial space points, using GRANIT.⁽¹⁹⁾ The granular composite fuel region consisted of PuO₂ particles in a UO₂ diluent surrounded by concentric cylindrical regions of zircalloy clad and water moderator. An isotropic albedo of unity was used for the outer boundary condition of the unit cell. Computed space and energy dependent spectra were used to generate one thermal group cross sections and reaction rates in the unit cell.

Epithermal group constants were calculated over the energy range 0.683 eV to 10 MeV using EGGNIT.⁽²⁰⁾ This code extends the Nordheim numerical integration method of resonance treatment to multiregion and granular composite geometries. Dancoff shielding correction was computed using DASQHE.⁽²²⁾ The 66 group slowing down spectrum calculated using the B₁ approximation and the ²³⁹Pu fission spectrum were used to generate one epithermal group cross sections and reaction rates in the unit cell.

Vibrationally Compacted UO₂ - PuO₂ Lattice Constants

Two group lattice constants for the vibrationally compacted UO₂ - PuO₂ fuels were calculated in the same manner as the particle size lattice constants. Thermal group constants were calculated over the energy range 0 to 0.683 eV with 30 neutron energy groups and 11 radial space points, using Battelle-Revised-THERMOS.⁽¹⁸⁾ An isotropic albedo of unity was used

for the outer boundary condition of the unit cell. Computed space and energy dependent spectra were used to generate one thermal group cross sections and reaction rates in the unit cell. Epithermal group constants were calculated over the energy range 0.683 eV to 10 MeV using HRG.⁽¹⁷⁾ The Dancoff correction factor and resonance spatial self shielding parameters that are required for HRG input were calculated using DASOHE.⁽²²⁾ The 66 group slowing down spectrum calculated using the B_1 approximation and the ^{239}Pu fission spectrum were used to generate one epithermal group cross sections and reaction rates in the unit cell.

Foil Perturbation Effect Calculations

Foil activation techniques were utilized in the determination of the relative reaction rates in all central cell constituents and the standard absorber (either copper or boron) inserted in the center of the PCTR. The presence of foil materials in the lattice perturbs the neutron flux in the vicinity of each foil, such that the induced activity is not truly representative of the flux in the surrounding unperturbed media. Thermal flux peaking will occur in a weakly absorbing foil irradiated in the highly absorbing fuel rod because of two principal reasons. The first, an edge effect, is caused by neutrons from the moderator streaming through the foil perpendicular to the axis of the fuel rod without appreciable absorption. The second effect, a positive source of neutrons at the foil location, is caused by the difference between the neutron absorption rate in the fuel and the neutron absorption rate in a foil. The neutron current away from the foil is dependent on the neutron transport properties of the surrounding media.

Two dimensional transport theory calculations in two energy groups were performed using DOTSN,⁽²³⁾ explicitly including foils in the lattice cell geometry used to measure activation rates. Thermal and epithermal group constants for each lattice composition were generated using GRANIT/EGGNIT and THERMOS/HRG for the particle size and vibrationally compacted fuels, respectively, as described in preceding sections. The R-Z geometries chosen for bare and cadmium covered foils irradiated in the fuel are shown in Figure 14.

Two radial models were used to check the method of cylindrification of the lattice surrounding the unit cell containing the foils. These models are shown in Figure 15. Reflecting boundaries were used at the outer surface and at the ends of both lattice models. Foils inserted into the lattice were positioned in the center of the 25.7 cm total axial length chosen for these calculations. Unperturbed (no foil) regions were selected at each end of the cell, 11.3 cm from the foils, where the flux perturbations from the foils were negligible, because the mean - free - path of neutrons in the lattice was small in comparison to the distance from the foils to the selected end regions.

The dimensions and compositions of the unit cell consisting of three cylindrical regions (a) fuel, (b) clad, and (c) moderator are listed in Tables I through III and Figure 14.

The axial perturbations of the two group fluxes in a vibrationally compacted UO_2 - 2.0 wt% PuO_2 fuel rod as calculated using DOTSN, are shown in Figures 16 and 17. The perturbations resulting from insertion of bare and cadmium covered foils into the fuel rod, relative to the unperturbed fluxes in the fuel rod, are shown in Figures 16 and 17, respectively. Differences in the neutron flux ratios because of the two different models

FIGURE 14: TWO-DIMENSIONAL GEOMETRIES FOR BARE AND CADMIUM COVERED FOILS IRRADIATED IN THE FUEL

FIGURE 15: RADIAL MODELS FOR THE LATTICE SURROUNDING THE UNIT CELL CONTAINING FOILS

FIGURE 16: NEUTRON FLUX RATIOS FOR BARE FOILS IRRADIATED IN THE FUEL

FIGURE 17: NEUTRON FLUX RATIOS FOR CADMIUM COVERED FOILS IRRADIATED IN THE FUEL

(Figure 15) for the radial lattice cells were negligible. However, a sufficient thickness of homogenized lattice region surrounding the foil location was used in the calculational model so that the foil perturbation effect was negligible at the reflecting boundaries.

Derivation of foil activation corrections has been presented in detail for thermal reactor lattices.⁽¹³⁾ Expressions for activation rates in the unperturbed (no foil) regions from two group theory are related to the measured values by

$$A_2^{\text{no foil}} = \frac{\left[A_{\text{total}}^{\text{bare, foil}} - A_{\text{total}}^{\text{Cd, foil}} \begin{matrix} \left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_{\text{Cd}} \\ \left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_{\text{bare}} \\ \left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_{\text{bare}} \end{matrix} \begin{matrix} 1 \\ 1 \\ 1 \end{matrix} \right] \left(\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right)_2}{1 - \frac{\left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_{\text{bare}} \left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_{\text{Cd}}}{\left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_{\text{Cd}} \left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_{\text{bare}}} \left[\frac{-\sigma_{\text{Cd, foil}}}{-\sigma_{\text{bare, foil}}} \right]_2} \quad (26)$$

$$A_1^{\text{no foil}} = \left[A_{\text{total}}^{\text{Cd, foil}} - \frac{A_2^{\text{no foil}}}{\left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_2} \begin{matrix} \left[\frac{-\sigma_{\text{Cd, foil}}}{-\sigma_{\text{bare, foil}}} \right]_{\text{Cd}} \\ \left[\frac{-\sigma_{\text{Cd, foil}}}{-\sigma_{\text{bare, foil}}} \right]_{\text{bare}} \end{matrix} \right] \left(\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right)_1, \quad (27)$$

where the superscripts bare and Cd refer to the bare foil and cadmium covered foil geometries, respectively. Subscripts 1 and 2 refer to activation produced by neutrons with energies > 0.683 eV, and < 0.683 eV, respectively.

The quantity $\left[\frac{\phi_{\text{no foil}}}{\phi_{\text{foil}}} \right]_i$ is the ratio of the volume-weighted average flux in the unperturbed region to the flux in the region containing a foil, calculated for both bare and cadmium covered foil geometries in two groups. The quantity $\left[\frac{-\sigma_{\text{bare, foil}}}{-\sigma_{\text{Cd, foil}}} \right]_2$ is the ratio of bare to cadmium covered foil activation cross sections for group 2.

The thermal-neutron spectrum-averaged cross section ratio, $(\bar{\sigma}_{\text{Cd, foil}} / \bar{\sigma}_{\text{bare, foil}})_2$ has been retained in the solutions since the group 2 flux goes to zero in the cadmium cover but can recover a few percent in the foils. This is especially true for foils that contain a scattering matrix such as the aluminum alloy foils. Due to conservation of energy and momentum, neutrons transferred from group 1 to group 2 by elastic scattering with aluminum retain a minimum energy

$$E'_{\min} = E \left[\frac{A - 1}{A + 1} \right]^2, \quad (28)$$

where A is the atomic mass of aluminum. This minimum energy equals 86% of the neutron energy, E, before collision. For example, in the cadmium covered foils the average energy $\bar{E}'$ of 0.683 eV neutrons elastically scattered into group 2 after a single collision is

$$\bar{E}' = \frac{1}{2} \left[1 + \left(\frac{A - 1}{A + 1} \right)^2 \right] 0.683 \text{ eV} = 0.636 \text{ eV}. \quad (29)$$

Using a more exact method that integrates over the distribution of initial neutrons, at energies between 0.683 and 0.792 eV where transfer from group 1 to group 2 is possible, the weighted average energy of group 2 neutrons in the cadmium covered foils, $\bar{E}'$, is 0.652 eV. Activation cross sections at this energy are compared with the unperturbed thermal-neutron spectrum-averaged cross sections to determine $(\bar{\sigma}_{\text{Cd, foil}} / \bar{\sigma}_{\text{bare, foil}})_2$. For the particular example described in this report, the effect of the difference in activation cross sections at 0.652 and 0.636 eV on the foil perturbation correction factors was negligible. In the determination it is assumed that the bare foil does not appreciably affect the shape of the thermal-neutron spectrum.

Perturbations caused by the Al - 1.8 wt% ^{239}Pu foils in a UO_2 - 2 wt% PuO_2 fuel rod were chosen to quantitatively demonstrate the method. Fission rates obtained from a bare and cadmium covered foil irradiated in the fuel rod are listed in Table VII. The values are ^{239}Pu fission rates relative to that of an identical foil irradiated simultaneously in a graphite thermal column. Also listed in Table VII are the two group results of calculations made using the DOTSN and THERMOS codes to obtain the quantities required for foil activation corrections. Using the values from Table VII in Equations 26 and 27 the activation rates in the unperturbed region

$$A_2^{\text{no foil}} = 8.501$$

$$A_1^{\text{no foil}} = 0.574$$

are obtained. A significant observation from this example is that foil perturbation corrections for group 1 and group 2 fluxes are of opposite sign. A 3.4% correction to the measured subcadmium reaction rate, due to flux peaking, and a 1.4% correction to the measured epicadmium reaction rate, due to flux dipping, result in a 4.8% reduction in the measured value for (cadmium ratio-1). An additive effect on the measured (cadmium ratio-1) is produced by flux perturbations due to insertion of foils in a fuel rod.

Additional DOTSN calculations for a bare 0.005-inch thick copper sector foil placed in the moderator gave values for $(\phi^{\text{foil}}/\phi^{\text{no foil}})$ of 0.998 and 0.994 for group 1 and group 2 fluxes respectively. The foil perturbation effects in the moderator are smaller in magnitude than those in the fuel since the absorbing properties of the foil and moderator are more nearly matched.

TABLE VII

EVALUATION OF UNPERTURBED ACTIVATION RATES

Al - 1.8 wt% ^{239}Pu Foil Irradiated in a UO_2 - 2 wt% PuO_2 Fuel Rod

Measured Quantities

$$A_{\text{total}}^{\text{Bare, foil}} = 9.365$$

$$A_{\text{total}}^{\text{Cd, foil}} = 0.566$$

$$(\text{Cd Ratio} - 1)^{\text{foils}} = 15.55$$

^{239}Pu fission rate of a Al - Pu foil irradiated in a fuel rod, relative to the ^{239}Pu fission rate of a Al - Pu foil irradiated simultaneously in a graphite thermal column, external to the reactor.

Calculated Quantities

$$\left(\frac{\phi^{\text{foil}}}{\phi^{\text{no foil}}} \right)_2^{\text{Bare}} = 1.0351$$

$$\left(\frac{\phi^{\text{foil}}}{\phi^{\text{no foil}}} \right)_2^{\text{Cd}} = 0.0359$$

$$\left(\frac{\sigma_{\text{Cd, foil}}}{\sigma_{\text{Bare, foil}}} \right)_2^{^{239}\text{Pu}} = 0.1261$$

$$\left(\frac{\phi^{\text{foil}}}{\phi^{\text{no foil}}} \right)_1^{\text{Bare}} = 0.9851$$

$$\left(\frac{\phi^{\text{foil}}}{\phi^{\text{no foil}}} \right)_1^{\text{Cd}} = 0.9192$$

DOTSN
Calculations

THERMOS Calculation

Activation Correction Factors

$$\frac{A_2^{\text{no foil}}}{A_{\text{total}}^{\text{Bare, foil}} - A_{\text{total}}^{\text{Cd, foil}}} = 0.966$$

$$\frac{(\text{Cd Ratio} - 1)^{\text{no foils}}}{(\text{Cd Ratio} - 1)^{\text{foils}}} = 0.952$$

$$\frac{A_1^{\text{no foil}}}{A_{\text{total}}^{\text{Cd, foil}}} = 1.014$$

Unperturbed Activation Rates

$$A_2^{\text{no foil}} = 8.501$$

$$A_1^{\text{no foil}} = 0.574$$

$$(\text{Cd Ratio} - 1)^{\text{no foils}} = 14.81$$

Corrections for foil perturbation effects using the two group transport theory calculational method were applied to all relative reaction rates listed in Table V. Results of the experimentally inferred two group reaction rates in the unperturbed media are listed in Table VIII. These reaction rates were used to evaluate values for k_{∞} in the unpoisoned lattice, in addition to the determination of the thermal utilization, f , the thermal disadvantage factor, $(\overline{\sigma\phi})_2^{H_2O} / (\overline{\sigma\phi})_2^{Fuel}$, the epithermal-to-thermal ^{238}U capture ratio, ρ^{28} , and the epithermal-to-thermal fission ratios for ^{239}Pu and ^{235}U , δ^{49} and δ^{25} , respectively, for each lattice composition. Corrections of the order of a few percent to the reaction rates can be significant in a theory-experiment correlation. However, the effect of these corrections on the values for k_{∞} inferred from experiment in the unpoisoned lattice is small, since the corrections tend to cancel in the analysis using Equation (16). If corrections for foil perturbation effects were not applied to the experimental data, values inferred for k_{∞} in the unpoisoned UO_2 - 2.0 wt% PuO_2 fueled lattice would be lowered less than one percent. Values for k_{∞} of unity (measured in the poisoned lattice) are not affected because activation rates are not used in the analysis.

TABLE VIII
EXPERIMENTAL VALUES FOR REACTION RATES IN UNPERTURBED LATTICE

UO ₂ - PuO ₂ Lattice (Zero Micron PuO ₂ Particulate Fuel Type 8SC)				
Reaction Rate	Unpoisoned Lattice		Poisoned Lattice	
	Group 2	Group 1	Group 2	Group 1
$\frac{(\sigma_a \phi)_{\text{Cu-63}}}{(\sigma_a \phi)_{\text{H}_2\text{O}}}$	4.56x10 ¹	4.44	1.18x10 ¹	1.65
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_a \phi)_{\text{Cu-63}}}{(\sigma_a \phi)_{\text{Fuel}}}$	3.39x10 ¹	4.40	8.73	1.63
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_f \phi)_{235}}{(\sigma_f \phi)_{\text{Fuel}}}$	4.16x10 ³	3.05x10 ²	1.07x10 ³	1.14x10 ²
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_f \phi)_{239}}{(\sigma_f \phi)_{\text{Fuel}}}$	7.15x10 ³	4.39x10 ²	1.88x10 ³	1.62x10 ²
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_c \phi)_{238}}{(\sigma_c \phi)_{\text{Fuel}}}$	2.07x10 ¹	2.24x10 ¹	5.33	8.87
$\phi_{\text{Thermal Column}}$				

UO ₂ - PuO ₂ Lattice (328 Micron Diameter PuO ₂ Particulate Fuel Type 8RA350)				
Reaction Rate	Unpoisoned Lattice		Poisoned Lattice	
	Group 2	Group 1	Group 2	Group 1
$\frac{(\sigma_a \phi)_{\text{Cu-63}}}{(\sigma_a \phi)_{\text{H}_2\text{O}}}$	4.68x10 ¹	4.45	1.26x10 ¹	1.68
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_a \phi)_{\text{Cu-63}}}{(\sigma_a \phi)_{\text{Fuel}}}$	3.48x10 ¹	4.37	9.35	1.66
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_f \phi)_{235}}{(\sigma_f \phi)_{\text{Fuel}}}$	4.30x10 ³	3.02x10 ²	1.15x10 ³	1.77x10 ²
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_f \phi)_{239}}{(\sigma_f \phi)_{\text{Fuel}}}$	5.13x10 ³	3.08x10 ²	1.46x10 ³	1.12x10 ²
$\phi_{\text{Thermal Column}}$				

UO ₂ - 0.9 wt% PuO ₂ Vibrationally Compacted Fuel (See Table II)				
Reaction Rate	Unpoisoned Lattice		Poisoned Lattice	
	Group 2	Group 1	Group 2	Group 1
$\frac{(\sigma_a \phi)_{\text{Cu-63}}}{(\sigma_a \phi)_{\text{H}_2\text{O}}}$	3.39x10 ¹	1.80	2.04x10 ¹	1.28
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_a \phi)_{\text{Cu-63}}}{(\sigma_a \phi)_{\text{Fuel}}}$	2.77x10 ¹	1.80	1.69x10 ¹	1.29
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_f \phi)_{235}}{(\sigma_f \phi)_{\text{Fuel}}}$	3.42x10 ³	1.32x10 ²	2.07x10 ³	9.45x10 ¹
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_f \phi)_{239}}{(\sigma_f \phi)_{\text{Fuel}}}$	5.76x10 ³	1.67x10 ²	3.52x10 ³	1.13x10 ²
$\phi_{\text{Thermal Column}}$				
$\frac{(\sigma_c \phi)_{238}}{(\sigma_c \phi)_{\text{Fuel}}}$	1.72x10 ¹	1.28x10 ¹		
$\phi_{\text{Thermal Column}}$				

PCTR Calculational Model

A two-dimensional (R-Z) diffusion theory model of the PCTR with the water tank installed in the test cavity was obtained by radial cylindrification of equivalent volumes for all PCTR zones, using the 2DB Code.⁽⁹⁾ This calculational model was used (a) to predict neutron spectrum equilibration and central cell end effects, (b) to evaluate small corrections to null reactivity boron concentrations due to neutron spectrum mismatch in the center cell and flux gradients between the radial buffer and center cell, and (c) to predict critical reactor fuel loadings for pre-experiment analysis.

To evaluate end cap effects on the axial averaged reaction rates in the center cell fuel with diffusion theory, an integral transport theory correction to the thermal neutron absorption cross section in the central cell fuel rod was used. A one dimensional, two group, diffusion theory calculation for a lattice unit cell using HFN⁽²⁴⁾ indicated that the thermal flux depression within the fuel is much less than that calculated by integral transport theory⁽¹⁸⁾ (see Figure 18). The argument of the Bessel function solution for the thermal flux in the fuel rod, $\phi_2(R) = \phi_2(0)I_0(\kappa R)$, in the diffusion theory calculation was changed to match the thermal flux profile in the fuel calculated by transport theory. Since $\kappa^2 = \Sigma_a/D$, the thermal-neutron absorption cross section of the fuel was changed, keeping the diffusion coefficient constant, to calculate two-dimensional thermal flux profiles in the center cell fuel rod, using diffusion theory in the 2DB PCTR calculational model.

The central test region of the water tank is composed of three parts: the center test cell and its two end buffers. Results from the previous light water moderated, unpoisoned lattice experiment⁽²⁾ indicate that thermalized

FIGURE 13: COMPARISON OF CALCULATED RADIAL THERMAL FLUX PROFILES

neutrons accumulate in the gap between the center cell and its end buffers due to the moderating effect, short mean free path, and because of the smaller absorption cross section in the gap region. The test fuel rod with this particular gap of zirconium and lucite, had an axially averaged (normalized to unity at the reactor centerline) thermal flux of 1.01954 (see Figure 19).

Comparison of the calculated axial variation of the thermal flux on the outer surface of the central cell fuel clad with measured sub-cadmium activation rates of 0.20-inch diameter gold pins on the clad surface,⁽²⁾ is shown in Figure 20. The good agreement in the thermal flux peaking in the ends of the lucite and polyethylene central cell containers builds confidence in the model.

The 2DB calculations were repeated using an aluminum filled gap (the lucite was replaced with aluminum end caps) such that a configuration having minimum thermal peaking effect without significant moderation or absorption could be found. The results of these calculations show that the axially averaged thermal flux with aluminum end caps is 1.00063 for the unpoisoned lattice and 1.01128 for the poisoned lattice (1315 wppm boron in the moderator of the test cell). The peaking effect is somewhat greater in the poisoned lattice but, intuitively, should be expected. With this being the case, the error due to thermal flux peaking with aluminum end caps was considered to be a reasonable minimum. The central test cell of the water tank, shown in Figure 3, was redesigned using aluminum end caps based on this result. A profile of the thermal flux peaking in the center cell, near the end caps, can be seen in Figures 21 and 22 for the unpoisoned and poisoned lattices, respectively.

FIGURE 19: AVERAGE THERMAL FLUX IN THE CENTER CELL FUEL UNPOISONED LATTICE WITH LUCITE END CAPS

FIGURE 20: AXIAL VARIATION OF THERMAL FLUX ON THE OUTER SURFACE OF THE FUEL CLAD

FIGURE 21: AVERAGE THERMAL FLUX IN THE CENTER CELL FUEL
 UNPOISONED LATTICE WITH ALUMINUM END CAPS

FIGURE 22: AVERAGE THERMAL FLUX IN THE CENTER CELL FUEL POISONED LATTICE WITH ALUMINUM END CAPS

The measurement of k_{∞} requires that the neutron energy distribution in the center test cell duplicates the neutron spectrum found in the center of a bare, just critical, array of identical cells. This is accomplished in the PCTR by surrounding the test cell with a layer of medium, the buffer, which is identical to the test sample and which brings the spectrum into equilibrium, i.e., alters the flux incident upon the test cell such that the epithermal-to-thermal flux ratio remains constant across the center test cell. Plots of the flux ratio versus distance from center cell are shown in Figures 23 and 24 for the unpoisoned and poisoned lattices, respectively. It can be seen that a sufficient volume of buffer has been loaded, making the neutron spectrum mismatch negligible, in both lattices.

The two-dimensional diffusion theory mockup of the PCTR water tank core was used in confirming and predicting critical loadings. The eight "open" control rods in the PCTR were not included in the PCTR mockup.

Results from the 2DB calculational model yielded a k_{eff} value of 1.055 for the unpoisoned water tank core. The eight "open" control rods are worth ~ 51 mk, resulting in a calculated k_{eff} value of approximately 1.004 for the critical loading. This is in good agreement with the measured unpoisoned critical loading,⁽²⁾ and builds confidence in the PCTR calculational mockup. The calculated k_{eff} value for the poisoned PCTR water tank loading (including the eight "open" control rods) is 1.017. This result is reasonable since the same multigroup constants for the PCTR driver fuel were used in the poisoned lattice loading as were used in the unpoisoned lattice loading. Twice as many drivers were required to achieve criticality in the poisoned lattice, shown in Figure 25, as were required in the unpoisoned lattice, shown in Figure 26. Corrections for the increased self-shielding would reduce the calculated k_{eff} for the poisoned lattice loading.

FIGURE 23: NEUTRON SPECTRUM EQUILIBRATION IN THE UNPOISONED LATTICE

FIGURE 24: NEUTRON SPECTRUM EQUILIBRATION IN THE LATTICE POISONED TO MULTIPLICATION OF UNITY

X - DRIVER
 C - 6" A1-U FUEL
 J - 8" A1-U FUEL
 POISON USED - YES ☒ NO ☐

H - H TUBE
 E - EMPTY
 N - 0.5" NAT. U
 ● - CONTROL ROD

LEVELING SLUG POSITIONS - INCHES OUT

	INNER	MIDDLE	OUTER
FRONT	0.0"	0.0"	0.0"
REAR	0.0"	0.0"	0.0"

71 DRIVERS

128 NATURAL URANIUM FLUX MODIFIERS (1/2" DIA X 36")

36 UO_2 -2.0 WT% PuO_2 RODS IN CENTER OF WATER TANK

1315 WPPM BORON IN H_2O , OR EQUIVALENT BORATED POLYETHYLENE

72 UO_2 -0.9 WT% PuO_2 RODS SURROUNDING TEST LATTICE

BORATED POLYETHYLENE EQUIVALENT TO 276.3 WPPM BORON IN H_2O

FIGURE 25: MEASURED CRITICAL LOADING FOR POISONED WATER LATTICE

X - DRIVER
 C - 6" A1-U FUEL
 J - 8" A1-U FUEL
 POISON USED - YES ☐ NO ☒

H - H TUBE
 E - EMPTY
 N - 0.5" NAT. U.
 ● - CONTROL ROD

LEVELING SLUG POSITIONS - INCHES OUT

	INNER	MIDDLE	OUTER
FRONT	0.0"	0.0"	0.0"
REAR	0.0"	0.0"	0.0"

35 DRIVERS

128 NATURAL URANIUM FLUX MODIFIERS (1/2" DIA X 36")

36 UO₂-2.0 WT% PuO₂ RODS IN CENTER OF WATER TANK - UNPOISONED

72 UO₂-0.9 WT% PuO₂ RODS SURROUNDING TEST LATTICE - UNPOISONED

FIGURE 26: MEASURED CRITICAL LOADING FOR UNPOISONED WATER LATTICE

The radial buffer fuel of the UO_2 - 2.0 wt% PuO_2 lattice had slightly different absorption and fission cross sections than the finite particle sized UO_2 - PuO_2 fuel loaded in the center cell. The two-dimensional PCTR calculational model was used to evaluate small corrections to the measured null-reactivity concentrations for the particulate fueled lattices. These corrections were due to neutron spectrum mismatch in the center cell, and flux gradients between the radial buffer and the center cell. Calculations were made for each center cell fuel composition and null-reactivity boron concentration in the water, (a) with the center cell installed, and (b) with the center cell removed. Pseudo-scattering cross sections ($\Sigma_{1 \rightarrow 1} = \Sigma_{2 \rightarrow 2} = 0.1$; $\Sigma_{1 \rightarrow 2} = \Sigma_{2 \rightarrow 1} = 0$) were used to simulate a vacuum and avoid large DB^2 terms in the void. Eigenvalue convergence of 5×10^{-6} was specified in the 2DB calculations.

The neutron multiplication of the PCTR with the center cell installed can be expressed in terms of production, P, and absorption, A

$$K_F = \frac{P_F}{A_F}, \quad (30)$$

and in the region considered as the center cell

$$K_C = \frac{P_C}{A_C}. \quad (31)$$

The value of k outside the center cell region is

$$k_0 = \frac{P_F - P_C}{A_F - A_C} = \frac{A_F K_F - A_C K_C}{A_F - A_C} \quad (32)$$

since the neutron multiplication in the region of the center cell and its intermediate surroundings has no leakage effects. The change in k of the region immediately surrounding the void is a measure of the effect of flux redistribution in the test lattice when the center cell is removed.

The value of k_{eff} for the PCTR when the center cell is removed will change due to redistribution effects. In terms of the region outside the center cell region

$$k_E = k'_0 = k_0 + \delta k_0. \quad (33)$$

The value of δk_0 is related to the error in the measured value of the center cell multiplication, δk_g , due to flux gradient effects by the expression

$$\delta k_g = \delta k_0 \left[\frac{A_0}{A_c} \right] = \delta k_0 \left[\frac{A_F - A_c}{A_c} \right]. \quad (34)$$

Substituting for δk_0 the correction to the measured multiplication of the center cell was evaluated using the expression

$$\delta k_g = \left[k_E - \left(\frac{A_F k_F - A_c k_c}{A_F - A_c} \right) \right] \left(\frac{A_F - A_c}{A_c} \right). \quad (35)$$

The correction to the measured multiplication due to neutron spectral mismatch was evaluated using the flux ratio (ϕ_1/ϕ_2) in the center cell calculated with the 2DB PCTR model, and the fundamental mode flux ratio (ϕ'_1/ϕ'_2) calculated from two group constants for the poisoned lattice using the neutron balance

$$\Sigma_{1 \rightarrow 2} \phi'_1 = \left(\Sigma_{a2} + D_2 B^2 \right) \phi'_2. \quad (36)$$

Since the buckling is zero in the fundamental mode spectrum of the poisoned lattice (i.e., $k_\infty = 1.0$), the flux ratio is given by

$$\frac{\phi'_1}{\phi'_2} = \frac{\Sigma_{a2}}{\Sigma_{1 \rightarrow 2}}. \quad (37)$$

The correction to the measured multiplication of the center cell due to neutron spectral mismatch was evaluated using the expression

$$\delta k_m = \frac{v\Sigma_{f1}[\phi_1/\phi_2] + v\Sigma_{f2}}{\Sigma_{a1}[\phi_1/\phi_2] + \Sigma_{a2}} - \frac{v\Sigma_{f1}[\phi'_1/\phi'_2] + v\Sigma_{f2}}{\Sigma_{a1}[\phi'_1/\phi'_2] + \Sigma_{a2}} \quad (38)$$

The total correction to the measured multiplication in the center cell, δk_c , is the sum of the contributions due to flux gradients between the radial buffer and the center cell, and neutron spectral mismatch in the center cell

$$\delta k_c = \delta k_g + \delta k_m. \quad (39)$$

Calculated boron sensitivities, $(\Delta wppm \text{ Boron} / \Delta k_\infty)$, were used to adjust the measured null-reactivity concentrations of boron in the water to correspond to $k_\infty \equiv 1.000$ in the matched spectrum of each center cell. These corrections are listed in Table IX.

TABLE IX

PuO₂ PARTICULATE EXPERIMENT-BORON CONCENTRATION ADJUSTMENTS

Particle Size Fuel Type	PuO ₂ Particle Diameter (microns)	Null Reactivity Boron Concentration in H ₂ O	Flux Gradient δk_g	Spectrum Mismatch δk_m	Total Correction δk_c	$\frac{\Delta \text{wppm Boron}}{\Delta k_\infty}$	Boron Concentration Adjustment (wppm)	Experimental Boron Concentration in H ₂ O for $k_\infty \approx 1.000$
8BMB	0	1225 $\pm$ 3	0.00168	0.00065	0.00233	-5193	-12	1213 $\pm$ 4
8SC	0	1294 $\pm$ 3	0.00080	0.00065	0.00145	↓	- 8	1286 $\pm$ 4
8BME	0	1274 $\pm$ 3	0.00025	-0.00015	0.00010	↓	0	1274 $\pm$ 4
8P50	52	1352 $\pm$ 3	0.00005	0.00025	0.00030	-5114	- 2	1350 $\pm$ 4
8P100	107	1334 $\pm$ 3	0.00002	0.00034	0.00036	-5031	- 2	1332 $\pm$ 4
8B100	107	1380 $\pm$ 3	-0.00003	0.00047	0.00044	↓	- 2	1378 $\pm$ 4
8LD100	107	1226 $\pm$ 3	0.00060	-0.00080	-0.00020	↓	+ 1	1227 $\pm$ 4
8B200	195	1277 $\pm$ 3	0.00058	-0.00022	0.00036	-4898	- 2	1275 $\pm$ 4
8P200	195	1262 $\pm$ 3	0.00050	-0.00030	0.00020	↓	- 1	1261 $\pm$ 4
8B350	328	1179 $\pm$ 3	0.00260	0.00080	0.00340	-4697	-16	1163 $\pm$ 4
8RA350	328	1236 $\pm$ 3	0.00140	0.00006	0.00146	↓	- 7	1229 $\pm$ 4
Vibrationally Compacted Fuels	Nominal PuO ₂ Particle Diameter (microns)							
UO ₂ -2.0 wt% PuO ₂	25	1353 $\pm$ 3	0.0	0.0	0.0	-5155	0	1353 $\pm$ 3
UO ₂ -0.9 wt% PuO ₂	25	276 $\pm$ 3	0.0	0.0	0.0	-2654	0	276 $\pm$ 3

Experiment-Theory Correlation

The results of the PCTR experiments can be correlated with theory on the basis of integrated neutronic parameters. The most direct correlation can be made in the poisoned lattices by a comparison between the measured values of $k_{\infty} \equiv 1$ and the values of k_{∞} calculated for the poisoned lattices assuming the concentrations of boron in the moderator that were experimentally determined to produce null-reactivity in the PCTR.⁽⁶⁾

The uncertainties of two experimentally determined quantities could influence the calculated results: plutonium enrichment, and boron concentration. The uncertainty on the plutonium enrichments of the particulate fuel rods⁽³⁾ is estimated to be ± 0.04 wt% PuO_2 , corresponding to an uncertainty in k_{∞} of ± 0.004 . The uncertainty on the boron concentrations in the moderator for $k_{\infty} \equiv 1$ is estimated to be ± 4 weight-parts-per-million (wppm), corresponding to an uncertainty in k_{∞} of ± 0.0007 . Thus, the probable uncertainty of the calculated values of k_{∞} due to these measured quantities is of the same size as the root-mean-square (rms) deviation of the calculated values from the average, as given in Table IV. This result shows that the effect of a change in PuO_2 particle size on neutron multiplication in the poisoned lattices can be calculated within the uncertainties on the measured quantities. However, the GRANIT-EGGNIT calculations consistently underestimate the experimental values of $k_{\infty} \equiv 1$ by about one percent.

A clean, straightforward comparison between experiment and calculation for the unpoisoned lattice is useful for evaluation of analytical methods used to predict UO_2 - PuO_2 fuel performance in light water moderated reactors. Such a correlation is applicable to the interior of a reactor fuel bundle without boron in the water, where the space and energy dependent neutron spectrum in a unit cell approaches the fundamental mode condition, which includes the effect of fundamental mode neutron leakage from the lattice. Comparison

of measured and calculated values for the neutron multiplication factor of the unpoisoned UO_2 - PuO_2 particulate fueled lattice, listed in Table X, shows that GRANIT-EGGNIT calculations consistently overestimate k_∞ by three percent. However, the effect of PuO_2 particle size on neutron multiplication in the unpoisoned lattice is calculated within the statistical uncertainty of the measured change in k_∞ as a function of PuO_2 particle size, similar to the results in the poisoned lattice. These results indicate that an unresolved problem exists in the space and energy dependent neutron spectrum calculation obtained from the THERMOS⁽²⁵⁾ portion of the GRANIT code.

Differences in the ratio of calculated to measured neutron multiplication factors between the unpoisoned and the poisoned lattices can be explained by experiment-theory correlation of lattice parameters listed in Table XI. Calculations for the UO_2 - PuO_2 particulate fueled lattices consistently overestimate the thermal disadvantage factor, $(\overline{\sigma\phi})_2^{\text{H}_2\text{O}}/(\overline{\sigma\phi})_2^{\text{Fuel}}$, by about 14 percent (i.e., 14 percent overestimate of absorptions in the water), which results in an underestimate of the thermal utilization, f , of 2 percent in the unpoisoned lattice and 6 percent in the poisoned lattice. The 14 percent overestimate of the neutron absorption rate in the boron dissolved in the water accounts for the change in the ratio of calculated-to-measured thermal utilizations for the unpoisoned and poisoned lattices. If the overestimation of the thermal disadvantage factor were the only calculational problem, the neutron multiplication factor, k_∞ , would be underestimated for all cases.

Sources of the remaining differences between theory and experiment can be inferred from a neutron balance study for the poisoned lattice. Relative reaction rates measured in the poisoned lattice were used to evaluate the quantity $(\frac{\text{TP}}{A_2' + A_{2a}'})$ for each fuel composition, listed in Table VI, using calculated values for $\bar{\nu}$ and the thermal capture-to-fission ratio, $\bar{\alpha}_2$. Since there is no leakage from the lattice when poisoned to k_∞ of unity, the quantity

TABLE X

PuO₂ PARTICULATE EXPERIMENT-THEORY CORRELATION OF UNPOISONED LATTICE k_{∞}

Particle Size Fuel Type	$\eta_2 f_2$	p	$\eta_1 f_1$	$\tau(\text{cm}^2)$	$L^2(\text{cm}^2)$	Experimental k_{∞}	Calculated k_{∞}	Ratio($\frac{\text{Calculation}}{\text{Experiment}}$)
8BMB	1.481	0.840	0.661	30.85	2.27	1.314 ± 0.005	1.346	1.024
8SC	1.495	0.837	0.667	30.62	2.19	1.322 ± 0.005	1.355	1.025
8BME	1.494	0.839	0.669	30.89	2.21	1.321 ± 0.005	1.356	1.026
8P50	1.504	0.835	0.684	30.62	2.16	1.328 ± 0.005	1.364	1.027
8B100	1.514	0.832	0.697	30.25	2.11	1.326 ± 0.005	1.371	1.034
8P100	1.511	0.834	0.696	30.41	2.13	1.317 ± 0.005	1.370	1.040
8LD100	1.487	0.849	0.691	31.91	2.33	1.304 ± 0.005	1.361	1.044
8B200	1.484	0.836	0.695	30.50	2.25	1.318 ± 0.005	1.349	1.024
8P200	1.488	0.836	0.697	30.51	2.32	1.314 ± 0.005	1.353	1.030
8B350	1.463	0.839	0.705	30.57	2.35	1.299 ± 0.005	1.334	1.027
8RA350	1.470	0.837	0.709	30.41	2.31	1.310 ± 0.005	1.341	1.024
Vibrationally Compacted Fuels						Average Value = 1.030 Root - Mean - Square Deviation = ± 0.007		
UO ₂ - 2.0 wt% PuO ₂	1.507	0.833	0.674	30.75	2.09	1.325 ± 0.005	1.365	1.030
UO ₂ - 0.9 wt% PuO ₂	1.204	0.864	0.512	30.07	3.22	1.104 ± 0.004	1.105	1.001

TABLE XI

EXPERIMENT-THEORY CORRELATION OF LATTICE PARAMETERS

UO₂-2.0 wt% PuO₂ (zero micron PuO₂ particulate fuel)

Lattice Parameter	Unpoisoned Lattice			Poisoned Lattice		
	Experiment	Calculation	Ratio $\left(\frac{\text{Calculation}}{\text{Experiment}}\right)$	Experiment	Calculation	Ratio $\left(\frac{\text{Calculation}}{\text{Experiment}}\right)$
k_{∞}	1.314 ± 0.005	1.346	1.024 ± 0.004	1.000 ± 0.004	0.990	0.990 ± 0.004
f_2	0.847 ± 0.008	0.825	0.97 ± 0.01	0.630 ± 0.006	0.593	0.94 ± 0.01
$(\overline{\sigma\phi})_2^{\text{H}_2\text{O}} / (\overline{\sigma\phi})_2^{\text{Fuel}}$	1.34 ± 0.01	1.56	1.16 ± 0.01	1.35 ± 0.01	1.55	1.14 ± 0.01
$(\overline{\sigma}_f^{25} / \overline{\sigma}_f^{49})_2^{\text{Fuel}}$	0.58 ± 0.01	0.62	1.07 ± 0.02	0.57 ± 0.01	0.59	1.03 ± 0.02

UO₂-2.5 wt% PuO₂ (328 micron PuO₂ particulate fuel)

k_{∞}	1.310 ± 0.005	1.341	1.024 ± 0.004	1.000 ± 0.004	0.984	0.984 ± 0.004
f_2	0.824 ± 0.008	0.818	0.99 ± 0.01	0.618 ± 0.006	0.586	0.95 ± 0.01
$(\overline{\sigma\phi})_2^{\text{H}_2\text{O}} / (\overline{\sigma\phi})_2^{\text{Fuel}}$	1.35 ± 0.01	1.51	1.12 ± 0.01	1.34 ± 0.01	1.50	1.12 ± 0.01
$(\overline{\sigma}_f^{25} / \overline{\sigma}_f^{49})_2^{\text{Fuel}}$	0.82 ± 0.01	0.84	1.02 ± 0.02	0.78 ± 0.01	0.80	1.02 ± 0.02

UO₂-0.9 wt% PuO₂ (nominally 25 micron PuO₂) Vibrationally Compacted Fuel

k_{∞}	1.104 ± 0.004	1.105	1.001 ± 0.004	1.000 ± 0.002	0.989	0.989 ± 0.002
f_2	0.755 ± 0.008	0.742	0.98 ± 0.01	0.679 ± 0.007	0.658	0.97 ± 0.01
$(\overline{\sigma\phi})_2^{\text{H}_2\text{O}} / (\overline{\sigma\phi})_2^{\text{Fuel}}$	1.22 ± 0.01	1.27	1.04 ± 0.01	1.21 ± 0.01	1.27	1.05 ± 0.01
$(\overline{\sigma}_f^{25} / \overline{\sigma}_f^{49})_2^{\text{Fuel}}$	0.62 ± 0.01	0.64	1.03 ± 0.02	0.59 ± 0.02	0.63	1.07 ± 0.03

$\left(\frac{TP'}{A_2' + A_{2a}'}\right)$ can also be inferred from a neutron balance, using the calculated escape-to-thermal probability, p' , $\left(\frac{TP'}{A_2' + A_{2a}'}\right) = \frac{k_{\infty}'}{p'} = \frac{1}{p'}$. (40)

Calculated values for the quantity $1/p'$ were consistently lower than experimentally determined values for $\left(\frac{TP'}{A_2' + A_{2a}'}\right)$ by 5 percent ($\pm 1.5\%$ rms deviation) for all $UO_2 - PuO_2$ particulate fueled lattices poisoned to unit multiplication.

One possible source of error in the calculations, which could account for about 0.5 percent of the observed difference between experiment and theory is the ^{238}U resonance integral. The dilute resonance integral for ^{238}U calculated from resonance parameter data, used in EGGNIT and HRG, is only 269 barns.⁽²⁶⁾ The recommended value of 280 barns for the ^{238}U dilute resonance integral⁽²⁶⁾ is based on accurate integral measurements. Such a 4 percent increase in epithermal ^{238}U capture would decrease the escape to thermal probability by about 0.5 percent.

The other possible source of error in the calculations, which could account for the remaining 4 percent difference between experiment and theory, is the spectrum averaged value of $\bar{\eta}^{239}$. About 80 percent of all fissions in the lattice are due to thermal neutrons absorbed in ^{239}Pu . Since calculated values for $\bar{\nu}$ and $\bar{\alpha}$ for ^{239}Pu in the range of thermal neutron energies were used in Equations (19) and (23) to evaluate the quantity $\left(\frac{TP'}{A_2' + A_{2a}'}\right)$, the value for $\bar{\eta}_2^{239} = \left(\frac{\bar{\nu}_2^{239}}{1 + \bar{\alpha}_2^{239}}\right)$ can be adjusted to obtain a neutron balance from Equation (40).

Values for $\bar{\eta}_2^{239}$, calculated for the $UO_2 - PuO_2$ particulate fueled lattice using GRANIT, must be reduced by 5 percent ($\pm 1.5\%$ rms deviation), to obtain a neutron balance for experiments with a poisoned lattice, with k_{∞} of unity. Similarly, the calculated value for $\bar{\eta}_2^{239}$ must be reduced by 2 percent to obtain a neutron balance for the poisoned $UO_2 - 0.9 \text{ wt\% } PuO_2$ fueled lattice, with k_{∞} of unity.

Energy-dependent fission rates in ^{239}Pu for two $\text{UO}_2 - \text{PuO}_2$ particulate fuels and the $\text{UO}_2 - 0.9 \text{ wt\% PuO}_2$ fuel, calculated from neutron spectra shown in Figure 27 using GRANIT for the water-moderated lattice poisoned to k_∞ of unity, are shown in Figure 28. The energy-dependent capture-to-fission ratio, $\alpha(E)$, for ^{239}Pu is also shown in Figure 28 to illustrate the sensitivity of η_2^{239} to changes in the neutron spectrum between 0.1 and 0.4 eV. The Nelkin scattering kernel for light water⁽²⁷⁾ and cross sections from the BNW master library were used for the GRANIT calculations. Since the cross sections of ^{239}Pu from the BNW master library are in agreement with the latest values of Gwin, et.al.⁽²⁸⁾, measured over the energy region 0.02 eV to 1.0 eV, these results support the contention that errors in the calculation of thermal neutron spectra, rather than in the cross section data, are the major sources of the differences between measured and calculated integral lattice parameters.

Time-of-flight thermal neutron spectra measured in water-moderated lattices of Al -5.0 wt% Pu fuel rods⁽²⁹⁾ show about 40 percent shallower dips around the 0.3 eV resonance of ^{239}Pu , and softer spectra below 0.02 eV, than are calculated. The more dependent on moderator properties (scattering kernel) the spectrum is, the larger the discrepancy between measurement and calculation. The fuel rods used for neutron spectrum measurements were the same diameter and had nearly the same ^{239}Pu nuclei density as the $\text{UO}_2 - 2.0 \text{ wt\% PuO}_2$ solid solution fuel used in the PCTR experiments. The $(\text{H}_2\text{O}/\text{Rod})$ volume ratios of the two lattices used for neutron spectrum measurements (2.8 and 5.0) bracket the $(\text{H}_2\text{O}/\text{Rod})$ volume ratio of 3.5 used for the PCTR experiments. If the neutron spectra in the fuel, calculated using THERMOS and GRANIT, are overly depressed around 0.3 eV, and below 0.02 eV, similar to the GTF calculations in Reference 29, the thermal ^{235}U to ^{239}Pu fission ratio, $(\sigma_f^{25}/\sigma_f^{49})_2$, and η_2^{239} will be

FIGURE 27: AVERAGE NEUTRON SPECTRUM IN PLUTONIUM FOR PARTICULATE FUEL RODS, CALCULATED USING GRANIT

FIGURE 28: ENERGY DEPENDENT FISSION RATES IN PLUTONIUM FOR THE LATTICE POISONED TO K_{∞} OF UNITY, CALCULATED USING GRANIT

overestimated. A comparison of the measured and calculated lattice parameters, listed in Table XI shows that calculations overestimate $(\bar{\sigma}_f^{25}/\bar{\sigma}_f^{49})_2$, by about 4 percent ($\pm 2\%$ rms deviation).

Energy-dependent fission rates for ^{235}U and ^{239}Pu , calculated for the lattice poisoned to k_∞ of unity using GRANIT, are shown in Figures 29, 30, and 31 for the UO_2 - 2.0 wt% PuO_2 (zero micron diameter PuO_2 particulate) fuel, the UO_2 - 2.5 wt% PuO_2 (328 micron diameter PuO_2 particulate) fuel, and the UO_2 - 0.9 wt% PuO_2 (nominally 25 micron diameter PuO_2 particles) vibrationally compacted fuel, respectively. Calculated fission rate fractions between 0.2 and 0.4 eV, and below 0.02 eV, must be increased by about 40 percent to give agreement between calculated and measured values for $(\bar{\sigma}_f^{25}/\bar{\sigma}_f^{49})_2$ in the UO_2 - 2.0 wt% PuO_2 (zero micron diameter PuO_2 particulate), and UO_2 - 2.5 wt% PuO_2 (328 micron diameter PuO_2 particulate) fuels. Increasing the energy-dependent fission rates by about 40 percent in those energy regions results in a decrease in the value for $\bar{\eta}_2^{239}$ of approximately 4 percent. Only a 20 percent change in the energy-dependent fission rate fractions was required to give agreement between calculated and measured values for $(\bar{\sigma}_f^{25}/\bar{\sigma}_f^{49})_2$ in the UO_2 - 0.9 wt% PuO_2 (nominally 25 micron diameter PuO_2 particles) fuel. The calculated value for $\bar{\eta}_2^{239}$ decreased by about 2 percent due to the adjustment of the calculated neutron spectrum in the UO_2 - 0.9 wt% PuO_2 fuel. Thus, if the calculated neutron spectrum in the fuel is adjusted to give a value for $\bar{\eta}_2^{239}$ which balances Equation (40), the calculated and measured values for $(\bar{\sigma}_f^{25}/\bar{\sigma}_f^{49})_2$ are also brought into agreement.

The expression used to evaluate the experimental values of k_∞ for the unpoisoned lattice is not sensitive to the calculated values used for $\bar{\eta}_2^{239}$ since errors in the values of $\bar{\eta}_2^{239}$ used to evaluate the quantity (TP/A_2) tend to cancel similar errors in values of $\bar{\eta}_2^{239}$ used to evaluate $(\text{TP}'/A_2' + A_{2a}')_2$ in Equation (16). If the calculated values for $\bar{\eta}_2^{239}$ were reduced 4 percent

FIGURE 29: Relative Fission Rates in UO_2 -2.0 wt% PuO_2 (Zero Micron Diameter PuO_2) Calculated Using GRANIT.

FIGURE 30: Relative Fission Rates in UO_2 -2.5 wt% PuO_2 (328 Micron Diameter PuO_2)
Calculated Using GRANIT.

FIGURE 31: Relative Fission Rates in UO_2 -0.9 wt% PuO_2 (Nominally 25 Micron Diameter PuO_2) Calculated Using GRANIT.

and the thermal utilizations were increased by 2 and 6 percent for the unpoisoned and poisoned PuO_2 particulate fueled lattices, respectively, the calculated values for k_∞ would agree with experimental values for k_∞ within 1 percent for both poisoned and unpoisoned lattices. Similarly, if calculated values for $\bar{\eta}_2^{239}$ were reduced 2 percent and the thermal utilizations were increased by 2 and 3 percent for the unpoisoned and poisoned $\text{UO}_2 - 0.9 \text{ wt\% PuO}_2$ fueled lattices, respectively, the calculated values for k_∞ would also agree with experimental values for k_∞ in both poisoned and unpoisoned lattices.

Additional corroboration of the inadequacy of thermal neutron spectrum calculations for the fuel, using the Nelkin scattering kernel in THERMOS, was obtained from correlation of values for k_∞ in the lattice measured by the unpoisoned technique⁽³⁰⁾, using both $\text{UO}_2 - 2.0 \text{ wt\% PuO}_2$ and $\text{UO}_2 - 0.9 \text{ wt\% PuO}_2$ vibrationally compacted fuels, with values determined from the poisoned technique. The excess neutron production the the test lattice, $k_\infty - 1$, is determined from the ratio of reactivity perturbations of a test lattice sample and a normalizing neutron absorber, using the unpoisoned technique. A small sample of the test lattice (known as the central cell) is inserted into a void in the center of the reactor. Surrounding this void is a region of material similar to the central cell, and the neutron spectrum in this region is adjusted to be as close as possible to the fundamental mode condition.⁽³¹⁾ The reactivity perturbation caused by the insertion of the central cell is normalized to the perturbation produced when a sample of copper is placed in the void. It is also necessary to measure the rate of neutron absorption in the copper and central cell.

Results obtained from this method are sensitive to the calculated value for $\bar{\eta}_2^{239}$ used to evaluate thermal neutron absorption in ^{239}Pu from measured fission rates. Changes in calculated values for $\bar{\eta}_2^{239}$ result in

nearly inversely proportional changes in values for $(k_{\infty} - 1)$ inferred from experimental measurements. If values for $\bar{\eta}_2^{239}$ calculated using THERMOS are reduced by 4 and 2 percent for the $UO_2 - 2.0 \text{ wt\% } PuO_2$ and the $UO_2 - 0.9 \text{ wt\% } PuO_2$ fueled lattices, respectively, as suggested by the previous analysis, values for k_{∞} in the unpoisoned lattices are 1.341 ± 0.019 and 1.106 ± 0.008 respectively.⁽³²⁾ These values are in agreement with values determined from the poisoned technique, listed in Table VI, and provide further verification of the experimental methods and analysis.

The adjustment of the calculated neutron spectrum in the fuel required to give values for $\bar{\eta}_2^{239}$ in agreement with experiment, tends to reduce the 14 percent discrepancy between calculated and measured thermal disadvantage factors in the $UO_2 - PuO_2$ particulate fueled lattices to about 5 percent. Discrepancies of this magnitude are seen in correlations for water-moderated uranium fueled lattices.^(33,34,35) Thermal disadvantage factors for uranium fueled lattices measured by dysprosium activation show less discrepancy between calculation and experiment than ^{235}U fission disadvantage factor measurements.⁽³³⁾ The reduced sensitivity of ^{164}Dy to neutrons above 0.1 eV relative to ^{235}U , shown in Figure 32, could explain the results of Honeck's correlations⁽³³⁾ if the calculated flux dip in the small resonance of ^{235}U at 0.3 eV were overestimated by THERMOS calculations for the uranium fueled lattices. Thermal disadvantage factors for the $UO_2 - PuO_2$ fueled water-moderated lattices in the PCTR were measured by copper activation. The ^{63}Cu activation cross section has a $1/v$ shape in the range of thermal neutron energies, which is more sensitive to neutrons above 0.1 eV than either ^{164}Dy or ^{235}U , as shown in Figure 32.

Calculated values for the epithermal-to-thermal ^{238}U capture ratio, ρ^{28} , were larger than measured values (given in Table VIII) by about 12%, in both unpoisoned and poisoned $UO_2 - 2.0 \text{ wt\% } PuO_2$ (zero micron PuO_2 particulate)

FIGURE 32: Cross Sections for ^{235}U Fission and ^{164}Dy Activation.

fueled lattices. The uncertainty in the individual measurements is estimated at $\pm 4\%$. The magnitude of experimental uncertainties preclude a judgement on the accuracy of ^{238}U resonance parameter data used in EGGNIT and HRG calculations. The discrepancy between calculated and measured values for ρ^{28} is due primarily to the calculational overestimate of the thermal disadvantage factor, and the resulting underestimate of the thermal utilization, f .

The inadequacy of the Nelkin scattering kernel for thermal neutron spectrum calculations in plutonium fueled lattices can be seen qualitatively by a comparison of theoretical and experimental scattering laws for water. A comparison of Nelkin's water kernel with experiment, made by Lindenmeier,⁽³⁶⁾ has shown that the most striking discrepancies appear for small momentum transfer. These discrepancies could account for the overly depressed flux around the 0.3 eV resonance of ^{239}Pu , and at energies less than 0.02 eV, as indicated by the measured relative reaction rates for ^{235}U and ^{239}Pu . The recent scattering model of Gotoh and Takahashi⁽³⁷⁾ shows much better agreement with experimental scattering laws for water. The use of this model should improve thermal neutron spectra calculations for plutonium fueled lattices moderated by water.

Reactivity Effect of PuO_2 Particle Size

The ability to predict the effect that plutonium particle size has on the neutron multiplication of a reactor system is important for the economic utilization of plutonium. The reactivity effect of changing the PuO_2 particle size in a water moderated $\text{UO}_2 - \text{PuO}_2$ fueled lattice, is complex due to the interaction of changes in ^{239}Pu fission, ^{239}Pu capture, and ^{240}Pu absorption reaction rates. Self-shielding of ^{239}Pu with increasing PuO_2 particle size reduces the ^{239}Pu fission reaction rate in the fuel, but also reduces the ^{239}Pu capture-to-fission ratio, α^{-239} , due to a larger proportionate increase in shielding of the 0.3 eV resonance. Increased shielding of the 1.0 eV resonance

of ^{240}Pu with increasing PuO_2 particle size contributes a positive reactivity effect. The magnitude of this positive reactivity effect is dependent on the isotopic fraction of ^{240}Pu in the plutonium. Combinations of the positive and negative reactivity effects due to PuO_2 particle shielding can result in either increases or decreases in the net reactivity, depending on the neutron spectrum, the relative ^{240}Pu content of the plutonium, and the change in the PuO_2 particle size.

The change in k_∞ as a function of PuO_2 particle size was inferred for a standard UO_2 - 2.0 wt% PuO_2 (8% ^{240}Pu) fuel having a density of 8.66 gm/cm^3 from the experimental values for k_∞ listed in Table X. Corrections to k_∞ for density and enrichment differences between the actual fuel compositions, listed in Table III, and the base fuel composition (2.0 wt% PuO_2 , 8.66 gm/cm^3) were calculated using GRANIT-EGGNIT. The inferred values for k_∞ as a function of PuO_2 particle size for UO_2 - 2.0 wt% PuO_2 fuel are compared with calculated values in Table XII and in Figure 33. The root-mean-square errors for the ratio (calculation-to-experiment) are within limits that are expected from the assigned experimental errors. The calculated curve was normalized to experiment using the average value for the ratio (calculation-to-experiment) of 1.03, listed in Table XII. The results inferred from experiment show that k_∞ is increased 52 milli-k by changing from fuel containing 328 micron diameter PuO_2 particles to fuel containing zero micron PuO_2 particles (solid solution), in good agreement with the change predicted by GRANIT-EGGNIT calculations.

TABLE XII

INFERRED VALUES OF k_{∞} AS A FUNCTION OF PuO_2 PARTICLE SIZE IN UO_2 -2.0 wt% PuO_2 FUEL RODS

Particulate Fuel Set	PuO_2 Particles Diameter (microns)	Experimental k_{∞} (Actual Fuel Composition)	Calculated Correction to k_{∞} for Density and Enrichment Differences of Fuel	UO_2 -2.0 wt% PuO_2 Fuel Enrichment (8.66 gm/cm ³ Fuel Density)		Ratio $\left(\frac{\text{Calc.}}{\text{Exp.}}\right)$
				Experimental k_{∞}	Calculated k_{∞}	
8BMB	0	1.314 $\pm$ 0.005	+0.002	1.316 $\pm$ 0.005	1.348	1.024
8SC	0	1.322 $\pm$ 0.005	-0.007	1.315 $\pm$ 0.005	↓	1.025
8BME	0	1.321 $\pm$ 0.005	-0.008	1.313 $\pm$ 0.005		1.027
8P50	52	1.328 $\pm$ 0.005	-0.025	1.303 $\pm$ 0.006	1.339	1.028
8B100	107	1.326 $\pm$ 0.005	-0.040	1.286 $\pm$ 0.007	1.331	1.035
8P100	107	1.317 $\pm$ 0.005	-0.039	1.278 $\pm$ 0.006	↓	1.041
8LD100	107	1.304 $\pm$ 0.005	-0.030	1.274 $\pm$ 0.007		1.045
8B200	195	1.318 $\pm$ 0.005	-0.032	1.286 $\pm$ 0.006	1.317	1.024
8P200	195	1.314 $\pm$ 0.005	-0.036	1.278 $\pm$ 0.006	↓	1.031
8B350	328	1.299 $\pm$ 0.005	-0.038	1.261 $\pm$ 0.006	1.296	1.028
8RA350	328	1.310 $\pm$ 0.005	-0.045	1.265 $\pm$ 0.007	↓	1.025

Average Value = 1.030

Root-Mean-Square Deviation = $\pm$ 0.007

FIGURE 33. k_{∞} As A Function of PuO_2 Particle Diameter for UO_2 -2.0 wt% PuO_2 (8.66 gm/cm³) Fuel Rods in a 1.0 Inch Square Pitch Lattice Moderated with H_2O .

CONCLUSIONS

An extensive experiment theory correlation of integral lattice parameters from PCTR experiments with a 1.0-inch square pitch light water moderated lattice containing UO_2 - PuO_2 fuel rods has been made. Several conclusions can be drawn from these analyses concerning the adequacy of calculational techniques used in reactor design to predict neutronic characteristics of light water moderated UO_2 - PuO_2 fueled lattices.

First, a basic discrepancy is indicated in the space and energy dependent thermal neutron spectrum as calculated using the THERMOS method with the Nelkin scattering kernel for light water, which causes

- a calculational overestimate of the thermal disadvantage factor that increases with plutonium concentration in the fuel,
- a calculational underestimate of the thermal utilization that increases with boron concentration in the water moderator, and
- a calculational overestimate of the flux depression around the 0.3 eV resonance of ^{239}Pu that results in an overestimate of the value of $\bar{\eta}_2^{239}$ and an overestimate of the ^{235}U to ^{239}Pu fission ratio. An improved scattering model suggested by Gotoh and Takahashi may reduce these discrepancies.

Secondly, while calculational errors in $\bar{\eta}_2^{239}$ and f tend to cancel in lattices poisoned to k_∞ of unity with boron in water, the same calculational errors in $\bar{\eta}_2^{239}$ and f for the unpoisoned lattices only partially compensate. For the unpoisoned lattices, calculational overestimates of k_∞ due to non-compensating errors in $\bar{\eta}_2^{239}$ and f increase with plutonium concentration in the fuel.

Agreement between measured values for k_{∞} in the unpoisoned lattice determined from both the null-reactivity technique and the unpoisoned technique provides further verification of the experimental methods and analysis.

Fuel element development for plutonium recycle has led to fuels composed of the mixed oxides of plutonium and uranium. These fuel elements are presently fabricated by distributing the plutonium oxide throughout the uranium oxide as particles. Since a change in the particle size is probably either in the event of a reactor transient or during normal fuel irradiation and burn-up, knowledge of the effect that plutonium particle size has on the neutron multiplication of a reactor system is important for the economic utilization of plutonium.

This series of experiments and calculations represent the first conclusive analysis in the area of PuO_2 particulate effects on neutron multiplication in water moderated lattices. Fuel rods of $\text{UO}_2 - \text{PuO}_2$ were used to study the effect on k_{∞} of changing the PuO_2 particle diameter from 328 microns to less than one micron. These additional conclusions were drawn concerning the particle effects:

- The relative change in thermal fission rate in ^{239}Pu due to change in PuO_2 particle size, predicted by GRANIT calculations, is in agreement with measured relative fission rates.
- The change in k_{∞} due to change in PuO_2 particle size, predicted by GRANIT-EGGNIT calculations, is in agreement with measured values in both the unpoisoned lattice and the lattice poisoned to k_{∞} of unity with boron dissolved in the water moderator.

Therefore, the calculational models used in GRANIT and EGGNIT adequately predict the neutronic effects of PuO_2 particles in these water moderated lattices.

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