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**REACTOR PHYSICS
QUARTERLY PROGRESS REPORT,
FEBRUARY—APRIL, 1954**

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ABSTRACT

I. Exponential Experiments

A measurement of the anisotropy in the diffusion of neutrons in a uranium rod lattice has been obtained by comparing the bucklings measured using exponential assemblies of two different diameters. A small anisotropy has been observed.

The measurements of intra-cell flux distribution and buckling for slightly enriched (0.9 per cent) uranium lattices in heavy water have been extended to include a lattice of 1-inch diameter uranium rods on a 6-inch square spacing. Other measurements include the attenuation of fast neutrons in heavy water, thermal neutron disadvantage factors in thorium rods, the attenuation of indium resonance neutrons by cadmium, and the self depression of thermal neutron flux by the foils used for flux distribution measurements.

II. Danger Coefficient Measurements

The danger coefficient technique has been used with the water boiler neutron source to compare the absorption cross sections of 18 samples of beryllium of known purity. Consistency of the measurements indicates that present techniques give a probable error of 0.002 cm^2 in cross section comparisons of this type.

This report is based upon studies conducted for the Atomic Energy Commission under Contract AT-11-1-GEN-8.

Previous Quarterly Progress Reports in This Series

NAA-SR-230 Reactor Physics Quarterly Progress Report, November, December, 1952 - January, 1953
NAA-SR-259 Reactor Physics Quarterly Progress Report, February-April, 1953
NAA-SR-275 Reactor Physics Quarterly Progress Report, May-July, 1953
NAA-SR-854 Reactor Physics Quarterly Progress Report, August-October, 1953
NAA-SR-925 Reactor Physics Quarterly Progress Report, November, 1953-January, 1954

I. EXPONENTIAL EXPERIMENTS

A. A Measurement of the Anisotropy in the Diffusion of Neutrons in a Uranium Rod Lattice (F. B. Estabrook, S. W. Kash, and E. Martin)

The change from a 5-foot diameter to a 4-foot diameter D_2O tank for the NAA exponential experiments has allowed investigation of the possibility that thermal neutron diffusion in a rod lattice may be anisotropic. A sufficient anisotropy would explain the small but persistent discrepancies between Chalk River and NAA measurements of bucklings of natural uranium rod lattices in D_2O .

The one group, one region, diffusion equation for a rod lattice may be written as

$$\frac{\partial}{\partial x} (D_x \frac{\partial \phi}{\partial x}) + \frac{\partial}{\partial y} (D_y \frac{\partial \phi}{\partial y}) + \frac{\partial}{\partial z} (D_z \frac{\partial \phi}{\partial z}) + (k-1) \Sigma_a \phi = 0$$

where D_x , D_y , and D_z are the values of the diffusion constant along the principal directions and the other symbols have the usual meanings. If the geometry is specified to be cylindrical with the z axis along the direction of the rods, the equation may be written as

$$\frac{1}{r} \frac{\partial}{\partial r} (r D_{\perp} \frac{\partial \phi}{\partial r}) + \frac{\partial}{\partial z} (D_{\parallel} \frac{\partial \phi}{\partial z}) + (k-1) \Sigma_a \phi = 0, \quad \dots (1)$$

where $D_{\perp} = D_x = D_y$ is the diffusion constant perpendicular to the rods and $D_{\parallel} = D_z$ is the diffusion constant parallel to the rods.

For an exponential experiment the "fundamental" solution to which the flux data may be fitted has the form

$$\phi = A J_0(\mu r) \sinh \nu (h-z) \quad \dots (2)$$

Substituting this solution into Eq. (1) gives

$$D_{\parallel} \nu^2 - D_{\perp} \mu^2 + (k-1) \Sigma_a = 0 \quad \dots (3)$$

It is convenient to set

$$\frac{D_{\parallel}}{\Sigma_a} = M_{\parallel}^2 \text{ and } \frac{D_{\perp}}{\Sigma_a} = M_{\perp}^2, \quad \dots (4)$$

and measure the amount of anisotropy by δ where

$$\frac{M_{\parallel}^2}{M_{\perp}^2} = 1 + \delta \quad \dots(5)$$

Thus

$$M_{\perp}^2 \mu^2 - M_{\parallel}^2 \nu^2 = M_{\perp}^2 [\mu^2 - (1+\delta)\nu^2] = k-1 \quad \dots(6)$$

If δ is zero we can unambiguously define a lattice buckling

$$B^2 = \mu^2 - \nu^2 \quad \dots(7)$$

If δ is not zero, the B^2 as defined will no longer be constant and characteristic of the medium, but will depend upon the macroscopic dimensions of the lattice. In particular, for a bare exponential assembly, B^2 will depend upon the radius of the lattice. For measurements made in two different tanks, the B^2 will differ by

$$B_4^2 - B_5^2 = \delta (\nu_4^2 - \nu_5^2) = \frac{\delta}{1+\delta} (\mu_4^2 - \mu_5^2) \quad \dots(8)$$

where the subscripts 4 and 5 refer to the 4-foot and 5-foot diameter tanks, respectively. The anisotropy itself will be given by

$$1 + \delta = \frac{\mu_4^2 - \mu_5^2}{\nu_4^2 - \nu_5^2} \quad \dots(9)$$

At the present time the buckling of only one lattice has been measured in both tanks. This lattice consisted of 1-inch diameter natural uranium rods on a 6-inch square spacing. The result was

$$1 + \delta = 1.03 \pm 0.03 \quad ,$$

which is insufficient to completely reconcile the difference between our buckling values and those of the Canadians, but interesting enough to continue the series of anisotropy measurements planned. Additional experimental details are given in the table below. The errors quoted are believed to be limits of uncertainty.

TABLE I
EFFECT OF THE DIAMETER OF AN EXPONENTIAL ASSEMBLY
ON THE MEASURED BUCKLING

Tank Size	μ (cm ⁻¹)	ν (cm ⁻¹)	$B^2 \times 10^4$ (cm ⁻²)
4 feet	0.0386 ± 0.0001	0.0274 ± 0.0003	7.39 ± 0.17
5 feet	0.0310 ± 0.0001	0.01546 ± 0.00015	7.22 ± 0.08

B. Enriched Lattice Measurement (S. W. Kash, B. A. Engholm)

The measurements¹ of thermal neutron buckling and intra-cell flux distribution for slightly enriched (0.9 per cent) uranium-heavy water lattices have been extended to a square array of 1-inch diameter uranium rods on a 6-inch spacing. The experimental results obtained using a 4-foot diameter exponential assembly are as follows:

ν	(Vertical Buckling)	0.0207 ± 0.0001 cm ⁻¹
μ	(Radial Buckling)(assumed)	0.0386 ± 0.0001 cm ⁻¹
B^2	$(\mu^2 - \nu^2)$	$(10.6 \pm 0.1) \times 10^{-4}$ cm ⁻²
F	(Uranium Disadvantage Factor)	1.25 ± 0.02
FF_m		1.92 ± 0.01

C. Thermal Neutron Diffusion Length in D₂O (S. W. Kash)

The diffusion length of thermal neutrons in the NAA D₂O is measured periodically and the results are used to check the purity of the D₂O. Recently, the 5-foot diameter exponential tank on top of the graphite thermal column was replaced by a 4-foot diameter tank, and L² of D₂O was remeasured.

The experimental data yields values for ν and μ , the vertical and radial buckling. From these, L² may be computed and thence the D₂O impurity. Two small corrections must, however, be made: ν is increased to correct for a small influx of epithermal neutrons through the bottom of the tank and μ is increased to account for the flux depression by the vertical foil holder. These corrections are discussed at length in previous NAA reports.^{2,3} The following table presents the results of the measurements in the 4-foot tank and also those of the last measurements in the 5-foot tank:

TABLE II
MEASUREMENT OF THE THERMAL NEUTRON DIFFUSION LENGTH IN D₂O

	4 ft tank	5 ft tank
ν measured (cm ⁻¹)	0.0401 ± 0.0001	0.0329 ± 0.0001
$\Delta \nu$ correction (cm ⁻¹)	0.0001	0.0001
μ measured (cm ⁻¹)	0.0384 ± 0.0001	0.0309 ± 0.0001
$\Delta \mu$ correction (cm ⁻¹)	0.0002	0.0001
L^2 (cm ²)	(79 ± 6) × 10 ²	(78 ± 6) × 10 ²

μ may be computed from the extrapolated radius of the tank by the relation $J_0(\mu a) = 0$. If the extrapolated radius, a , is taken to be the physical radius of the inside of the tank plus an equivalent aluminum thickness for the sides of the tank plus the extrapolation distance, i. e.,

$$a = \frac{ID}{2} + t(A1) \frac{\lambda_{tr}(D_2O)}{\lambda_{tr}(A1)} + 0.71 \lambda_{tr}(D_2O) ,$$

then values for μ of 0.0382 and 0.0308 are obtained for the 4- and 5-foot tanks, respectively. The agreement between these values and the measured values of μ is excellent.

The large error in L^2 is inescapable by the present technique. However, within the limits of the accuracy of the measurements, the purity of the D₂O has apparently not changed.

Using the formulas discussed in a previous progress report,⁴ the D₂O purity is calculated to be 99.70 ± 0.04 atomic per cent. This purity based on the L^2 measurement is in agreement with the value of 99.73 ± 0.02 atomic per cent which was recently obtained by mass spectrographic analysis at the Canadian Chalk River Laboratory.

D. Epithermal Neutron Attenuation Factor in D₂O (S. W. Kash)

It has been found necessary to make a correction² for epithermal neutrons in the diffusion length measurements at NAA. These neutrons enter through the bottom of the tank and provide a distributed thermal source. To correct the measured thermal flux a term proportional to $e^{-\xi z}$ must be added, where ξ is

an apparent attenuation factor for fast neutrons in D_2O . Previous experiments indicated a value for ξ of 0.096 cm^{-1} . Recent, more comprehensive measurements have determined ξ to be 0.117 cm^{-1} . For these measurements, higher flux values obtained and 40-mil thick cadmium boxes covered the indium foils, so that the thermal neutron leakage through the boxes was reduced to a negligible value. The 20-mil cadmium boxes normally used in the exponential measurements provide insufficient thermal neutron shielding for experiments on non-multiplying lattices, where the cadmium ratio is high.

Computations indicate that the epithermal neutrons result from the slowing down of first collision fast neutrons. In particular, if the Cd-covered data are corrected for the $1/R^2$ fall-off, where R is the distance from the center of the water boiler neutron source, the true ξ is found to be 0.097 cm^{-1} . This value is in good agreement with other measurements on the fast neutron removal cross section of D_2O .⁵ The data are plotted in Fig. 1 where the $1/R^2$ corrected curve is also shown. The values quoted for ξ were determined from the datum points nearest the bottom of the tank.

E. Flux Measurements in Thorium Rods (D. H. Martin)

Flux traverses inside a 1-inch diameter thorium rod have been made using dysprosium foils for thermal neutron flux detection. The thorium was in the form of cylindrical slugs which replaced the slightly enriched uranium slugs in one rod of a lattice of uranium rods in D_2O . Measured thermal neutron flux inside the rod is shown in Fig. 2. By integrating under the curve of Fig. 2 with a planimeter, the disadvantage factor or ratio of flux at the surface to the average flux inside the rod was found to be 1.11 ± 0.02 . This experimental value is larger than the diffusion theory value of 1.047, but the discrepancy between theory and experiment is consistent with similar discrepancies observed with uranium rods.

F. Epithermal Neutron Attenuation in Cadmium (D. H. Martin)

A measurement has been made of the factor which is used to correct for the attenuating effect of the cadmium box in making neutron flux measurements using cadmium-covered foils. This factor is obtained by irradiating a group of identical foils in a constant flux, the foils being contained in cadmium boxes which vary in thickness. A semi-log plot of foil activation vs box thickness

Fig. 1. Activity of Cadmium Covered Indium Foils Exposed on the Axis of the D_2O Tank

Fig 2. Thermal Neutron Flux Distribution in a 1-Inch Diameter Thorium Rod

yields a straight line beyond about 0.030-inch box thickness. This straight line represents the attenuation of the neutrons whose energies are above that of the cadmium "cut-off" energy but which activate the foil material contained in the box. Extension of this line back to zero box thickness gives the correction factor required to obtain the true activation of a bare foil by "epi-cadmium" neutrons.

The correction factor was measured in two different neutron spectra using indium foils of two different sizes and several thicknesses. Gold foils were also tested in one of these cases. The exposures were made on a rotating aluminum wheel which was placed in a stringer hole in the graphite of the NAA water boiler neutron source. The resulting correction factors for a 0.040-inch cadmium box, together with earlier measurements, are given in the following table:

TABLE III
CORRECTION FACTORS FOR USE WITH CADMIUM COVERED INDIUM FOILS

Researcher	Foil Weight (mg/cm ²)	Foil Size (cm)	Medium	Cd Ratio (Approximate)	Correction Factor for 0.040-inch Cd Box
Walker ⁶	3.59	2 x 3.1	Water	10	1.077
Dacey, et al ⁷	58.8	3.02 (diam)	Water and Solutions	10	1.120
Rush ⁸	93.5	4 x 6.35	Water	10	1.123
Kunstadter ⁹	94.5	4 x 6.35	Graphite	50	1.150
Martin	94	1.00 x 1.00	Graphite of W. B. N. S.	6	1.098
	88	1.00 x 1.00	Graphite of W. B. N. S.	6	1.094
	69	1.00 x 1.00	Graphite of W. B. N. S.	6	1.09
	29	1.00 x 1.00	Graphite of W. B. N. S.	6	1.09
	88	1.00 x 1.00	Graphite of W. B. N. S.	16	1.095
	88	0.20 x 1.00	Graphite of W. B. N. S.	6	1.10

Within the accuracy of the measurements, the correction factor is not dependent upon indium foil thickness, foil area or cadmium ratio over the range of these variables studied. The corresponding correction factor using gold foils was found to be 1.02.

The effects of foil dimensions and neutron spectrum on the correction factor were investigated in an attempt to resolve discrepancies between Kunstadter's measurements⁹ and those at NAA. As shown by Table III, the discrepancy has not been resolved. For 0.020-inch thick cadmium boxes with 88 mg/cm² indium foils a correction factor of 1.05 is now being used at NAA rather than the 1.07 value used previously.

G. Detector Foil Flux Perturbations (D. H. Martin)

A measurement has been made to determine the amount of self perturbation of thermal neutron flux caused by the detector foils currently in use at NAA. Of particular interest were the 1 centimeter by 2 millimeters by 0.005 inch foils of lead-indium alloy which are used in intra-cell flux distribution measurements in uranium-heavy water lattices. In order to check the perturbation caused by these foils, identical flux traverses were made through 1-inch diameter uranium rods and into the surrounding heavy water, using bare and cadmium covered foils 0.005 inch thick but with three different areas: 1 centimeter by 2 millimeters, 5 millimeters by 2 millimeters, and 2 millimeters by 2 millimeters. The rods used were 0.901 per cent (enriched) uranium and 0.49 per cent (depleted) uranium. Five normalization foils were placed at 6-inch intervals along the axis of the tank to provide an absolute comparison between runs.

Analysis of the data reveals no significant differences in the shape of the curves obtained with different size foils. In addition, plots of foil activation per unit volume vs foil size were obtained with the aid of the normalizing foils and no change in activation as a function of foil size was indicated. Thus, within the estimated 2 per cent accuracy of the experiment, no perturbation of the neutron flux is caused by the foils used to measure intracell flux distributions.

II. DANGER COEFFICIENT MEASUREMENTS

(G. E. Modesitt and B. A. Engholm)

A. Danger Coefficient Measurements on Uranium

In the previous quarterly report¹⁰ preliminary results were given for a comparison by the danger coefficient technique of the uranium used at NAA with that used in the ZEEP at Chalk River. This comparison was undertaken in an attempt to locate the source of discrepancies in lattice buckling measurements at the two sites. Additional measurements and analysis of data indicate that the uncertainty of 0.001 cm^2 in the absorption cross section measurements reported previously was somewhat optimistic. The probable error of the danger coefficient method as used for the uranium comparisons is now believed to be about 0.003 cm^2 .

B. Modification of the Danger Coefficient Apparatus

In order to shorten the time necessary for each danger coefficient measurement and to improve the accuracy of the measurements, a "through tube" system was installed in the water boiler reactor. The "through tube" consists of a magnesium tube 12 feet long, 7/8-inch in diameter, and 1/16-inch wall thickness which is inserted through the central exposure facility of the water boiler and extends beyond the concrete shield on either side of the reactor. Four solid lucite rods, each 3 feet in length and just under 7/8-inch in diameter serve as fillers for the region inside the tube and also as push rods for pushing the samples into the central exposure facility. By eliminating the necessity for shutting down the reactor when changing samples, the through tube system was expected to speed up the measurements. A further increase in the measurement rate was introduced by using a single reactivity measurement on each sample instead of three as was previously done. The time required for a complete comparison of an unknown sample with a standard is now just under 1/2 hour compared to the previous value of about 1-3/4 hours. After installation of the through tube, the control rod was recalibrated using the boron standards.¹¹ The new calibration curve differed only slightly from the old one.

C. Danger Coefficient Measurements on Beryllium

A series of measurements on the feasibility of using the water boiler as a production test facility for the determination of the absorption cross section

of impurities in beryllium was begun. Eighteen beryllium samples, 1/2-inch in diameter, were supplied by the Brush Beryllium Company. There were six samples in each of three lengths, 1 inch, 2 inches, and 4 inches. A known amount of cobalt or silver impurity was present in 12 of the samples. A total of 94 measurements was made using the beryllium samples and the "through tube" arrangement. The reproducibility of the measurements indicates that the probable error for cross section comparisons using the new arrangement is 0.002 cm^2 .

The largest sample supplied was 1/2-inch in diameter and 4 inches long, giving a beryllium absorption cross section of 0.015 cm^2 . The probable error in the cross section comparison is then 13 per cent of the beryllium cross section for this sample. Larger beryllium samples could be used without affecting the sensitivity of the measurements. Samples 3/4-inch in diameter and 8 inches long would be a convenient size for the present danger coefficient apparatus. These larger samples would have a beryllium cross section of 0.066 cm^2 and the probable error of a measurement would be 3 per cent of the absorption cross section of the beryllium.

Investigations into the sources of error are proceeding. It is hoped that the cause of the apparent small reactivity drifts can be located and corrected so that the accuracy of the measurements can be improved. Also, if the reactivity drifts can be decreased, it may be possible to increase the speed of the measurements by using the standard sample between every second or third unknown sample rather than before and after every unknown sample as is done at present.

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