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THERMAL ACTIVATION METHOD FOR
 ρ_{28} MEASUREMENTS IN SLIGHTLY ENRICHED
URANIUM OXIDE LATTICES

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THERMAL ACTIVATION METHOD FOR
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URANIUM OXIDE LATTICES

May 1963

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ABSTRACT

This report describes the thermal activation method for measuring ρ_{28} in heterogeneous slightly enriched UO_2 lattices and gives results for 12 water-moderated lattices. The technique has three primary advantages over conventional U^{238} cadmium-ratio measurements: (1) uranium activities are measured by chemically processing the irradiated fuel to avoid problems associated with the use of U^{238} foils, (2) cadmium-covered uranium measurements are eliminated to avoid cadmium perturbations, and (3) ρ_{28} is obtained directly, rather than by subtracting one from the cadmium ratio, to avoid large errors when the cadmium ratio approaches unity. The technique compares total absorptions in U^{238} to thermal absorptions in an intermediate material, usually dysprosium, having a much higher probability of thermal absorptions. Then the thermal absorptions in dysprosium and uranium are normalized in a thermal irradiation. The technique is shown to be capable of measuring ρ_{28} with an accuracy of 1%, independent of the value of ρ_{28} and relatively free from systematic errors.

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1. INTRODUCTION

Resonance absorption in U^{238} plays a basic role in the physics of slightly enriched uranium lattices—particularly those lattices in which a substantial fraction of the fissions and absorptions are epithermal. The conventional index of resonance absorption in lattices is the cadmium ratio for the reaction:

where C_{28} is the ratio of total-to-epicadmium absorptions in U^{238} . The parameter required for theoretical comparisons is ρ_{28} , the ratio of epicadmium-to-subcadmium absorptions, derived from the measured quantity by

$$\rho_{28} = (C_{28} - 1)^{-1}. \quad (1-2)$$

Accurate measurements of ρ_{28} in lattices are important because they test the ability of the theoretical model to predict the correct neutron spectrum and check the accuracy of input cross sections and resonance parameters. Since ρ_{28} is closely related to the conversion ratio, these measurements also have a direct practical application to the design of reactor cores.

It has been apparent for some time that standard cadmium-ratio measurements are subject to large systematic errors, since serious discrepancies (5 to 10%) exist between measurements and predictions.¹ The most obvious source of experimental error lies in the use of U^{238} foils. The foils may not correctly sample the spatial distribution of U^{238} absorptions across the fuel rod, or the counters may not correctly integrate the foil activity because of problems of geometry, self-shielding and interfering activity. A smaller, but not negligible, potential source of error is the perturbation of the epicadmium flux by the cadmium in the cadmium-covered measurement. A third difficulty, particularly

serious in epithermal lattices, is that the error in ρ_{28} for a given error in C_{28} becomes very large as C_{28} approaches unity.

The Spectral Shift Control Reactor (SSCR) Basic Physics Program*, which was initiated at the Critical Experiment Laboratory in 1961, provided an opportunity to develop improved techniques for ρ_{28} measurements. The objective of this program was to study the nuclear properties of slightly enriched uranium-oxide lattices moderated by D_2O - H_2O mixtures. Since the neutron spectrum in many of these lattices was quite epithermal, accurate U^{238} resonance-absorption data were required for theoretical comparisons. Furthermore, a technique for the direct measurement of ρ_{28} (or $C_{28} - 1$) was needed, since C_{28} in the most epithermal lattice was only 1.06; i. e., a 1% error in C_{28} would introduce an error of 16% in ρ_{28} .

The thermal activation technique, developed during this program, is a method for the direct measurement of ρ_{28} (or $C_{28} - 1$). The technique also eliminates the other two problems that were described. First, the U^{238} absorptions are measured by dissolving the activated fuel rod itself and chemically removing fission products and other interfering activities. Second, the cadmium-covered uranium measurement is eliminated, so that the effect of cadmium perturbations is minimized.

This report describes the thermal activation technique for ρ_{28} measurements, including details on experimental procedures and the results of typical measurements in slightly enriched uranium-oxide lattices. It summarizes the work originally reported in Reference 2 and additional refinements and improvements given in Project Quarterly Technical Reports.³⁻⁶

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2. THEORETICAL BASIS

The thermal activation technique is a method for the direct measurement of ρ_{28} in slightly enriched uranium-oxide lattices. Although applicable to other lattices, it is particularly useful for epithermal lattices in which ρ_{28} is very large ($C_{28} \rightarrow 1$). Since the U^{238} thermal absorptions are so low in these lattices, epithermal absorptions in U^{238} are related to thermal absorptions in another material having a much higher probability of thermal absorptions than does U^{238} . Then, in a separate calibration experiment, the thermal absorptions in the two materials are compared in a well-thermalized flux, where the two thermal-absorption rates are more nearly comparable. The accuracy of the measurement is increased when the intermediate, reference material has a much larger cadmium ratio in the lattice than U^{238} . For this reason dysprosium is usually used, although a few comparisons were made with copper.

The ratio of thermal-to-epithermal absorptions in U^{238} in a fuel rod in a critical experiment lattice, $C_{28} - 1$, is

$$C_{28} - 1 = \left(\frac{A_U^{Th}}{A_X^{Th}} \right)_R \left(\frac{A_X^B}{A_U^C} \right)_R \left(\frac{C_X - 1}{C_X} \right)_R \quad (2-1)$$

where A is the saturated activity of the product isotope (U^{239} or Dy^{165}), the subscripts U and X refer to U^{238} and the intermediate material (usually Dy^{164}), the superscripts B , C , and Th refer to bare, cadmium-covered, and thermal (subcadmium) irradiations, C is the cadmium ratio, and R denotes an irradiation in the critical experiment lattice. A_U is measured by dissolving the irradiated fuel, removing fission products by solvent extraction, and counting the U^{239} activity. The Dy^{165} activity is obtained from a Dy - Al foil having the same diameter as the fuel, simultaneously irradiated in the fuel rod.

The ratio of thermal activities is

$$\left(\frac{A_U^{Th}}{A_X^{Th}} \right)_R = \left(\frac{N_U (\sigma \phi)_U \epsilon_U E_U}{N_X (\sigma \phi)_X \epsilon_X} \right)_R \quad (2-2)$$

where N is the total atoms of U^{238} in the fuel sample or total atoms of Dy^{164} in the Dy-Al foil, $(\sigma \phi)$ is the effective thermal activation cross section in the fuel, ϵ is the counting efficiency, and E is the efficiency of the chemical separation.

If another Dy-Al foil and another sample of fuel are irradiated together in any convenient neutron flux, denoted ϕ , a similar equation is

$$\left(\frac{A'_U{}^{Th}}{A'_X{}^{Th}} \right)_\phi = \left(\frac{N'_U (\sigma \phi)'_U \epsilon'_U E'_U}{N'_X (\sigma \phi)'_X \epsilon'_X} \right)_\phi \quad (2-3)$$

where the primes refer to the new samples. Both samples are counted the same way, so $\epsilon = \epsilon'$, but the efficiencies of the two chemical separations may differ.

Equations 2-2 and 2-3 combine to form the ratio of thermal activities,

$$\left(\frac{A_U^{Th}}{A_X^{Th}} \right)_R = \left(\frac{N_U E_U (\sigma \phi)_U}{N_X (\sigma \phi)_X} \right)_R \left(\frac{N'_X (\sigma \phi)'_X}{N'_U E'_U (\sigma \phi)'_U} \right)_\phi \left(\frac{A'_U{}^{Th}}{A'_X{}^{Th}} \right)_\phi \quad (2-4)$$

and Equation 2-1 becomes

$$C_{2R} - 1 = \left(\frac{A_X^B}{A_U^C} \right)_R \left(\frac{C_X - 1}{C_X} \right)_R \left(\frac{A'_U{}^{Th}}{A'_X{}^{Th}} \right)_\phi W_U W_X K \quad (2-5)$$

where

$$W_U = \frac{(N_U E_U)_R}{(N'_U E'_U)_\phi} \quad (2-6)$$

$$W_X = \frac{(N'_X)_\phi}{(N_X)_R} \quad (2-7)$$

$$K = \left(\frac{\sigma_U \phi_U}{\sigma_X \phi_X} \right)_R \left(\frac{\sigma'_X \phi'_X}{\sigma'_U \phi'_U} \right)_\phi \quad (2-8)$$

The factors W_U and W_X normalize the activities to the same weights of material after the chemical separations. To simplify the equations from this point, these factors are dropped and the saturated activities A and A' are redefined to signify weight-normalized activities.

The ratio of thermal activities in the ϕ -irradiation (Equation 2-5) can be replaced by the bare-activity ratio by measuring the cadmium ratios.

$$C_{28} - 1 = \left(\frac{A_X^B}{A_U^C} \right)_R \left(\frac{C_X - 1}{C_X} \right)_R \left(\frac{A_U^B}{A_X^B} \right)_\phi \left(\frac{C_U - 1}{C_U} \right)_\phi \left(\frac{C_X}{C_X - 1} \right)_\phi K \quad (2-9)$$

With the exception of K all the ratios in Equation 2-9 are measurable. K is a small correction to account for the slight non-1/ v behavior of Dy^{164} in the R - and ϕ -irradiations. When copper is used as the intermediate foil, $K = 1.000$. Details on the calculation of K are given in the appendix. It is pointed out that the error introduced by K is negligible in comparison with other errors, since K departs only slightly from unity and it operates on $C_{28} - 1$, rather than C_{28} .

The terms in Equation 2-9 corresponding to the ϕ -irradiation can be combined to form a calibration factor. Since they are independent of the critical experiment lattice properties, only one calibration factor measurement is needed if calibration samples are retained for weight normalization with samples irradiated in other critical experiment lattices. For least error the ϕ -irradiation should be performed in a well-thermalized flux; e. g., a thermal column, so that C_U and C_X are very large.

Equation 2-9 contains $(A_U^C)_R$, the uranium activity in a cadmium-covered irradiation in the critical experiment lattice. To avoid perturbations of the epicadmium uranium activity, as discussed in Section 4.2, a bare uranium activation can be substituted for the cadmium-covered one. Then,

$$\frac{C_{28} - 1}{C_{28}} \equiv \frac{1}{1 + \rho_{28}} \left(\frac{A_X^B}{A_U^B} \right)_R \left(\frac{C_X - 1}{C_X} \right)_R \Phi K \quad (2-10)$$

where the calibration factor Φ , combining all terms involving the ϕ -irradiation in Equation 2-9, is

$$\Phi \equiv \left(\frac{A_U^B}{A_X^B} \right)_{\phi} \left(\frac{C_U - 1}{C_U} \right)_{\phi} \left(\frac{C_X}{C_X - 1} \right)_{\phi} \quad (2-11)$$

In summary, the measurement of ρ_{28} or $C_{28} - 1$ depends primarily on the ratio of weight-normalized U^{239} and Dy^{165} activities obtained in a bare irradiation in the critical experiment lattice of interest and a once-measured calibration factor. Although a cadmium-covered lattice irradiation is required to obtain C_X , the cadmium ratio of the intermediate foil, the uranium activity need not be measured in the cadmium-covered run. Furthermore, the measurement of C_X need not be very accurate, since C_X is usually quite large and the error in $[(C_X - 1)/C_X]_R$ is correspondingly small.

3. EXPERIMENTAL PROCEDURES

3.1. Lattice Irradiation

The R-irradiations (Equation 2-10) were performed in the critical experiment lattices under study. Over a period of several years 12 slightly enriched uranium-oxide lattices varying in fuel enrichment, M/W ratio (nonmoderator-to-moderator volume ratio), and moderator composition (% D_2O in H_2O) were studied. Lattice details and typical results are discussed in Section 4.

The lattice geometry and loading arrangements are shown in Figures 3-1 through 3-3. The first experiments were performed with fuel rods of 4.02%-enriched UO_2 swaged in stainless steel tubes. The fuel (UO_2) diameter was 0.444 inch and the outer diameter was 0.475 inch. The wall thickness was approximately 0.016 inch and no gap existed between the fuel and clad. The last experiments were performed with fuel rods of 2.46%-enriched UO_2 pellets in aluminum tubes. The fuel (pellet) diameter was 0.406 inch, the outer diameter was 0.475 inch, and the wall thickness was 0.032 inch, so a 0.0025-inch radial gap existed between the pellets and the inside of the clad. Additional details are given in Section 4.4.

Irradiation samples of the swaged 4.02% fuel were prepared by cutting standard fuel rods into a number of short sections, called fuel slugs. The faces were accurately machined and sprayed lightly with Krylon to prevent spread of contamination. As shown in Figure 3-2, the data slugs usually were 1/2-inch long with 1/4-inch-long guard slugs at each end to facilitate alignment. Only the data slugs were processed and counted after irradiation. In the first experiment (Core III) the data slugs were 2-1/2-inches long and no guard slugs were used, and in Cores I and IX the data and guard slugs were 1-inch and 1/2-inch long, respectively. The fuel slugs were accurately positioned in the lattice between cut and sealed ends of a standard fuel rod, and the joints

were wrapped with two layers of 0.002-inch-thick Mylar tape to preserve the alignment and keep the joints watertight.

The loading arrangement for the pelletized 2.46% fuel is shown in Figure 3-3. In this case the data slugs were standard 3/4-inch-long UO_2 pellets, with intermediate foils on each side, loaded in the center of a standard fuel rod in the lattice.

In most experiments the intermediate foils were 0.007-inch-thick Dy-Al foils (≈ 4 wt % Dy) having the same diameter as the fuel; i. e., 0.444 inch for the 4.02% fuel and 0.406 inch for the 2.46% fuel. Copper was also used as the intermediate material to check the validity of the non- $1/v$ correction for dysprosium. These foils were 0.004-inch-thick metallic copper having the same diameter as the fuel. The intermediate foils were protected from fission-recoil contamination by covering them on each side with 0.0005-inch-thick aluminum foil.

The geometry for the cadmium-covered irradiation is also shown in Figures 3-2 and 3-3. The fuel slug and intermediate foils were completely enclosed by a 0.020-inch-wall cadmium sleeve and covered at the ends with 0.020-inch-thick cadmium discs having the same diameter as the ID of the sleeve. In some experiments cadmium perturbations were studied by shortening the length of the fuel slugs and cadmium sleeves. Since the cadmium-covered uranium activity is not needed for ρ_{28} , the data slugs were not always processed and counted. The cadmium-covered irradiation was, however, required for C_X .

Bare and cadmium-covered runs were normalized by at least four gold monitor foils positioned in the core about 8 inches away from the fuel slug. The irradiations were performed at power levels between 500 and 1000 watts for a period of 20 minutes with all control blades withdrawn to avoid flux perturbations.

3.2. Chemical Processing

Two methods of removing fission products and other interfering activities from the irradiated UO_2 fuel were investigated—sodium-acetate precipitation⁷ and solvent extraction⁸. Both methods were found to be rapid enough to permit U^{239} counting and efficient enough to yield the correct 23.5-minute half-life for at least 8 half-lives. The solvent-extraction method was used exclusively in this work.

After irradiation, the fuel slug was placed in a 500-ml beaker, and the UO_2 was dissolved in 50 to 100 ml of concentrated nitric acid warmed to accelerate the reaction. For the 4.02%-enriched swaged fuel the UO_2 was completely dissolved from the stainless steel cladding in about 10 minutes with no attack of the cladding. The sintered 2.46%-enriched pellets had to be crushed first, and dissolution was complete in about 15 minutes.

Approximately 0.050 ml of the uranyl-nitrate solution (20 to 50 mg UO_2) was pipetted into a 125-ml separatory funnel containing about 10 ml of tributyl phosphate (TBP) and 10 ml of 6 N HNO_3 . After thorough mixing and settling, the aqueous phase was withdrawn and discarded, and the organic phase was washed twice with 10-ml portions of 6 N HNO_3 .

Ammonium diuranate was then precipitated by adding 10 ml of 8 M NH_4OH to the separatory funnel. After shaking and partial settling, 2 to 4 ml of ethanol were added to reduce the surface tension between the aqueous and organic phases. By stirring the interface carefully with the tip of a glass stirring rod, nearly all of the precipitate was transferred to the lower (aqueous) phase. The upper (organic) phase was then withdrawn by a vacuum aspirator.

The precipitate was redissolved by adding 1 ml of concentrated nitric acid, and the solvent extraction was repeated with 10 ml of fresh TBP. After the final ammonium diuranate precipitation, about 15 ml of the aqueous phase containing most of the precipitate were withdrawn into an Atomlab-Ekstein stainless steel centrifuge cone fitted and O-ring sealed to a 1-1/4-inch-diameter superpure aluminum (99.999%) counting planchet. Figure 3-4 shows the centrifuge cones and counting planchets. It was found that the thickness of the deposit varied somewhat from the center to the edge of the planchet, so the planchet was masked with filter paper having a 1/2-inch-diameter hole in the center.

After adding a few drops of glycerin to retard flaking, the material was centrifuged for about 5 minutes at 3100 rpm in an International Equipment Company clinical centrifuge. The filter-paper mask was then removed, the 1/2-inch-diameter deposit was carefully washed with ethanol, and traces of precipitate under the mask were removed with a Q-tip. The deposit was dried for 3 to 6 minutes under an infrared-heat lamp and covered with a layer of Scotch tape ($\approx 10 \text{ mg/cm}^2$) to absorb alpha particles and protect the sample.

The entire process, from reactor shutdown to beginning of counting, could be completed in about 60 minutes. At least two separate samples of the dissolved irradiated UO_2 from each reactor run were simultaneously processed to improve the precision of the data. Figure 3-5 shows the equipment and laboratory space required for the solvent extraction process.

3.3. Counting

The U^{239} activity of the uranium samples deposited on the counting planchets after chemical processing was beta counted in end-window gas-flow (methane) proportional counters (see Figure 3-6). Each planchet was counted at least 5 times for one minute in each of three counters. Count rates varied from about 100,000 counts/min to a minimum of about 10,000 counts/min after about 90 minutes of counting time. When different shelves were used for counting the activity, appropriate shelf ratios were measured and applied.

The background was investigated by following the decay curves of a few samples over a week period. The 23.5-minute U^{239} activity was predominant for the first 100 minutes. After about 300 minutes the count rate followed the 2.33-day Np^{239} decay quite accurately. During the counting period of interest, 60 to 150 minutes after shutdown, the decay curve could be resolved into the major U^{239} component and a very small Np^{239} component, using the Np^{239} component obtained from counting after 300 minutes. Therefore, the data were routinely corrected for background by counting each planchet for 10 minutes after a 900-minute decay and extrapolating the Np^{239} activity back to the counting period. The background correction was usually less than 0.5% and never greater than about 1%.

Figure 3-7 shows typical decay curves obtained by beta counting the planchets after chemical processing and correcting for background as described above. The correct U^{239} half-life was obtained over a period as long as eight half-lives, within the accuracy of the measurements.

The Dy-Al and copper intermediate foils were beta counted the same way using three end-window gas-flow proportional counters. The foils were carefully washed before counting. Count rates were at least as high as the uranium samples, except for the cadmium-covered Dy-Al

foils, where count rates were between 5000 and 10,000 counts/min. Each intermediate foil was counted at least once on each side for one minute in each of the three counters.

3.4. Weight Normalization

Since the chemical separations were not quantitative, the relative uranium content of the uranium samples after processing had to be normalized. The usual procedure was to irradiate sets of samples mounted on a wheel, which rotated in a relatively constant and fairly thermal neutron flux, and to beta count the resulting fission-product activity.

The wheel assembly, shown in Figure 3-8, is a 7-3/4-inch-diameter graphite cylinder, recessed for a 5-1/2-inch-diameter lucite plate and provided with a shaft and 60-rpm geared-drive motor. The lucite plate has machined depressions of suitable size to hold the samples and is covered by a thin lucite disc. Irradiations were usually done with the wheel assembly mounted in the 8-inch-diameter beam port of the Lynchburg Pool Reactor (LPR), as illustrated in Figure 3-9. Additional thermalization was achieved by interposing a 2-foot-cube graphite thermal column between the LPR core and the wheel assembly. In this position the samples were usually irradiated for about 10 minutes with the LPR operating at 5 KW.

After irradiation the fission-product activities on the uranium planchets were beta counted in the same equipment and geometry as the original U^{239} count. To avoid interference from the activation of manganese in commercially available aluminum planchets, superpure (99.999%) aluminum counting planchets were procured and used in all experiments but the first (Core III). Counting began about 40 minutes after shutdown to permit the 9.5-minute- Mg^{27} and 2.3-minute- Al^{28} activities to decay. The uranium samples were counted about five times in sequence, and their activities were measured relative to a simultaneously irradiated "slave" or reference uranium sample. The weight normalizations were obtained by averaging the ratios of sample to "slave" activity at corresponding decay times. A small correction ($< 1/2\%$) was applied for the natural uranium background by counting each sample prior to irradiation. Supplementary experiments showed that the Scotch tape covering the planchets did not affect the weight normalization.

The Dy-Al intermediate foils were weight normalized using essentially the same procedure; i. e., wheel irradiation followed by beta counting in the same equipment and geometry used in the original count. The average saturated activities computed from at least four separate counts in each of the three counters were used as weight-normalizing factors. The copper intermediate foils were sufficiently uniform to require no weight normalization.

The final step in the weight normalization required an intercomparison, on an arbitrary but consistent basis, between the weights of the uranium samples and the Dy-Al (or Cu) intermediate foils. This was accomplished by using the calibrations described in the next section. However, to avoid recalibrating for each experiment, one or more of the uranium and Dy-Al (or Cu) calibration standards were wheel irradiated with several of the uranium and Dy-Al samples from the critical experiment lattice irradiations. The activity ratios then related the weights of the uranium samples to the uranium standard and the weights of the Dy-Al samples to the Dy-Al standard.

The validity of this procedure for weight normalization was investigated in some detail. The first requirement is that all of the atoms in the uranium and Dy-Al (or Cu) samples be exposed to the same integrated flux. This condition was achieved by making the samples sufficiently thin (≈ 1 mg U/cm² and ≈ 2 mg Dy/cm²) to minimize thermal self-shielding and flux-depression effects and by using a fairly thermal neutron flux. A second requirement is that the counting procedure is identical to that used in connection with the original irradiation in the critical experiment lattice, to avoid errors due to sample self-shielding and variations in sample uniformity and thickness. This was done with the Dy-Al foils, but the uranium samples were handled somewhat differently. Both activities were beta counted. After the original irradiation and chemical processing, the U²³⁹ activity of the uranium samples was counted, but the fission-product activity was counted after the weight-normalization irradiation. The equivalence of the two procedures was shown in the following ways:

1. After a one-week decay the natural-uranium activity in the 180-kev peak was gamma counted in a 400-channel scintillation spectrometer. The integral counts from a number of different uranium samples were found to be closely proportional to the fission-product beta counts after the samples were reirradiated.

2. Several samples were fired for several hours in an oven at 900 C. The U_3O_8 residues were carefully weighed and agreed with the fission-product beta-count method of weight normalization to within 0.6%.

3. In one series of experiments the uranium samples were both beta and gamma counted after the critical experiment irradiation and again after the weight-normalization irradiation. Although the weight-normalization factors disagreed by as much as 10% from sample to sample, the weight-normalized activities agreed to within 1% when beta and gamma countings were compared.

We conclude that the procedure used for weight normalization is correct to approximately 1%. The comparisons also show that either beta or gamma counting is satisfactory, if the same counting procedure for the original and for the weight-normalization irradiations is used consistently.

3.5. Calibration Factor

The ratios in Equation 2-10 that are obtained in a separate ϕ -irradiation are independent of the critical experiment lattice and may be lumped together as a calibration factor,

$$\Phi = \left(\frac{A_U^B}{A_X^B} \right)_{\phi} \left(\frac{C_U^{-1}}{C_U} \right)_{\phi} \left(\frac{C_X}{C_X^{-1}} \right)_{\phi} \quad (3-1)$$

For highest accuracy the calibration factor should be measured in an irradiation facility having a well thermalized flux, so that the ratios of cadmium ratios in Equation 2-11 approach unity. The flux level should also be high enough to produce useful activity in very thin uranium samples.

Calibration factors were measured several times during the course of the program to take advantage of improved techniques. The most recent values were obtained in a series of experiments in the thermal column of the University of Virginia (U-Va) Research Reactor. These results agree very well with earlier measurements in the beam port of the LPR.

The standard procedure for all calibrations involved the following steps. Uranium samples were prepared by dissolving small portions

of the fuel (4.02%-enriched slugs or 2.46%-enriched pellets) in hot, concentrated nitric acid. The uranyl-nitrate solution was pipetted onto 0.5-inch-diameter aluminum and cadmium planchets and dried under an infrared-heat lamp. The deposit thickness was about 14 mg UO_2/cm^2 in all the U-Va samples and ranged from about 10 to 160 mg UO_2/cm^2 in the LPR samples. As will be shown later, self-shielding and flux-depression effects were negligible for sample thicknesses less than about 200 mg UO_2/cm^2 . The intermediate foils for the 2.46%-enriched fuel were 0.007-inch-thick foils of Dy-Al alloy (≈ 4 Wt % Dy) and 0.004-inch foils of copper metal, both 0.406 inch in diameter. For the 4.02%-enriched fuel only 0.444-inch-diameter by 0.007-inch-thick Dy-Al foils were used.

The uranium and intermediate-foil samples were irradiated in sets on the wheel described in Section 3.4. In the U-Va work the wheel was mounted in a graphite stringer in the thermal column, and the reactor was operated for 20 minutes at 1000 KW. In the LPR calibration the wheel was mounted as shown in Figure 3-9, and the reactor was operated for 10 minutes at 2 to 5 KW.

After irradiation the intermediate foils were beta counted and the uranium samples were dissolved, chemically processed, and U^{239} beta counted according to the procedures described in Sections 3.2 and 3.3. The counting equipment used in Lynchburg was taken to the University of Virginia so that the U-Va calibrations could be performed under comparable conditions. Counting began about 45 minutes after reactor shutdown. Total counts accumulated from each sample in each run exceeded 100,000 except for the cadmium-covered runs, in which the total counts were about 10,000 for uranium, 5,000 for dysprosium, and 500 for copper. Bare and cadmium-covered runs were normalized by four Dy-Al monitor foils mounted on the stationary structure of the wheel assembly. All samples were weight normalized by reactivation and beta counting as described in Section 3.4.

In the series of LPR calibrations the effects of flux depression and self-shielding due to sample thickness were investigated by irradiating bare and cadmium-covered 4.02%-enriched UO_2 samples varying in thickness from about 10 to 160 mg UO_2/cm^2 . Although both A_U^B and C_U varied with sample thickness, as shown in Figure 3-10, those terms

in the calibration factor (Equation 2-11) that depend on uranium reduce to the thermal activity alone; i. e.,

$$\Phi \propto A_U^B \left(\frac{C_U^{-1}}{C_U} \right) = \left(A_U^B - A_U^C \right) \quad (3-2)$$

and the thermal activity was found to be essentially independent of sample thickness. These results are consistent with a calculated thermal self-shielding and flux depression for 160 mg UO_2/cm^2 of less than 0.5% ($\Sigma_a t/2 \approx 0.005$). The same type of calculation for the Dy-Al samples (2 mg Dy/ cm^2) gives a thermal self-shielding and flux depression less than 0.5% ($\Sigma_a t/2 = 0.004$).

The calibration factor for Cores I and IX was measured quite early in the program, with somewhat lesser precision than the later U-Va and LPR calibrations. The techniques were essentially the same, except that the factor $(C_U^{-1})/C_U$ was measured in separate runs with a depleted uranium foil. The first calibration factor (for Core III) was obtained before techniques were standardized and is subject to an appreciably larger uncertainty.

The calibration factors are summarized in Table 3-1, where the core numbers refer to the applicable critical experiment lattice. Only the first two columns are internormalized, but the close agreement between these two calibration factors supports the validity of the standard technique used in all cases. The unweighted average of the LPR and U-Va calibrations is used for Cores IV, X, XI and XII.

The tabulated results are averages of 12 runs with 12 samples of each type (LPR), 6 runs with 12 samples of each type (combined U-Va), and 3 runs with 6 samples of each type (LSR). Although C_U was appreciably lower, the LPR calibration factor has comparable accuracy because of the larger number of runs. As pointed out before, the UO_2 -sample thickness varied in the LPR runs. Although C_U and A_U^B/A_{Dy}^B changed with sample thickness, Φ_{Dy} remained constant. The values of C_U and A_U^B/A_{Dy}^B are arbitrarily listed for a sample thickness of about 40 mg/ cm^2 .

Table 3-1. Calibration Factors

Source Fuel enrichment Intermediate foil Core numbers	LPR 4.02% Dy-Al IV, X, XI, XII	U-Va 4.02% Dy-Al/Cu IV, X, XI, XII	U-Va 2.46% Dy-Al/Cu XIII thru XVII	LSR 2.46% Dy-Al I, IX
C_U	3.72*	51.72 ± 6.20	45.68 ± 4.27	65.2 ± 1.8
C_{Dy}	601 ± 3	2038 ± 14	2259 ± 201	> 1000
C_{Cu}	--	1591 ± 131	1488 ± 67	--
$(C_U^{-1})/C_U$	0.731*	0.9807 ± 0.0023	0.9781 ± 0.0020	0.985 ± 0.001
$C_{Dy}/(C_{Dy}^{-1})$	1.002 ± 0.001	1.0005 ± 0.0000	1.0004 ± 0.0000	1.001 ± 0.001
$C_{Cu}/(C_{Cu}^{-1})$	--	1.0006 ± 0.0001	1.0007 ± 0.0000	--
$A_U^B/A_{Dy}^B \times 10^{-2}$	5.248*	3.867 ± 0.039	2.826 ± 0.010	1.86 ± 0.10
$A_U^B/A_{Cu}^B \times 10^{-2}$	--	1.681 ± 0.020	1.287 ± 0.005	--
$\Phi_{Dy} \times 10^{-2}$	3.870 ± 0.066	3.793 ± 0.039	2.765 ± 0.010	1.83 ± 0.10
$\Phi_{Cu} \times 10^{-2}$	--	1.649 ± 0.020	1.260 ± 0.005	--

* Depends on sample thickness (see text)

For Core III use $(2.2 \pm 0.1) \times 10^{-3}$

For Cores IV, X, XI, XII use average of LPR and U-Va, $(3.83 \pm 0.04) \times 10^{-2}$

Figure 3-1. Critical Experiment Lattice Geometry

M/W Ratio	Pitch, in.
1.006	0.595
1.195	0.571

M/W Ratio	Pitch, in.
1.001	0.595
0.651	0.670

Figure 3-2. Loading Arrangement (4.02%-Enriched Fuel)

Bare Irradiation

Cadmium-Covered Irradiation

Figure 3-3. Loading Arrangement (2.46%-Enriched Fuel)

Figure 3-4. Centrifuge Cones and Counting Planchets

Figure 3-5. Chemical Separation Equipment

Figure 3-6. Counting Equipment

Figure 3-7. U^{239} Decay Following Chemical Separation

Figure 3-8. Wheel Irradiation Equipment

Figure 3-9. LPR Thermal Irradiation Facility

Figure 3-10. Uranium Activity Vs Sample Thickness in LPR Calibrations

4. RESULTS AND CONCLUSIONS

4.1. Non-1/v Correction

The use of dysprosium as the intermediate-foil material requires a small non-1/v correction K in the final equation (2-10) for $C_{28}-1$ and ρ_{28} . The validity of the calculation given in the appendix was checked in two lattices, by repeating the measurements with copper as the intermediate-foil material. Although copper is less advantageous than dysprosium because its cadmium ratio is much lower, it does have a 1/v cross-section dependence, so K_{Cu} is unity.

The results of the measurements are summarized in Table 4-1. The difference between $C_{28}-1$ derived from the dysprosium data using the calculated K and the copper data using $K = 1$ is small and consistent with experimental errors. The difference also changes sign, suggesting the K is also relatively free from systematic bias. Therefore, the quoted error in K , given as 20% of $K-1$, appears reasonable and makes only a small contribution to the total error in $C_{28}-1$.

Table 4-1. Comparison of Dy and Cu as Intermediate Foils

	Core XIV	Core XV
$(A_{Dy}^B / A_U^B)_R$	5.399 ± 0.023	7.426 ± 0.029
$(A_{Cu}^B / A_U^B)_R$	20.68 ± 0.15	24.54 ± 0.14
$\left[(C_{Dy}^{-1}) / C_{Dy} \right]_R$	0.9130 ± 0.0003	0.9367 ± 0.0013
$\left[(C_{Cu}^{-1}) / C_{Cu} \right]_R$	0.5375 ± 0.0066	0.6586 ± 0.0031
$\Phi_{Dy}, \times 10^{-2}$	2.765 ± 0.011	2.765 ± 0.011
$\Phi_{Cu}, \times 10^{-2}$	1.260 ± 0.005	1.260 ± 0.005
K_{Dy}	1.043 ± 0.009	1.031 ± 0.006
K_{Cu}	1.000	1.000
$(C_{28}^{-1})_{Dy}$	0.166 ± 0.002	0.247 ± 0.003
$(C_{28}^{-1})_{Cu}$	0.163 ± 0.003	0.255 ± 0.003
Difference	+ 0.003	- 0.008

4.2. Cadmium Perturbation

Although the measurement of C_{28} by the thermal activation technique does not require a cadmium-covered uranium irradiation in the critical experiment lattice (see Equation 2-10), the cadmium-covered uranium activity $(A_U^C)_R$ was measured in some of the lattices. These data can be analyzed to show the perturbation by the cadmium sleeve in standard cadmium-ratio measurements.

A reduction of the epithermal uranium activation in the fuel rod can be expected from the following effects:

1. The cadmium sleeve displaces moderator around the fuel rod (see Figure 3-2), thus increasing the Dancoff shielding of U^{238} resonances.
2. The cadmium sleeve depresses the source of fast neutrons in the immediate neighborhood of the fuel rod, thus decreasing the epithermal U^{238} activation.

The true cadmium ratio, C_{28} , is defined as

$$C_{28} = A_U^B / A_U^C \quad (4-1)$$

where A_U^B and A_U^C are the unperturbed bare and cadmium-covered uranium saturated activities. If the cadmium sleeve in the cadmium-covered measurements perturbs A_U^C so that

$$F = A_U^C / A_U^C \quad (4-2)$$

where A_U^C is the perturbed cadmium-covered activity, then the apparent cadmium ratio is

$$C_a = A_U^B / A_U^C = 1/F \times C_{28} \quad (4-3)$$

The difference between C_a obtained from Equation 4-3 and C_{28} derived by the thermal activation method (Equation 2-10) can be attributed to the perturbation by the cadmium sleeve in the C_a measurement. This hypothesis was tested in one of the lattices (Core X) by measuring both C_a and C_{28} with cadmium sleeves of varying lengths.

The geometry of Figure 3-2 was used, but the lengths of the fuel and guard slugs were reduced proportionally. The results are listed in Table 4-2.

Table 4-2. Perturbation by Cadmium Sleeve in Core X

Length of cadmium, in.	C_a (Equation 4-3)	C_{28} (Equation 2-10)	F
2-3/4	1.275 ± 0.019	1.194 ± 0.004	0.937 ± 0.017
1-3/4	1.247 ± 0.019	1.196 ± 0.004	0.959 ± 0.017
5/8	1.228 ± 0.024	1.195 ± 0.004	0.973 ± 0.023

These data also show the improvement in precision possible by the direct measurement of C_{28} -1.

As seen in Figure 4-1, measurements could not be made with a total weight of cadmium less than 4 gm (5/8 in. sleeve), because of the constant contribution of the cadmium discs at the ends of the fuel slugs. However, C_a curve can be forced to extrapolate to the C_{28} value at zero cadmium weight (and F to unity) either by assuming a nonlinear relation or by drawing another straight line through the error bars. Furthermore, the values need not exactly converge, since the Dancoff shielding effect was present in all cases; i. e., the length of cadmium sleeve was changed but its OD was not. Similar measurements with the 2.46%-enriched fuel, in the geometry of Figure 3-3 with no moderator displacement, suggest an increase in C_a of about 1% due to moderator displacement alone.

The hypothesis that the difference between C_a and C_{28} is due to the cadmium perturbation of the epithermal uranium activity is supported by the following observations.

1. C_a decreases as the quantity of cadmium is reduced.
2. C_{28} is constant with the quantity of cadmium. Therefore, the difference between C_{28} and C_a must be due to a parameter affected by cadmium. The only parameter present in the C_a measurement and absent in the C_{28} measurement is the cadmium-covered uranium activity.

3. When the effect of Dancoff shielding is taken into consideration, C_a tends to converge to C_{28} as the quantity of cadmium goes to zero.

4. The departure of F from unity is difficult to measure accurately because it depends on C_a , which can only be measured to 1-2%. However, the following general conclusions that support the hypothesis can be drawn from F measurements in a number of lattices. First, F was less than or equal to one in all lattices. Second, F was smallest when the moderator was light water and increased toward unity as D_2O was added to the moderator; i. e., the lattice became more epithermal.

It is concluded that cadmium perturbations can cause appreciable errors in conventional cadmium-ratio measurements, even if fairly small quantities of cadmium are used, particularly in light-water-moderated lattices. The thermal activation technique is free from this effect, since it does not require a cadmium-covered uranium measurement.

4.3. Effect of Cadmium on C_{Dy}

The thermal activation technique does not require a cadmium-covered uranium activation, but the cadmium-covered dysprosium (or copper) intermediate-foil activity is needed. The effect of cadmium perturbations on the intermediate-foil activity is minimized by using a material like dysprosium that is a strong thermal absorber. Then its cadmium ratio, C_D , in the lattice is very large and the error in $(C_D - 1)/C_D$ of Equation 2-10 is small.

Cadmium perturbations of the cadmium-covered dysprosium activity were considered during the experiments on Core X described in the preceding section. C_D decreased slightly as the length of the cadmium sleeve was reduced, but the maximum change in $(C_D - 1)/C_D$ was only 1/3% because C_D was so large. Since this term affects $C_{28} - 1$ rather than C_{28} , the influence of the perturbation may be considered negligible.

Another possible source of error in C_D involves neutron thermalization. The standard irradiation geometries of Figures 3-2 and 3-3 show that epithermal neutrons can be thermalized by scattering

in the UO_2 between the cadmium covers and the Dy-Al intermediate foil. This effect, which increases the cadmium-covered activity and decreases the cadmium ratio, has been observed in U^{235} cadmium-ratio measurements in comparable geometries.⁴

The importance of this error was studied by repeating cadmium-covered dysprosium measurements in many of the lattices with the Dy-Al intermediate foil completely enclosed by cadmium; i. e., no UO_2 between the cadmium and the Dy-Al foil. In all cases C_D was slightly larger, as expected, since this type of measurement should be entirely free from thermalization. However, the change in $(C_D - 1)/C_D$ was again very small—between 0.5 and 1.5% for all lattices.

4.4. Typical Results

Typical results of C_{28-1} and ρ_{28} measurements by the thermal activation technique on 12 water-moderated, slightly enriched UO_2 lattices are summarized in Table 4-3. The lattice properties for each core are described in Section 3.1. Additional details on the two types of fuel rods are given in Table 4-4.

Table 4-3. Summary of Results

Core No.	I	III	IV	IX	X	XI	XII	XIII	XIV	XV	XVI	XVIII
Enrichment, %	4.02	4.02	4.02	4.02	4.02	4.02	4.02	2.46	2.46	2.46	2.46	2.46
M/W	1.006	1.006	1.006	1.006	1.195	1.195	1.195	1.001	1.001	1.001	0.651	0.651
Mole % D ₂ O	0.0	76.5	69.7	49.7	0.0	70.1	49.7	0.0	70.0	49.8	85.5	70.0
(A ^B _{Dy} /A ^B _U) _R	10.98	30.1	2.323	6.330	4.470	1.810	2.589	11.17	5.399	7.426	5.162	7.490
	±0.29	±0.2	±0.085	±0.303	±0.061	±0.011	±0.052	±0.09	±0.023	±0.029	±0.021	±0.0±1
(C _{Dy}) _R	16.16	6.34	6.560	9.128	12.81	5.773	8.130	25.28	11.49	15.80	11.34	16.28
	±0.57	±0.02	±0.120	±0.250	±0.27	±0.072	±0.019	±1.17	±0.04	±0.30	±0.11	±1.60
$[(C_{Dy}^{-1})/C_{Dy}]_R$	0.938	0.842	0.8476	0.8904	0.9219	0.8268	0.8770	0.9604	0.9130	0.9367	0.9119	0.9386
	±0.002	±0.001	±0.0028	±0.0030	±0.0017	±0.0030	±0.0001	±0.0018	±0.0003	±0.0013	±0.0008	±0.0061
$\phi_{Dy} \times 10^{-2}$	1.83	0.22	3.83	1.83	3.83	3.83	3.83	2.765	2.765	2.765	2.765	2.765
	±0.10	±0.01	±0.04	±0.10	±0.04	±0.04	±0.04	±0.011	±0.011	±0.011	±0.011	±0.011
K	1.028	1.028	1.070	1.048	1.033	1.080	1.056	1.018	1.043	1.031	1.043	1.029
	±0.006	±0.016	±0.013	±0.010	±0.007	±0.016	±0.011	±0.004	±0.009	±0.006	±0.009	±0.006
C ₂₈ -1	0.243	0.065	0.089	0.123	0.197	0.067	0.102	0.439	0.168	0.250	0.159	0.253
	±0.018	±0.003	±0.004	±0.009	±0.004	±0.001	±0.003	±0.006	±0.002	±0.003	±0.002	±0.004
p ₂₈	4.12	15.4	11.2	8.13	5.08	14.9	9.80	2.28	5.95	4.00	6.29	3.95
	±0.31	±0.7	±0.5	±0.06	±0.10	±0.2	±0.29	±0.03	±0.07	±0.05	±0.08	±0.06

Table 4-4. Properties of Slightly Enriched UO_2 Fuel Rods

Parameter	4.02%-Enriched UO_2 ²	2.46%-Enriched UO_2 ⁵
Outer diameter, in.	0.4755 ± 0.0015	0.4748 ± 0.0006
Wall thickness, in.	0.0160 ± 0.0005	0.032 ± 0.001
Wall material	#304 stainless steel	#6061 aluminum
Fuel (pellet) diam, in.	0.444 ± 0.002	0.4054 ± 0.0005
Total length, in.	≈ 71.5	61.6 ± 0.2
Active (fuel) length, in.	66.7 ± 0.3	60.4 ± 0.3
Weight of UO_2 , gm/rod	1600 ± 2	1306 ± 1
Weight of U, gm/rod	1408 ± 2	1151 ± 1
Weight of U^{235} , gm/rod	56.61 ± 0.10	28.42 ± 0.02
Enrichment, wt U^{235} /wt U, %	4.020 ± 0.005	2.459 ± 0.001
Fuel density, gm/cm ³	9.46 ± 0.10	10.29 ± 0.05

In analyzing the data, the calculations of the non- $1/v$ correction K from the appendix were used, and the calibration factors Φ_{Dy} were taken from Table 3-1. The final values of $C_{28}-1$ were increased by $(1.0 \pm 0.5)\%$ to take into account neutron thermalization, as discussed previously although the correction is within existing errors. The errors are based on the consistency of repeated runs, unweighted and propagated by standard methods. In each core at least three bare runs (six uranium samples) and at least one cadmium-covered run were made, except in the first few cores where fewer runs were made.

As shown in Figure 4-2, the results appear to be internally consistent. The errors in the best measurements on the latest cores (XIII through XVII) are 1 to 2% in $C_{28}-1$ or ρ_{28} . The accuracy of the first few measurements did not reach this level because techniques were still being developed, particularly in the calibrations. Therefore, the results for 4.02% enrichment and $M/W = 1.0$, which are included for completeness, should not be weighted too strongly.

4.5. Conclusions

The results show that the thermal activation technique is capable of a precision within about 1% for $C_{28}-1$ or ρ_{28} , if measurements are made carefully and repeated several times. This is a substantial improvement over conventional cadmium-ratio measurements, where a precision within 1% for C_{28} implies an error of as much as 10 to 20% in $C_{28}-1$ and ρ_{28} when C_{28} is close to unity.

The thermal activation technique also appears to be relatively free from systematic error, at least within the precision of the measurements. The two primary sources of systematic error in conventional measurements are eliminated by:

1. Dissolving the fuel and measuring the integrated uranium activity, to avoid all the problems inherent in the use of depleted uranium foils, and
2. Avoiding cadmium perturbations in the uranium activity by eliminating cadmium-covered uranium measurements. By using a strong thermal absorber such as dysprosium for the intermediate foil, perturbations in the cadmium-covered dysprosium

measurement are made negligible.

Other potential sources of systematic error have been investigated. The correction for non- $1/v$ behavior is relatively simple and was confirmed by comparative measurements with copper (Section 4.1). The calibrations were repeated several times under different conditions; and the results agreed where comparisons were possible (Section 3.5). The important weight-normalization step was checked by both beta and gamma counting and by weighing, as discussed in Section 3.4. Finally, the procedure for chemical processing and U^{239} beta counting was periodically tested by measuring decay curves (Sections 3.2 and 3.3).

Theoretical comparisons of the most recent data have not been completed. However, a preliminary analysis of the first set of data, given in Reference 11, appears satisfactory.

Figure 4-1. C_{28} Vs Length of Cadmium Sleeve

Figure 4-2. $C_{28} - 1$ Vs D_2O Concentration in Moderator

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APPENDIX
Non-1/v Correction for Dy¹⁶⁴

APPENDIX

Non-1/v Correction for Dy¹⁶⁴

The measurement of ρ_{28} by the thermal activation technique requires a small correction for the non-1/v behavior of Dy¹⁶⁴ when it is used as the intermediate foil. The correction K defined in Equation 2-8 is

$$K = \left(\frac{\sigma_U \phi_U}{\sigma_D \phi_D} \right)_R \left(\frac{\sigma_D \phi_D}{\sigma_U \phi_U} \right)_\phi$$

where the subscripts U and D refer to U²³⁸ and Dy¹⁶⁴ and the products $\sigma_i \phi_i$ are "effective" thermal-reaction rates in the critical experiment lattice under study (R) and in the calibration irradiation in a well-thermalized flux (ϕ).

To determine the importance of the deviation from 1/v, both a 1/v cross section and the Dy¹⁶⁴ cross section, taken from the estimate of $\sigma(E)$ given in WASH-1029, were averaged over a variety of thermal-neutron distributions. The average cross section, defined

$$\bar{\sigma} \equiv \frac{\int_{0.001 \text{ ev}}^{0.625 \text{ ev}} \sigma(E) \phi(E) dE}{\int_{0.001 \text{ ev}}^{0.625 \text{ ev}} \phi(E) dE}$$

was obtained by numerical integration. For each spectrum, g-factors were calculated from

$$g \equiv \bar{\sigma} / \sigma(0.0253 \text{ ev})$$

The deviation from 1/v for Dy¹⁶⁴ was determined by

$$\text{Non-1/v factor} \equiv g(\text{Dy}^{164}) / g(1/v).$$

Three different forms of the distribution function were investigated. The Maxwell-Boltzmann distribution, valid for a graphite moderator with little absorption, was used in the standard form

$$\phi(E) = 1000 E \exp(-E/0.0253).$$

A Wigner-Wilkins distribution, valid for a pure-hydrogen moderator, was obtained from Figure 6 of Reference 9. The particular distribution chosen was 3.73 barns of $1/v$ absorber per hydrogen atom. The third form, probably more valid for D_2O -moderated lattices, is the heavy-moderator distribution reported by Cohen.¹⁰

Relations between these three forms of thermal-neutron distributions were studied and are summarized below:

1. Maxwell-Boltzmann

This distribution is obtained from the Wigner-Wilkins case of zero absorption, and from the heavy-moderator case with the distribution parameter $\Delta = 0$.

2. Heavy Moderator

The distribution parameter Δ is computed from

$$\Delta = 2 \Sigma_a \sum_i \Sigma_{si} / m_i \quad (A-1)$$

for mixtures of moderating materials. In this expression Σ_a is the effective macroscopic absorption cross section in the lattice, Σ_{si} is the macroscopic scattering cross section of the i^{th} element, and m is the mass of the i^{th} element relative to the neutron.

3. Wigner-Wilkins

If hydrogen is assumed to have unit mass and $\sigma_s = 20.5$ b, an effective value of Δ is computed

$$\Delta = \frac{\Sigma_a}{N_H} \times \frac{2}{20.5} \cong 0.1 \frac{\Sigma_a}{N_H}.$$

For comparison, the Wigner-Wilkins distribution for $\Sigma_a/N_H = 3.73$ b/atom and the heavy-moderator distribution for $\Delta = 0.4$ were plotted together. The two distributions agreed sufficiently well that the heavy-moderator approximation can be considered reasonable for D_2O - H_2O

moderated lattices, with the hydrogen counted as mass one.

Non- $1/v$ and g factors are given in Figure A-1 as functions of the distribution parameter Δ . The curves were obtained by numerical integration using $\phi(E)$ from the work of Cohen¹⁰. Equation A-1, with mass one for hydrogen, mass two for deuterium, $\sigma_H = 20.5$ b, and $\sigma_D = 3.4$ b, provided Δ for the critical experiment lattices under study (Figure A-2). The non- $1/v$ factor for each lattice was obtained by cross-plotting Figures A-1 and A-2, and the non- $1/v$ factor for the ϕ -irradiation in graphite was taken from Figure A-1 with $\Delta = 0$ (Maxwell-Boltzmann). The correction K is then

$$K = (\text{Non-}1/v \text{ factor})_{\phi} / (\text{Non-}1/v \text{ factor})_R$$

since the uranium cross section varies as $1/v$. The results are summarized in Table A-1. The uncertainty in K is estimated to be about $\pm 20\%$ of $K-1$ on an absolute basis. The relative error should be much smaller.

Table A-1. Computation of K

Core no.	Enrichment, %	M/W ratio	Moderator, mole % D ₂ O	Δ	Non-1/v factor, R	K
I	4.02	1.006	0.0	0.71	0.971	1.028 $\pm$ 0.006
III	4.02	1.006	76.5	2.49	.922	1.083 $\pm$.017
IV	4.02	1.006	69.7	2.07	.933	1.070 $\pm$.014
IX	4.02	1.006	49.7	1.34	.952	1.050 $\pm$.010
X	4.02	1.195	0.0	0.87	.966	1.033 $\pm$.007
XI	4.02	1.195	70.1	2.44	.924	1.080 $\pm$.016
XII	4.02	1.195	49.7	1.60	.945	1.056 $\pm$.011
XIII	2.46	1.001	0.0	0.44	.980	1.018 $\pm$.004
XIV	2.46	1.001	70.0	1.17	.957	1.043 $\pm$.009
XV	2.46	1.001	49.8	0.79	.968	1.031 $\pm$.006
XVI	2.46	0.651	85.5	1.18	.957	1.043 $\pm$.009
XVII	2.46	0.651	70.0	0.74	0.970	1.029 $\pm$ 0.006

Figure A-1. Non-1/v and g-Factors Vs Δ

Figure A-2. Δ Vs Moderator Composition

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