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QUARTERLY REPORT - PHYSICS UNIT

D. E. Davenport
J. E. Faulkner
J. Ozeroff

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PHYSICS UNIT

October, November, December, 1953

By

D. E. Davenport, J. E. Faulkner, J. Ozeroff

Applied Research Sub-Section - Technical Section
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INTRODUCTION

This report is intended to be a description of current work being done by the Physics Unit at the Hanford Atomic Products Operation. The work is concerned mainly with the study of graphite-moderated natural uranium reactors for the production of plutonium on a large scale. In addition, varied physics problems associated with the operation of an industrial atomic plant are occasionally dealt with.

LATTICE PHYSICSExponential Experiments

Measurements have proceeded in two exponential piles with lattice spacings of 6-3/16 inches and 7-1/2 inches. An effective lattice spacing of 12-3/8 and 15 inches have also been obtained by removing alternate process tubes from the piles and filling the holes with graphite bars. The 7-1/2 inch lattice pile consists of a lattice cube, 8 feet 5 inches on a side with a 20-inch layer of graphite upon which the pile is supported. The dimensions of the 6-3/16 inch lattice pile are 96 inches by 97 inches with a height of 97 inches and a supporting graphite base of 20 inches. The neutron flux was supplied by four 1/2-gram Ra-Be neutron sources which each have a neutron emission of about 6.6×10^6 neutrons/sec. These sources were placed symmetrically in a diamond array in the base of the lattice at $(\pm a/4, 0)$ and $(0, \pm b/4)$, where a and b are the effective widths of the pile; that is, the physical dimension plus twice the extrapolation length ($2\lambda = 2"$). BF_3 counters and cylindrically mounted In foils were used for neutron detection.

The buckling was measured with large hollow slugs (1.66" O.D. and 0.81" I.D.) in the 6-3/16, 7-1/2, 12-3/8, and 15 inch lattice spacings. The measurements were taken with and without the internal hole filled with water. The results of these measurements are summarized below, where the buckling unit is 10^{-6} cm^{-2} .

Slug Dimensions: I.D. = 0.81", O.D. = 1.66"

$V_{\text{H}_2\text{O}} = 2.37 \text{ cc/cm}$ $V_{\text{Al}} = 1.9 \text{ cc/cm}$

<u>Lattice Dimension</u>	<u>Dry Buckling</u>	<u>Wet Buckling</u>
6-3/16"	-66.5 ± 1	-52.6 ± 1
7-1/2"	98.1 ± 1	92.7 ± 1
12-3/8"	51.1 ± 3	35.0 ± 3
15"	-29.7 ± 3	-41.2 ± 3

The buckling for a slug of 1.176-inch diameter was measured in the 6-3/16 and 7-1/2 inch lattices dry and with water in the cooling annulus. The lattices were adapted for the smaller slug size by the insertion of graphite sleeves in the process-tube holes. The amount of water per unit of length in the cooling annulus was 1.73 cc per cm length. The values for the buckling are listed below:

<u>Lattice Dimension</u>	<u>Dry Buckling</u>	<u>Wet Buckling</u>
6-3/16"	77.2 ± 1	83.9 ± 1
7-1/2"	122.0 ± 1	98.6 ± 1

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Future plans of the exponential pile experiments include measurement of the effect of water annulus variation on buckling. These experiments are designed to check theoretical equations purporting to predict the effect of different water annuli.

At present, two slug sizes and three different water annuli are planned.* The dimensions planned and progress of material procurement are tabulated below.

Slug Size**	Annulus		Procurement Status
	Thickness	Volume	
0.926"	0.123 cm	1.03 cc/cm	Experiment partially complete. All equipment at hand.
0.926"	0.229 cm	1.976 cc/cm	Tubes are ready, all equipment on hand.
0.926"	0.469 cm	4.417 cc/cm	Polystyrene rings and process tubes on hand.
1.36"	0.193 cm	2.33 cc/cm	Experiment complete, reported in HW-27354.
1.36"	0.320 cm	3.60 cc/cm	Work on die for making polystyrene rings has started

The reboring of the 1.66" O.D. by 0.810" I.D. hollow slugs to an I.D. of 1.100" is 80% complete. The expected work completion date is January 10, 1954. Much difficulty has been encountered in reboring the slugs because of tool breakage. Work hardening of the uranium, slack in the turret lathes and inexperienced operators are prime causes of the "slow-up".

Plans have been made for the measurement of the bucklings of lattices loaded with regular Hanford slugs (1.36" O.D.) with a 1/2" hole drilled down the axis to provide for expansion of the metal at high operating temperatures. The actual drilling of the slugs cannot begin until the present reboring of hollow slugs is complete. However, the one-half inch aluminum plugs that are to fit into the ends of the slugs to keep out the cooling water have been machined. The one-half inch drills have been ordered and the man-power scheduled with operations to start on January 15, 1954.

Uranium Quality

There has been some question of the uranium quality of the different size slugs used in the exponential pile measurements--primarily due to the fact that the metal was obtained at different times. In the exponential experiments, the nuclear qualities are the only qualities of interest and require only a check of the reactivity of the metal. These tests have been run in the 305 Test Pile and the results compared with the standards used by the Manufacturing Department in major pile reactivity determinations for the past six years. The check shows that the quality of the large (1.66") slugs does not vary from the 1.36" slugs by any amount larger than the statistics of the measurement. The 0.926"

* Several lattices will be measured; but, the initial measurements will take place in the 6-3/16" and 7-1/2" lattices.

** All dimensions are the bare metal dimensions. The 0.926" slugs are placed in 1.00" Al cans. The 1.36" slugs are 1.44" cans and are the regular Hanford slugs.

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and 1.176" slug quality checks have not been completed.

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Small Exponential Pile Experiment

Exponential piles of the current Hanford Size utilize rather large quantities of graphite and uranium. The graphite amounts to 30-35 tons and the uranium varies between 5 and 15 tons depending on the slug diameter and lattice spacing, etc. These larger piles are required if the buckling is to be determined with high accuracy due to the uncertainty in the extrapolation length. In the 8' x 8' piles, an uncertainty of $\pm 1/4$ inch in the effective size of the pile will result in an uncertainty in the measured buckling of about $\pm 3 \times 10^{-6} \text{ cm}^{-2}$ or 3 microbucks. This same uncertainty in size of a small 4' x 4' x 4' pile will lead to an uncertainty in the buckling of about $\pm 25 \times 10^{-6} \text{ cm}^{-2}$, which is an intolerably large one.

However, if a small pile is built and the buckling measured of a known lattice, that is a lattice spacing which has previously been measured in the larger pile, then it might be possible to adjust the extrapolation length so as to give the correct buckling. Having once been determined, this effective extrapolation length could be used for the measurement of unknown lattices with the small pile. The extrapolation length would be expected to remain reasonably constant with lattice spacing. It is evident that if a small pile could be used, with some sacrifice in accuracy, a large number of lattices could be surveyed at a considerable reduction in the graphite and uranium required. (About one-eighth the amount of graphite and uranium depending on the size). There would also be a saving in the construction time of the pile, loading time, etc.

With these ideas in mind, a small pile was built. The dimensions of this pile are 48" x 47.7" and the height is 61.9". The pile is supported by a graphite base layer which is 48" x 47.7" by 19.9" high. Cadmium sheet was placed on the top, sides and the bottom of the supporting graphite layer, so as to reduce the reflection of neutrons into the pile from the walls and floor of the building. The lattice spacing in this pile is 6-3/16".

The buckling was measured with three different slug sizes. The 1.66" O.D., 0.81" I.D. slug with the internal hole filled with water, was measured first. Following this, the 1.36" diameter standard slugs were used. These slugs were then removed and graphite sleeves inserted so as to accommodate slugs of diameter 0.926". Four Ra-Be neutron sources were placed symmetrically in a diamond array in the base of the pile at $(0, \pm a/4)$, $(0, \pm b/4)$; a and b being the effective widths of the pile. A BF_3 counter was used for detection of the neutrons. Measurements were taken at five positions on the vertical center line of the pile. The vertical distances above the source plane for these positions are 55.0, 77.7, 86.4, 102.2, 117.9 cm. The cadmium ratio was low in all cases at the first measurement position (55.0 cm from source plane). Therefore, only the top four measurement positions were used in determining the relaxation length. Since the slowing down ranges to thermal for the three groups of Ra-Be neutrons is 31, 43, and 64 cm, it is not unreasonable to expect a deviation in the cadmium ratio at the 55 cm position.

For each traverse, the relaxation length was determined and the extrapolation length adjusted to give the best over-all agreement between the known buckling and the value obtained from the small pile measurement. The bucklings are

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compared below for an extrapolation length of 1.33 inches, which gives approximately the best fit to the three lattices.

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<u>Lattice Spacing</u>	<u>Slug Size</u>	<u>Known Value x 10⁶ cm²</u>	<u>Small Pile Value x 10⁶ cm²</u>
6-3/16"	1.66" O.D., 0.81" I.D. (Wet)	-52.7	-45.3
6-3/16"	1.36" (Dry)	2.0	-5.0
6-3/16"	1.17" (Wet)	83.9	81.2

These results would suggest an uncertainty of ± 10 microbucks in the value of the buckling as measured in the small pile. Only in the case of the 1.36" slug size was the uranium confined to the lattice region. For the other two slug sizes the process tubes with uranium extended beyond the surface of the pile (about 2 feet on each side). This might possibly produce a perturbation on the thermal flux or in effect change the extrapolation length. The least-squares fit to the data was best for the 1.36" slug size. There are two things which can be done that should increase the accuracy of the measurements. The addition of several layers to the top of the pile will permit a more accurate determination of the relaxation length since it will provide several additional measurement points, and the thermal source approximation will be better for positions further from the sources. Confining the uranium to the lattice region of the pile in all cases will eliminate any possible extraneous effect from uranium outside the lattice region. With the incorporation of these items it should be possible to obtain the buckling with this small pile to ± 5 microbucks which would be accurate enough for purpose of survey measurements.

Core Measurement of Buckling

A reduction in the amount of graphite and uranium required for exponential measurements might also be accomplished by a core type of buckling measurement. For this measurement the central region of the 8' x 8' pile is loaded with the slug under investigation while the blanket material is composed of some readily available slug size. This would seem to offer the potential advantage of less uncertainty due to effective pile size error, provided the blanket-core buckling differences can be corrected for.

The process tubes in the central region of the 8' x 8' exponential pile with 6-3/16" lattice spacing were removed and graphite sleeves inserted so as to accommodate a different slug size in this region of the pile. At present, three different core sizes have been used with two different core bucklings. The blanket material was the 1.17" slug size with water in the cooling annulus (buckling ≈ 84 microbucks). The size of the core section was seven lattice units high and the widths used were 4, 6, and 8 lattice units. The core region extended from the front to back surface of the pile. (See Figure I).

The relaxation length was determined for each core width with a dry 0.926" slug loading (buckling ≈ 114 microbucks). If the thermal source theory is applied to the core section disregarding the difference of its buckling from that of

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FIGURE I

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FRONT ELEVATION 6 3/16" LATTICE
CORE LOADING

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the blanket lattice ($\Delta B = 30$ microbucks), then the value of the buckling for the 8×7 lattice unit loading is obtained with an uncertainty of about ± 7 microbucks due to the curvature in the $\ln \phi$ versus z plots from which the relaxation length is obtained. For the smaller core sections, the calculated buckling is progressively in greater error, due to the perturbation of the surrounding lattice being more pronounced. An 8×7 lattice unit core section of the 0.926" slugs with water in the cooling annulus was also measured. In this case, the buckling difference between core section and surrounding material is ~ 14 microbucks. The buckling obtained from the relaxation length in the core section is in agreement with the known value within the experimental accuracy of ± 1 microbuck. It appears likely that the above method will be feasible where the buckling difference is not greater than about 15 microbucks. Further work will be done developing theoretical corrections for the difference in buckling between the blanket and core lattices and determining the minimum core size necessary for accurate buckling measurements.

Lattice Diffusion Length Measurement

Previous diffusion length measurements had been taken in the 6-3/16", 7", 7-1/2", 8", and 8-3/8" lattices with P-10 slugs replacing the natural uranium. The diffusion length was measured both with water in the cooling annulus and dry. (P-10 slugs are composed of an aluminum-lithium alloy. These slugs were originally produced for the production of tritium from the ${}^3\text{Li}^6 ({}_0\text{N}^1, {}_2\text{He}^4), {}_1\text{H}^3$ reaction.) Since the total cross section for absorption was about the same as for the uranium slugs, a diffusion length in this material might yield information about the diffusion length calculations for a uranium lattice. At present, the calculated and measured diffusion lengths in the P-10 material are in disagreement by

about 10%; that is, the formulas $L^2 = \sum L_i f_i$ or $L^2 = \frac{1}{3N\sigma_a N\sigma_t}$ do not seem to

predict the physical measurement. However, before definite conclusions can be reached regarding the above discrepancy, it is apparent that additional cell traverses must be taken so as to obtain very accurate flux-volume cross section averaging over the lattice cell.

Cell Traverse

Intra-cell traverses of the neutron flux were taken previously in the 6-3/16, 7, 7-1/2, 8, and 8-3/8 inch lattices with metal diameter of 1.36 inches and in the 6-3/16 and 7-1/2 inch lattices with metal diameter of 0.926 inches. Flat indium disks, 3/4 inch in diameter and about 100 mg/cm² in thickness, which were mounted on thin graphite wafers, were used as the detector. Bare and cadmium-covered exposures were taken in the moderator of the lattice. The thermal utilizations estimated from these traverses are as follows:

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<u>Slug Size</u>	<u>Lattice Spacing</u>	<u>f_u (wet)</u>	<u>f_u (dry)</u>
1.36"	6-3/16"	0.910	0.939
1.36"	7"	0.891	0.924
1.36"	7-1/2"	0.881	0.915
1.36"	8"	0.871	0.905
1.36"	8-3/8"	0.868	0.901
0.926"	6-3/16"	0.874	0.905
0.926"	7-1/2"	0.833	0.868

Due to the low flux values in the exponential piles it is difficult to obtain traverses of the neutron flux in the uranium slug of sufficient accuracy to be of interest. An accurate traverse of the 1.36" diameter slugs was accomplished by an exposure of a slug in the 305 Test Pile. In the case of the 0.926" slug, it was necessary to assume the shape of the flux in the metal characteristic of an $I_0(\sqrt{\kappa}r)$ function determined by the surface to center ratio of the flux which had been measured previously. Therefore, the above values for the 0.926" slug size are subject to considerable uncertainty.

Correlation of Lattice Measurements with Theory

Using Project Handbook type formulas for the calculation of thermal utilization and resonance escape probabilities, and in particular those formulas of Rumsee and Volkoff (MT-221), an effort has been made to correlate the experimentally measured bucklings with those calculated from theory.

In doing this several assumptions were made as follows:

- 1) The thermal neutron absorption cross sections for graphite, uranium aluminum, and water were assumed to be known.
- 2) The thermal utilization could be calculated, as well as the lattice diffusion length and Fermi age.
- 3) The resonance cross section necessary for the calculation of resonance escape (p) was not known accurately and could be varied within reasonable limits to insure a fit.
- 4) The product $\eta\epsilon$ was also available for adjustment.

The method of calculation is as follows:

The critical buckling may be represented by the usual formula:

$$B = \frac{k - 1}{L_t^2 + k\gamma}$$

k = multiplication constant, L_t = lattice diffusion length.

γ = Fermi age.

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For any particular slug size we may reasonably assume that $\eta\epsilon$ is constant. If we use the preceeding equation and solve for $\eta\epsilon$ we obtain:

$$\eta\epsilon = \frac{1 + BL_t^2}{pf(1-B\tau)}$$

The bucklings for five different lattice spacings with 1.36" diameter slugs (standard Hanford slug size) were measured both wet and dry; that is with water in the cooling annulus (thickness 1.925 mm) and without.

We may then write $\eta\epsilon$ for these ten different cases (5 wet and 5 dry) in terms of the measured buckling, calculated lattice diffusion length, Fermi age, thermal utilization, and resonance escape. By equating $\eta\epsilon$'s, we obtain an equation in two unknowns (two resonance escape probabilities) assuming the other quantities known:

$$\frac{p_1 f_1 (1 - \tau_1)}{1 + B_1 L_1^2} = \frac{p_2 f_2 (1 - B_2 \tau_2)}{1 + B_2 L_2^2}$$

Now using the expression for p in the above, it is possible to calculate the value of the effective resonance cross section which will insure that the calculated buckling will be in agreement with the measured value. By the above method we can obtain an average value of σ_{eff} which will best fit our data, and also fix the product $\eta\epsilon$ as well. The effective resonance cross section which best fits the data is 1.64 barns for the 1.36" slug size. From this cross section, the volume part of the resonance integral may be calculated to be 7.87 barns if the empirical expression

$$\int \sigma(E) \frac{dE}{E} = A \left(1 + \frac{2.5}{M/S + 0.085} \right)$$

is assumed for this integral. It is the product $\eta\epsilon$ which is fixed by the above analysis. If the value of ϵ is calculated then an η of about 1.29 can be implied.

The constant η and the calculated resonance cross section for a 0.926" slug, based on the 7.87 barn cross section for the volume part of the resonance integral, appear to fit the 0.926" slug size measured bucklings reasonably well. A document is being compiled which will cover this analysis and its extension to other slug sizes in detail.

Thermal Blackness of a Solid Cylindrical Slug

We have issued a formal report, HW-29992, on the following subjects. The thermal blackness, i.e., probability of capture for an incident thermal neutron, of a solid cylindrical slug is formulated in terms of probability integrals on a transport theoretic basis. A simple extension produces the thermal blackness of a dry process tube assembly. Calculation of the detailed thermal spectrum on slug centerline through application of the transport integral equation permits us to correct our blacknesses for spectral hardening effects. Additional applications of the latter technique are pointed out. Using small-source

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theory, thermal utilization is calculated directly from process-tube assembly thermal blackness. Sample numerical calculations are performed and checked with experiment.

Calculated Thermal Blackness of P-10 Process-Tube Assemblies

In HW-29992, report mentioned above, a value for the blackness of a P-10 process tube assembly was calculated, from first principles to be, $\rho = 0.58$. This result is too large to produce agreement with the single layer measurements conducted by R. K. Cole who reported $\rho = 0.506 \pm 0.018$.* When the existence of end caps is implicitly considered by reducing the density of P-10 material the requisite amount, one obtains the reduced thermal blackness, $\rho = 0.53$. Thermal spectral hardening would reduce ρ again by approximately 0.01 yielding, $\rho = 0.52$.

This latter figure is within the experimental accuracy of Cole's measurements and serves as a positive vindication of both measurements and calculation.

To further buttress the above, we quote the small-source analysis of exponential pile data performed by H. Neumann (HW-29454) who obtained a P-10 process-tube assembly thermal blackness of 0.508 ± 0.020 , which is also in excellent agreement with our number.

Transport Theory - Flux Traverse in a Fuel Slug

It is recognized that thermal flux distributions within so highly absorbing a medium as a fuel slug will not be accurately presented as results of elementary diffusion theory. Since these distributions are of practical interest in the computation of thermal utilizations, it is necessary to turn to the more accurate transport theory for their construction. Two region (in our case fuel rod and moderator) transport theory problems are intrinsically of great difficulty, it being necessary to take recourse to various approximation schemes, such as the spherical harmonic method, in order to extract useful information from them. In the following, we shall develop a new method of obtaining approximate solutions to multiregion transport theoretic problems which is based on the integral equation formulation of transport theory.

In HW-29110, the integral equation determining the thermal flux on the center-line of an infinite solid cylindrical slug embedded in a graphite moderator was revealed as

$$\begin{aligned} \psi(0) = & \frac{\sum_{SI} \sum_I}{2\pi} \int_0^{2\pi} d\mathcal{V}' \int_0^{R_I} \psi_I(r') r' dr' \int_1^\infty K_0(\sum_I r' x) dx \dots \\ (1) \quad & \dots + \frac{\sum_{SII}}{2\pi} \int_0^{2\pi} d\mathcal{V}' \int_{R_I}^\infty \psi_{II}(r') r' dr' \int_1^\infty \left[\frac{\mathcal{V}_{R_I}}{r'} + \sum_{II} \right] K_0 \left[(\mathcal{V}_{R_I} + \sum_{II} r') x \right] dx, \end{aligned}$$

identical nomenclature to the above quoted report being used (with the exception that we replace ρ with $\mathcal{V}$). It is not in general possible to construct in practical form the necessary Green's functions required to produce an integral equation determining the flux for all r , i.e., the equation in symbolic form,

* Cole, R. K., HW-30147

$$(2) \quad \varphi(r) = \int \varphi(r') G(r|r') dr'$$

due to the geometric complexities of optical pathlength. We shall base this work on our ability to displace the observation point (aufpunkt) an infinitesimal distance from the slug centerline and still write down the various Green's functions correct to the proper order of small quantities. Such an operation generalizes equation (1) to

$$(3) \quad \varphi(r) = \frac{\sum_I \sum_{SI}}{2\pi} \int_0^{2\pi} d\mathcal{V}' \int_0^{R_I} \varphi_I(r') r' dr' \int_1^\infty K_0 \left(\sum_I \sqrt{r^2 + r'^2 - 2rr' \cos \mathcal{V}'} \right) x dx$$

$$+ \frac{\sum_{SII}}{2} \int_0^{2\pi} d\mathcal{V}' \int_{R_I}^\infty \varphi_{II}(r') r' dr' \int_1^\infty \left[\frac{\sqrt{r^2 + R^2 - 2rR \cos(\mathcal{V}' - \frac{r'-R}{r'R} \sin \mathcal{V}')}}{\sqrt{r^2 + r'^2 - 2rr' \cos \mathcal{V}'}} \right] \{ + \sum_{II} \dots \}$$

$$\dots x \dots K_0 \left[\left(\sqrt{r^2 + R^2 - 2rR \cos(\mathcal{V}' - \frac{r'-R}{r'R} \sin \mathcal{V}')} \right) \right] \dots$$

$$\dots + \dots \sqrt{r^2 + r'^2 - 2rr' \cos \mathcal{V}'} \sum_{II} \} x \, dx,$$

where $r \ll R_I$.

Equation (3) may now be differentiated n times with respect to r and the limit $r \rightarrow 0$ then passed to, revealing an integral equation which generates the nth derivative of the flux distribution on slug centerline. In the particular case considered here, all odd derivatives are seen to vanish by symmetry arguments.

The lowest non-vanishing derivative is therefore of second order and, after integration over $\mathcal{V}'$, may be written

$$(4) \quad \varphi^{(2)}(0) = \frac{\sum_I^2 \sum_{SI}}{2} \int_0^{R_I} \varphi_I(r') F_{j2}(\sum_I r') dr' + \frac{\sum_{SII}}{2} \int_{R_I}^\infty \varphi_{II}(r') r' \left\{ \left[\frac{R_I}{r'^3} - \frac{1}{R_I r'} \right] \frac{F_{j0}(\sum_I R_I + \sum_{II} r')}{\sum_I R_I + \sum_{II} r'} \right.$$

$$\dots + \left[\frac{\sum_I^2 - 2 \sum_{II}^2 + 6 \sum_I \sum_{II}}{r'} - \frac{\sum_{II}^2}{R_I} - \frac{R_I \sum_{II}^2}{r'^2} \right] \left[K_1(\sum_I R_I + \sum_I r') - F_{j2}(u) \right]$$

$$\left. \dots + \frac{\sum_I^2}{2} \left[\frac{R_I}{r'} + \sum_{II} \right] \left[\frac{F_{j2}(u)}{u} - K_2(u) - (u)K_1(u) + (u)F_{j4}(u) \right] \right\} dr',$$

where

$$u = \int R_I + \sum_{II} \sim$$

In the above equation

$$(5) \quad F_{jn}(x) = \frac{1}{x^n} \int_x^\infty t^n K_0(t) dt = \frac{2^{n-1}}{x^n} \left[\Gamma\left(\frac{n+1}{2}\right) \right]^2 - K_{jn}(x) ,$$

where the $K_{jn}(x)$ have been extensively tabulated by G. M. Muller (HW-30323).

If one assumes that diffusion theory is approximately correct, the flux within a slug is approximately proportional to $I_0(\mathcal{K}_I \sim)$. This allows the approximate definition of an inverse diffusion length, $\mathcal{K}_I$. Expansion of $I_0(\mathcal{K} \sim)$ in series form about $\sim = 0$ and identification of the appropriate second derivative term relates $\mathcal{K}_I$ to equations (4) and (1).

$$(6) \quad \mathcal{K}_I = \sqrt{\frac{2\psi^{(2)}(0)}{\psi(0)}} .$$

Were the more accurate non- I_0 flux distribution within the slug required, we could determine, through an identical procedure, the higher order derivatives of thermal flux at $\sim = 0$ and expand in Taylor series about this point.

$$(7) \quad \psi(r) = \psi(0) + \frac{r^2 \psi^{(2)}(0)}{2!} + \frac{r^4 \psi^{(4)}(0)}{4!} + \dots$$

Experimental traverses and P_3 approximation results yield flux distributions which correspond to $\mathcal{K}_I$ being a radially increasing function of $\sim$. Stated alternately, the center of the slug supports a flux distribution which is flatter, and the periphery of the slug, a flux distribution which is steeper than the diffusion theoretic result. Were an analogous $\mathcal{K}_I$ found for the fourth derivative term, through a procedure similar to equation (6), it should therefore exceed in magnitude the result of equation (6).

Previous experience with integral equations of type equations (1) and (4) reveals an insensitivity to the form of the flux distribution placed under the integral sign. In practice, therefore, we replace $\psi_I(r')$ and $\psi_{II}(r')$ by unity. The integrals are not tractable to analytic expression and must be performed numerically. If greater accuracy is desired, a second iteration may be performed after $\mathcal{K}_I$ is determined from equation (6) by substituting the well known diffusion theory functional forms corresponding to this $\mathcal{K}_I$ under the integral signs. Still greater accuracy would repeatedly iterate equation (7), a procedure which, in theory at least, would ultimately converge to the exact transport theoretic flux distribution. Of course, in practice these manipulations rapidly become impractically complex.

The techniques developed above have been illustrated by consideration of the thermal flux distribution in a cylindrical fuel element. They are, however,

applicable to any problem of neutron diffusion possessing a geometric point of sufficient symmetry to allow construction of Green's functions for that point. A possibly fertile field of application would be to fast tamped reactors, which also lie within the domain of transport theory.

Table of the Function $Kj_n(x) = \frac{1}{x^n} \int_0^x u^n K_0(u) du$

This table which has been prepared on the Hanford IBM equipment is almost ready for publication. It covers the range $n:0(1)31$, $x:0(0.01)2(0.02)5$. Final checks indicate that the accuracy of all table entries is approximately one part in 10^7 .

Small-Source Theory

Methods for measuring the small-source theory parameters with present facilities are being considered. An exponential pile was made available by the Exponential Physics Sub-Unit for these measurements. They also gave much valuable assistance in performing the experiments.

The small-source theory parameters to be measured are the thermal blackness (ρ), resonance blackness (σ), and the constant ($h = \eta \epsilon$).

From a previous experiment, it was determined that ρ could be measured by loading one layer of metal in an exponential pile and measuring the flux above this loaded layer. The rest of the pile was empty and the sources were in their normal position in the bottom of the pile. From the ratio of the flux measured before and after the layer was loaded, the thermal blackness can be calculated. An experiment was performed using canned slugs of lithium-aluminum, which is an absorber of thermal neutrons. These slugs were inclosed in an aluminum process tube. This experiment was then repeated with only two columns of metal instead of a whole layer and consistent results were obtained. Then, one column was loaded and the flux was measured at several points along a line through the center of the column and perpendicular to it. These were then compared to calculated values at these points and found to agree within experimental error.

The layer experiment was repeated with natural uranium (Hanford standard slug) and from the ratio of the flux measurements before and after the pile was loaded, relationship between ρ , h , and σ can be found.

This was repeated for just one column of uranium metal. For one column,

$$R-1 = \frac{\pi a \beta e^{\chi(Z-Z_1)}}{2} \left\{ \frac{h \left[K_0(\chi \rho) - K_0\left(\frac{\rho}{\theta}\right) \right]}{2 \pi \theta D \left(\frac{1}{\theta} - \chi^2 \right)} \frac{K_0(\chi \rho)}{2 \pi D} - \frac{a^2 \chi e^{-a^2/4\theta_0} \left[K_0(\chi \rho) - K_0\left(\frac{\rho}{\theta - \theta_0}\right) \right]}{4 \pi D \theta_0 (\theta - \theta_0) \left(\frac{1}{\theta - \theta_0} - \chi^2 \right)} \right\}$$

$$\frac{1 - a \rho \left\{ \frac{-K_0(\chi a)}{4D} + \frac{h \left[K_0(\chi a) - K_0\left(\frac{a}{\theta}\right) \right]}{4 \theta D \left(\frac{1}{\theta} - \chi^2 \right)} + \frac{\chi K_1(\chi a)}{2} \right.}{- a \chi e^{-\frac{a^2}{4\theta_0}} \left\{ \frac{K_0(\chi a) - K_0\left(\frac{a}{\theta - \theta_0}\right) + \frac{2D}{\chi \theta_0} \frac{K_1\left(\frac{a}{\theta - \theta_0}\right)}{\sqrt{\theta - \theta_0}} - 2D \chi K_1(\chi a) \right\} + \frac{h \left[\frac{1}{\theta} K_1\left(\frac{a}{\theta}\right) - \chi K_1(\chi a) \right]}{2 \theta \left(\frac{1}{\theta} - \chi^2 \right)} \right\}}$$

where

a = radius of column

$\frac{1}{\lambda}$ = diffusion length of graphite

ρ = perpendicular distance from center of columns to point of measurement

R = ratio of thermal fluxes measured before and after metal column was loaded

θ_0 = age of resonance

θ = age to thermal

Z = height of point of measurement above neutron sources

Z_1 = height of metal column above sources.

To obtain another relationship between these constants, a series of measurements were taken perpendicular to the single-loaded column and parallel to the base of the pile. The sources were loaded in the pile base. This horizontal traverse was made with small flat indium foils of $3/4$ inch diameter, first bare and then cadmium-covered. From these two traverses, two curves were drawn which show the thermal neutron sink effect of the uranium and the slowing down of the fast neutrons produced by the uranium. However, because of insufficient strength of the sources to drive the experiment and a relatively high background in the counters, the resonance absorption effect was masked by the counting statistics. The traverse was repeated by the Exponential Pile Sub-Unit with cadmium-covered gold foils, and the expected effect of the resonance absorption found. To increase the exposure of the foils, and therefore the counting statistics, five sources were placed 4 inches apart along a line $8-3/8$ inches directly below the metal column and parallel to it. A horizontal traverse was made perpendicular to the column. This was repeated with the uranium covered with cadmium and another traverse taken at the same position with just the sources in the pile.

If the pile is considered infinite the flowing down density becomes, for the cadmium-covered uranium column,

$$q(r\theta) = \sum_{\substack{\text{over source} \\ \text{ages}}} S_{\theta} \left\{ \frac{1}{(4\pi\theta)^{3/2}} \left\{ 2e^{-\frac{(b^2+Z_0^2+X^2)}{4\theta}} + 2e^{-\frac{(4b^2+Z_0^2+X^2)}{4\theta}} \right. \right. \\ \left. \left. + e^{-\frac{(Z_0^2+X^2)}{4\theta}} \right\} - \frac{4\pi a r \sqrt{\pi\theta_0(\theta-\theta_0)}}{\sqrt{\theta(4\pi\theta_0)^{3/2}} [4\pi(\theta-\theta_0)]} \right\}^{3/2} \left\{ 2e^{-\frac{\theta\theta_0(Z_0^2-X^2-4b^2)+4b\theta_0Z^2\theta^2}{4\theta_0\theta(\theta-\theta_0)}} \right. \\ \left. + 2e^{-\frac{\theta\theta_0(Z_0^2-X^2-b^2)+b^2\theta_0^2-Z^2\theta^2}{4\theta_0\theta(\theta-\theta_0)}} + e^{-\frac{\theta\theta_0(Z_0^2-X^2)-Z^2\theta^2}{4\theta_0\theta(\theta-\theta_0)}} \right\}$$

where

θ = age to foil resonance

θ_0 = age to uranium resonance

X = distance perpendicular to column from column to point of measurement

Z_0 = perpendicular height from source line to column

b = distance between point sources

S_{θ} = relative strength of each age group of source neutrons

From a similar horizontal traverse of the thermal flux, the constant h can be determined.

Work is continuing on these measurements with different source and metal arrangements.

Another way proposed to measure h and r is to load one or several columns of metal with one type of foil; then repeat the measurement with another foil which has a resonance above the resonance energy of the metal. The only foil material found suitable for the second measurement is manganese. This experiment will be tried when a manganese foil becomes available.

Spatial and Energetic Dependence of Slowing-Down Density and Resonance Escape Probability

Using a laplace transform, the age equation may be transformed into an integral equation for the slowing-down density q as a function of neutron position r and

lethargy $u = \ln \frac{E_{\text{fission}}}{E_{\text{neutron}}}$ for a lattice work, provided that the (absorbing) fuel rods, j , are treated as line sources and sinks, for the purpose of giving flux patterns:

$$(1) \quad q(u, \rho) = \frac{\xi \sum_s}{4\pi D_0 u} \sum_{\text{all fuel rods } j} S_j e^{-\xi \sum_s |\vec{\rho} - \vec{\rho}_j|^2 / 4D_0 u} - \frac{V}{4\pi D_0} \sum_j \int_0^u \frac{e^{-\xi \sum_s |\vec{\rho} - \vec{\rho}_j|^2 / 4D_0 (u-u')}}{u-u'} q(u', \rho_j) \frac{\sum_a(u')}{F(u')} du'$$

where

D_0 = moderator diffusion coefficient

ξ = average logarithmic energy decrement

$\sum_s$ = moderator scattering cross section

$\sum_a(u)$ = fuel rod absorption cross section

$F(u)$ = fuel rod disadvantage factor

S_j = fission neutron source strength, j^{th} rod

V = cross-sectional area of fuel rod.

It can be shown that (1) may be reduced to the conventional formulas if it be assumed that the ratio of absorptive events to scattering events is small and that the ratio of the flux at the surface of the fuel rod to the average flux in the moderator is unity.

Quite simple solutions to (1) have been obtained in two of the simplest approximations. Let $D_0 u / \xi \sum_s = \tau(u)$

$$A. \quad \frac{\sum_a(u)}{F(u)} = \frac{\sum_{a0}}{F(u_0)} \int (u-u_0).$$

1. Single fuel rod in infinite moderator

$$q(\tau, \rho) = \frac{e^{-\rho^2/4\tau}}{4\pi\tau} - \frac{\sum_a(u_0)V}{4\pi D_0 u_0 F(u_0)} \frac{e^{-\rho^2/4\tau(u-u_0)}}{4\pi\tau(u-u_0)}$$

2. Infinite lattice work. If $S_j = J_0 \mathcal{H}(\rho_j)$,

$$q(\tau, \rho) = \left(1 - \frac{\sum_a(u_0)V}{\xi \sum_s F(u_0)b^2} \right) \frac{e^{-\mathcal{H}^2\tau}}{b^2} J_0(\mathcal{H}\rho) = p \frac{e^{-\mathcal{H}^2\tau}}{b^2} J_0(\mathcal{H}\rho),$$

p = resonance escape probability.

B. $\frac{\sum a(u)}{F(u)} = \frac{\sum a}{F}$, a constant, for $u \geq u_1 \geq u_{\text{fission}}$

1. Single fuel rod in infinite moderator

$$q(u, \rho) = \frac{e^{-\rho^2/4\tau}}{4\pi\tau} + \frac{\sum a^v}{16\pi^2\tau} e^{-\rho^2/4\tau} \text{Ei}\left(-\frac{\rho^2 u_1}{4\tau(u-u_1)}\right) + \left(\frac{\sum a^v}{4\pi D_0 F}\right)^2 \frac{1}{4\pi\tau} \text{Ei}(-\rho^2/4\tau) \text{Ei}\left(-\frac{a^2 u_1}{4\tau(u-u_1)}\right)$$

a = fuel rod radius

2. Infinite lattice-work. $S_j = J_0(\kappa \rho_j)$.

$$q(u, \rho) = e^{-\frac{4\pi\tau\sum a}{F b^2}} \frac{e^{-\kappa^2\tau}}{b^2} J_0(\kappa\rho) = p \frac{e^{-\kappa^2\tau}}{b^2} J_0(\kappa\rho).$$

For A2 and B2, the sums have been replaced by appropriate integrals. b^2 is the lattice area associated with a single fuel rod.

The principal limitation to application of the methods implied by (1) is the lack of knowledge of neutron cross sections in the resonance range, since transport techniques may, in principle, be used to calculate disadvantage factors quite accurately.

These calculations represent generalizations of work initially conceived and performed by W. A. Horning.

Prototype LTR

A. Introduction

For some time consideration has been given at Hanford to the measurement of lattice constants by critical experiments. Several methods have been proposed.⁽¹⁾ All of these involve determining the effects of a small piece of the lattice to be tested on the reactivity of an already critical driving assembly.

An early design for such a critical assembly is discussed briefly in HW-27354. Since the time at which this document was issued the magnitude of the reactor has been changed considerably. The LTR now has overall dimensions of 7' x 7' x 6' with a test cavity in its center about 2' x 2' x 3'. Dimensions of all mechanical

(1) Stewart, Hurwitz, LaViolette, Sampson, and Snyder - KAPL-331
W. A. Horning - HW-25107
W. J. Ozeroff - HW-26566

components have been reduced accordingly. At present, work is being directed toward the completion of a hazard report.

B. Notes on Small Source Theory

One of the proposed methods for interpreting data from the Lattice Testing Reactor is based on the concept of slug constants which are independent of lattice spacing in graphite piles. The proposal is to measure the slug constants in the LTR, and then calculate the buckling and other pile parameters which are dependent upon lattice spacing from these slug constants. The critical equation relating buckling (λ^2) to the slug constants can be written for dry lattices as: (2)

$$(1) \quad (\lambda^2 + \lambda^2 + \frac{1}{A}) e^{+\lambda^2 \tau} = \frac{C}{A}$$

where

$$A = \frac{2Db^2}{\pi a \beta} + \frac{b^2 K_0(\lambda a)}{2\pi} - \frac{D \lambda b^2 K_1(\lambda a)}{\pi} - \frac{1}{\lambda^2} (1 - \rho_1 \lambda K_1(\lambda \rho_1))$$

$$C = h \left(1 - \frac{2\pi a \gamma}{b^2} \right)$$

$\frac{1}{\lambda}$ is the graphite diffusion length

τ is the "age" to thermal in the pile

D is the graphite diffusion coefficient (1/3 of the transport mean free path for graphite)

b is the lattice spacing and $\pi \rho_1^2 = b^2$.

a is the radius of the process tube assembly

h, β , γ are the slug constants

h is called the slug multiplication and is the average number of fast neutrons produced per thermal neutron absorbed in the slug. ($h = \eta \epsilon$)

β is called the thermal blackness and is the probability that a thermal neutron entering the process assembly will be captured.

γ is proportional to the probability that neutrons of resonance energies will be captured after entering the process tube assembly.

H. Neumann has shown that the diffusion length of a pile in which the flux anisotropy is small, is given by: (3)

$$(2) \quad \frac{1}{L^2} = \lambda^2 + \frac{1}{A}$$

(2) HW-28646, Slug Constants for Inhomogeneous Piles, W. A. Horning (to be published)

(3) HW-29454, Diffusion Lengths in Heterogeneous Cylindrical Assemblies, H. Neumann

Substituting (2) into equation (1) gives

$$(3) \quad (\chi^2 L^2 + 1)e^{\chi^2 \tau} = \frac{CL^2}{A}$$

Comparing equation (3) with the usual age theory critical equation of

$$(\chi^2 L^2 + 1)e^{\chi^2 \tau} = k_{\infty}$$

shows that

$$(4) \quad k_{\infty} = \frac{CL^2}{A}$$

and replacing L^2 from equation (2) gives for k_{∞} in terms of slug constants:

$$(5) \quad k_{\infty} = \frac{C}{1 + \lambda^2 A}$$

Since $k_{\infty} = \frac{1}{f} \eta \epsilon p$ it is possible to pick these various factors out of equation (5). By definition $h = \eta \epsilon$. σ is related to the resonance capture so it might be expected that $p = (1 - \frac{2\pi a \sigma}{b^2})$ and this can be shown to be true for the

assumptions used in deriving σ ; namely, that resonance capture can be assumed to occur only at some single average energy.

Thus $C = \eta \epsilon p$ so that

$$(6) \quad \frac{1}{f} = \lambda^2 A + 1$$

showing that f is related to the thermal blackness and the cell dimensions. Solving equation (2) for A and combining with equation (6) gives:

$$L^2 = (1 - f) \frac{1}{\lambda^2}$$

which is the usual formula for the relation between the graphite diffusion length and the pile diffusion length.

H. Neumann has suggested that by using the results of the exponential experiments for dry lattice bucklings in equation (1), it might be possible to find slug constants for the standard Hanford slug. This was tried and using a value of $h = 1.34$ as calculated by the Exponential Pile group for their data, values of β and γ can be estimated from experimental values of χ^2 , giving:

$$\beta = 0.459$$

$$\gamma = 3.88 \text{ cm}$$

This value of blackness for the dry process tube assembly seems to be very reasonable when compared to some recent transport calculations.⁽⁴⁾ Using uranium cross sections given in BNL-170, they calculate for a Hanford dry process-tube assembly $\rho = 0.435$, and if more recent values of the cross sections are used this blackness value will be increased.

In this quarterly report, H. Neumann shows that γ can be related to the resonance integral by the relation

$$\gamma = \frac{(NV)_u}{2\pi a \sum_s} \int \frac{\sigma_a}{\bar{F}(E)} \frac{dE}{E}$$

where $\bar{F}(E)$ is related to the disadvantage factor for resonance neutrons, and when averaged over the resonance energies it can be taken as approximately unity. Then for a Hanford process assembly dry

$$\gamma = 5.21 \times 10^{23} \int \sigma_a \frac{dE}{E}$$

and using the value of $\gamma = 3.88$ cm

$$\int \sigma_a \frac{dE}{E} = 7.45 \text{ barns for dry process assembly}$$

G. W. Stuart has suggested a method for estimating the value of the resonance integral of the slug from this value as follows:

R_s = slug radius

a = process assembly radius

$$\left(\int \sigma_a \frac{dE}{E} \right)_{\text{process assembly}} = \left(\int \sigma_a \frac{dE}{E} \right)_{\text{slug}} \frac{\int_0^\theta \cos \phi d\phi}{\int_0^{\pi/2} \cos \phi d\phi}$$

$$\text{or } \left(\int \sigma_a \frac{dE}{E} \right)_{\text{process assembly}} = \left(\int \sigma_a \frac{dE}{E} \right)_{\text{slug}} \frac{R_s}{a} ; \quad \frac{R_s}{a} = .79$$

(4) HW-29992, Thermal Blackness of a Solid Cylindrical Slug, R. W. Woodruff and G. W. Stuart

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HW-30508

hence

$$\frac{\left(\int \sigma_a \frac{dE}{E} \right) \text{ slug}}{\int \frac{dE}{E}} = \frac{7.45}{(0.79)(5.6)} = 1.68 \text{ barns}$$

This agrees with the value calculated by the Exponential Pile group of 1.676 barns.

In the solution of the "small-source theory" differential equation which leads to equation (1), it is possible to simplify the mathematics by replacing the slowing-down Kernel as follows:

$$\frac{e^{-|\vec{p}-\vec{p}_1|^2/4\tau}}{4\pi\tau} = AK_0(B|\vec{p}-\vec{p}_1|)$$

where A and B are found by fitting the area and the second moment.

Then

$$A = \frac{1}{2\pi\tau} \quad B = \frac{1}{\sqrt{\tau}}$$

When this is done then the critical equation becomes

$$(\kappa^2 + \lambda^2 + \frac{1}{A})(1 + \kappa^2\tau) = \frac{C}{A}$$

which can be recognized as the two-group critical equation. This equation can be solved for κ^2 using the values of slug constants obtained above. Figure 2 shows a plot of buckling vs. lattice spacing obtained from this equation. It can be seen that the curve is in quite good agreement with the experimental data.

To summarize:

- 1) Using a combination of small-source theory and exponential experimental results, values for the slug constants of a Hanford slug can be obtained.
- 2) The values of these slug constants seem quite reasonable when compared with independent information.
- 3) These slug constants, when combined with the two-group critical equation yield a curve of buckling vs. lattice spacing for the Hanford slug which is in good agreement with experiment.

Although this is admittedly a rather circular process, it would seem to be of some significance that the experimental curve of buckling vs. lattice spacing can be fitted with three parameters, and that these parameters are simply the slug constants defined by the small-source theory of W. A. Horning.

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FIGURE 2

C. A Weighting Theorem for the LTR

In order to discuss safety aspects of the LTR and to plan experiments which can be done with it, it is useful to be able to assign a value to the reproduction factor of the reactor under varying experimental conditions. However, since the reactor is composed of several completely different regions, the value of such a quantity is not immediately obvious even when the properties of the individual regions are well known. In order to relate the reactivity, k' , of a test core placed in the test cavity of the reactor to the pile reactivity and therefore to the period of the pile as a whole, the following weighting theorem is used:

$$(1) \quad k_p = \frac{N \rho h J}{N \rho h J} + \frac{N' \rho' h' J'}{N' \rho' h' J'} + C J''$$

where

- N = number of driver rods
- ρ = blackness of driver rods
- h = multiplication of driver rods
- J = current into the driver rods
- k = reproduction factor of driver rod region

N', ρ', h', J' and k' are the corresponding quantities in the core region.

$C J''$ is a term added to account for capture in and leakage from the reflector region.

From the definitions of h and ρ as given in Part B of this report, it can be seen that according to (1)

$$k_p = \frac{\text{number of neutrons in generation } n+1}{\text{number of neutrons in generation } n}$$

which is in accord with the usual definition of the reproduction factor.

The ratios J'/J and J''/J are assumed to be constant for any k' . It has been shown that for a graphite core with $k' = 0$, $(J'/J) = 1.85$ and varies quite slowly as k' is changed. Also, for given k' values, the product $\rho' h'$ can be obtained from diffusion theory as described in Part B above. N, ρ, h, k, k' , and N' are fixed by the design of the LTR and the particular core to be tested. Thus, for given k' , only the constant $C J''/J$ is unknown. This constant can be obtained by setting $k_p = 1$ for $k' = 0$, a condition which will be established in the initial loading of the reactor.

Figure 3 shows a plot of k_p vs. the buckling of the test core for 3 x 3 lattice unit cores. In calculating the values on this curve, it was assumed that the core contained fuel only in the form of an enriched uranium-aluminum alloy. With this type of loading, p , the resonance escape probability is unity and h is approximately constant. If it is further assumed that two cores of the same buckling, one containing enriched uranium-aluminum alloy and the other containing natural uranium, have the same overall effect on the pile, the values of k_p on this curve are the ones which might be encountered in experiments done with the LTR. All experiments which are at present anticipated involve cores whose bucklings are less than 200 microbucks, but even with these, excess reactivities of the order of 5% seem possible. An original form of this weighting theorem was suggested by Dr. W. A. Horning.

D. Control Rod Strength

The control rods to be used in the LTR have been designed to preserve the cylindrical symmetry of the reactor and to effect the longitudinal flux uniformly. This design consists of a tube the length of the core which is fixed in the pile and is made essentially of four-inch sections alternately aluminum and cadmium. Another tube slides inside the fixed tube and contains alternately cadmium and enriched uranium slugs. The concentrations of the cadmium and uranium are set so that the entire rod is always "black" to thermal neutrons. In the "closed" position the rod becomes a complete cadmium rod, and in the "open" position, the rod becomes alternate four-inch sections of cadmium and U-235 which returns about two fast neutrons to the pile per neutron absorbed. Eight such rods are then placed symmetrically around the core.

In order to calculate the strength of these control rods it is necessary to study the neutron interaction between the core being tested, the enriched driver rods, and the control rods.

Since provisions are to be made for the ends of the pile to level the longitudinal flux, the calculations can be made considering the radial flux only with the following picture:

FIG. 3

CORE BUCKLING VS PILE REACTIVITY

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The small-source theory differential equation for the flux in such a pile can be written as:

$$(1) \quad D(\nabla^2 - \lambda^2)\phi = \sum_i Q_i \delta(|\vec{\rho} - \vec{\rho}_i|) - \sum_i \frac{h_i Q_i e^{-|\vec{\rho} - \vec{\rho}_i|/\tau}}{4\pi\tau}$$

where the symbols have been described in Part B. Q_1 is called the sink density and is the total number of neutrons absorbed per unit length per second. Using the "two-group" approximation as mentioned in Part B where the gaussian is replaced by a K_0 function, the following solution is found for the flux as a function of $\vec{\rho}$:

$$(2) \quad \phi = - \frac{Q_1}{2\pi D} \sum_i^{\text{drivers}} K_0(\lambda|\vec{\rho} - \vec{R}_i|) + \frac{h_1 Q_1}{2\pi D(1 + \tau\lambda^2)} \sum_i^{\text{drivers}} \left\{ K_0(\lambda|\vec{\rho} - \vec{R}_i|) - K_0\left(\frac{1}{\tau}|\vec{\rho} - \vec{R}_i|\right) \right\} \\ - \frac{Q_3}{2\pi D} \sum_i^{\text{Rods}} K_0(\lambda|\vec{\rho} - \vec{R}_{3i}|) + \frac{h_3 Q_3}{2\pi D(1 + \tau\lambda^2)} \sum_i^{\text{Rods}} \left\{ K_0(\lambda|\vec{\rho} - \vec{R}_{3i}|) - K_0\left(\frac{1}{\tau}|\vec{\rho} - \vec{R}_{3i}|\right) \right\} \\ - \frac{Q_2 R_2}{D} K_0(\lambda R_2) I_0(\lambda \rho) - \frac{Q_c}{2\pi D} \int_0^{R_c} \int_0^{2\pi} J_0(\lambda \rho') K_0(\lambda|\vec{\rho} - \vec{\rho}'|) \frac{\rho' d\rho' d\phi}{b_c^2} \\ + \frac{h_c Q_c}{2\pi D(1 + \tau\lambda^2)} \int_0^{R_c} \int_0^{2\pi} J_0(\lambda \rho') \left\{ K_0(\lambda|\vec{\rho} - \vec{\rho}'|) - K_0\left(\frac{1}{\tau}|\vec{\rho} - \vec{\rho}'|\right) \right\} \frac{\rho' d\rho' d\phi}{b_c^2}$$

In this equation the fast leakage from the edge of the reflector has been assumed to be negligible and the sums over the core rods have been replaced by integrals assuming the sink density can be written as $[Q_c J_0(\lambda \rho)]$ with Q_c the sink function of the center rod. Then the Q_i 's are determined by the bounding condition

$$(3) \quad Q_1 = \beta_1 \oint J_{1n} d\rho' d\phi$$

at the surface of the rods. Evaluating the Q_i terms give four homogeneous equations the determinate of which is the critical condition for the reactor. In this case, first the core is assumed to be graphite only ($Q_c=0$). The control rods are taken to be in the "open" position, which for LTR gives $h_3 = .889$, and then the critical condition is found which determines the U-235 concentration needed in the driver rods. With h and β thus determined for the driver rods various values of the buckling, λ^2 , of the test core can be used in the equation and new critical conditions found with h_3 left as the only unknown. Figure 4 shows a plot of h_3 versus the buckling of the core assuming that the core rods are composed of enriched material with no resonance capture. This plot was made for various values of the control rod radius (R_3) to find that radius which would

FIG 4

BUCKLING OF CORE REGION
VS.
CONTROL ROD MULTIPLICATION

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give the desired range of control without an undesirable loss in sensitivity. Since $h_3 = 0$ is the maximum strength of the control rods, a change of h_3 from .889 to 0 corresponds to the full four-inch motion of the control rods from open to closed. For this calculation, the core region is taken to have a radius of $R_c = 36$ cm, the driver rods are on a radius of $R_1 = 51.6$ cm and the extrapolated edge of the pile has an equivalent radius of $R_3 = 125.9$ cm.

E. Strength of the Safeties

The safety control for the LTR consists of two main features. These are: (1) the control rods, which are designed with a magnetic clutch and spring loading such that they can be quickly snapped to the "closed" position when the pile scrams, (2) the main safety, which consists of four two-foot circular discs made of cadmium sandwiched between two steel plates. These discs are held by electromagnets in vertical slots above the reactor. When a scram occurs the discs fall to a position just outside of the driver rods, thus decreasing the effectiveness of the reflector. In order to calculate the control strength of these discs, it is convenient to take an equivalent ring completely surrounding the driver rods with an average blackness which is found by weighting the actual blackness of the discs with the ratio of the surface area of the discs to the surface area of the equivalent cylinder. This ring can then be put in as another term in equation (2), Part D. The equivalent blackness of this ring for the LTR is estimated to be .176. Then making calculations as discussed in Part D, using various values of the core buckling (k^2) it was found that with the control rods "open" and the discs in the reactor, it could not be made critical until a core of enriched material having a buckling of about $1000 (10)^{-6} \text{ cm}^{-2}$ was inserted into the reactor.

A conservative estimate of the strength of the safety discs has been made by G. M. Muller. He found that the effect of the discs caused a $900 (10)^{-6} \text{ cm}^{-2}$ change in the buckling of the driver region. Using the weighting formula discussed in Part C, it is possible to estimate what this corresponds to in terms of the core buckling. This calculation indicates that the discs could control a core with a buckling of $800 (10)^{-6} \text{ cm}^{-2}$. It can be shown that this is a conservative estimate and hence the two calculations are in good agreement. These calculations show that the safety discs will be strong enough to make the pile subcritical for any presently foreseen condition of the reactor.

We are indebted to Mrs. Virginia Clark, who performed many of the calculations necessary in determining the strength of the control rods and safety discs.

F. Reactivity Loss as a Result of Fuel Melt

If all the mechanical safety mechanisms should fail at a time when there is excess reactivity available in the reactor, the power level of the reactor will climb until the fuel in the reactor melts. At this time the fuel, if it is arranged in an advantageous geometry, will assume a less effective position and the reactor will lose reactivity, perhaps enough to become subcritical.

In the LTR, the fuel in the driver rods will be fabricated in the form of hollow cylinders, whose inside diameter, .96 inches, $= \frac{1}{\sqrt{2}}$ outside diameter,

1.36". At the time the fuel melts, therefore, it will assume the shape of a

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cylinder whose cross-sectional area is a semi-circle, hereafter referred to as a collapsed cylinder. It is of interest to know how much reactivity is lost when this transformation takes place. To determine this quantity, an experiment was performed by W. P. Stinson. Mixtures of MgO and various concentrations of amorphous boron were packed in aluminum containers of the desired shapes. The density of the MgO, as packed in the sample holders, was such that its scattering cross section was about 0.94 of that of the same volume of aluminum. The effects of these assemblies on the reactivity of the Hanford Test Pile have been measured. The results of these measurements are summarized in Figure 5. Curve 1 shows the loss in reactivity of the Hanford Test Pile when a hollow cylinder of the dimensions mentioned above is placed in one of its columns, vs. the concentration of absorber in the cylinder, and curve 2 shows the same quantity for the collapsed cylinder. Curve 3 is a plot of Σ_a vs. boron density.

In order to make this information more meaningful in the particular case of interest, it is useful to discuss the experimental results in terms of another parameter, namely, the blackness, ρ , of the tube containing the absorbing material. This blackness is the probability that a thermal neutron which crosses the surface of the tube containing the absorbing material will be absorbed. It is important to note that the shape of this tube does not change when the geometry of the absorbing material changes from a hollow to a collapsed cylinder. For this reason, it can be seen that the blackness of the tube will decrease when the absorbing material melts.

The blackness of this tube, when it contains a hollow cylinder, can be calculated as a function of the density of the absorbing material by diffusion theory. Calculated blackness vs. absorber density are shown in curve 1, Figure 6. The blackness of the tube, when it contains a collapsed cylinder of absorbing material, however, is quite difficult to calculate, and it is this quantity which is obtained from the experiment. This quantity is deduced from the experimental results as follows:

- a) Assume that two tubes which have the same effect on the reactivity of the Hanford Test Pile have the same blackness, ρ , regardless of the geometrical shape of the absorber inside the tubes.
- b) Then a line of constant reactivity, such as that shown at 200 in Figure 5, is also a line of constant blackness.
- c) The blackness of the tube containing a hollow cylinder is known for this line from the diffusion calculation.
- d) Therefore, the blackness of the tube containing the collapsed cylinder is also known, and from the intersection of this line of constant blackness with curve 2, Figure 5, a value of the density of absorber in the collapsed cylinder corresponding to this blackness can be obtained from curve 3, Figure 5.
- e) Repeating this process for several values of constant reactivity, curve 2, Figure 6, can be plotted, giving the blackness of the tube containing the collapsed cylinder vs. absorber density.

Thus, for a hollow cylinder of $\rho = 0.3$, which is approximately the value needed in the LTR, $\Delta \rho$ upon melting is approximately 0.027.

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FIG. 6
BLACKNESS VS. TOTAL ABSORPTION

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then

$$\frac{\Delta \rho}{\rho} = 9\%$$

Finally, using the weighting theorem, discussed in Part C of this report, with a graphite core, and holding all parameters constant except the blackness of the driver rods and the reactivity of the driver region, k , this $\frac{\Delta \rho}{\rho}$ gives a $\frac{\Delta k}{k}$ for the reactor of 4%.

No effort has been made to estimate the uncertainties of this number. It is felt that the largest error probably results from the inaccuracy with which the blackness of the tube containing a hollow cylinder is given by diffusion theory. This value is probably off by at least 10%. However, this error will contribute a somewhat smaller uncertainty in the final value of $\Delta \rho$.

An error in $\frac{\Delta k}{k}$ will also result from uncertainties in the weighting theorem used, but it does not seem possible to evaluate the magnitude of this error at present.

Thermal Column in C Pile Shielding Facility

The possibility of building a thermal column into the 36-inch shield plug facility on the X-2 level at C Pile has been investigated. Briefly, such a column would be constructed by removing the shielding plug from the biological shield and replacing this with graphite and additional lead shielding to screen out pile gamma rays. The existing 6-inch lead curtain would then be lowered over the face of the column and an exponential-type graphite pile erected in front of this.

Preliminary calculations, using one-group diffusion theory, indicate a thermal neutron flux of about 10^6 n/cm² sec. would be available to a test lattice in a unit of approximate dimensions 5' x 5' by 6' long. This is an increase in flux by nearly a factor of one hundred over the flux currently available in existing exponential piles at this site.

Measurements which one might make on a test lattice in the described facility are as follows: (1) diffusion length, (2) thermal utilization, (3) conversion ratios and (4) blackness. The problem of making buckling measurements in such a unit was discussed with D. E. Davenport of the Exponential Physics Sub-Unit. It appears that little can be gained over the buckling measurements obtained in the present exponential piles. One of the reasons for this is that the savings in time resulting from the higher flux would just about be offset by corrections one may have to make due to neutrons being scattered by the concrete floor which the pile must rest on. However, cell measurements may be made without this concern.

One of the disadvantages of this facility would be the large amount of gamma shielding required to reduce the pile gamma rays to tolerance limits. It appears that a minimum of six inches of additional Pb will be required.

A detailed report of this study will be issued shortly.

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Relative Conversion Efficiency Determination

Work has been done on development and evaluation of a method for experimentally determining the U-235 to Pu-239 conversion efficiency of a test lattice relative to the standard Hanford Lattice. The measurement is made as follows:

Natural and depleted uranium foils are irradiated in slugs in the standard lattice and in the test lattice. The relative plutonium production for the two lattices is then determined by counting the 100 kilovolt plutonium X-ray produced in the decay of 2.3 day neptunium-239, and relative fission rates are determined by counting the fission product activities. The conversion efficiency of the Hanford production reactors (determined by mass spectrometric and radiochemical methods, and extrapolated to zero exposure) when combined with foil measurements will give an absolute value for the conversion efficiency of the test lattice.

Preliminary foil exposures indicate that a difference in conversion efficiency of 3% between the two lattices may be detected without difficulty.

NUCLEAR PHYSICS

Neutron Spectrometer Facilities

A. The Ratio of Pu-239 to U-235 Fission Cross Sections

The Hanford crystal neutron spectrometer facilities are presently being used in a program for investigating the slow neutron properties of the fissionable isotopes important for a knowledge of fundamental reactor theory. The first of these investigations has been that of the variation with energy of the ratio of Pu-239 to U-235 fission cross sections. This measurement was made from 0.020 e.v. to 1.0 e.v. using plutonium and uranium foils mounted back-to-back in a dual ionization fission chamber. The fission chamber, counting equipment, and the fissile foils were prepared by the Experimental Physics Unit at Knolls Atomic Power Laboratory. The results of this experiment are shown in Figure 7 together with a curve obtained from the presently accepted values of the fission cross sections. The curve of the experimental data in Figure 7 has been normalized by an indirect analysis assuming that the deviations observed from presently accepted values are due to the Pu-239 cross section. This analysis yields a value of 796 barns for σ_f^{49} at 2200/m/sec. with an f value of 0.995. This experiment also yields the energy variation of the ratio of $(1 + \alpha)$ for Pu-239 to U-235 from

$$\frac{(1 + \alpha)^{49}}{(1 + \alpha)^{25}} = \frac{\sigma_a^{49}}{\sigma_a^{25}} \times \frac{\sigma_f^{25}}{\sigma_f^{49}}$$

The ratio of $(1 + \alpha)$'s is shown in Figure 8 with the normalization of $\sigma_f^{49}/\sigma_f^{25}$ previously discussed. The data of Figure 8 as shown are fit to a straight line given by

$$\frac{(1 + \alpha)^{49}}{(1 + \alpha)^{25}} = 0.82 E(\text{ev}) + 1.109$$

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FIGURE 8

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The results of this investigation have been published in HW-30128. (5) Weight assignments will be made for the foils used in this experiment by comparison with standard foils now in preparation at KAPL. A standard error of about ± 1 per cent for the absolute value of the ratio of $\sigma_f^{49}/\sigma_f^{25}$ seems likely at this time.

B. The Energy Variation of Alpha of Uranium 235

A chamber which uses a thin boron foil for a standard beam monitor is being built by F. A. White at KAPL and will be used to measure the energy variation of $(1 + \lambda)^{25}$ in an experiment to be undertaken shortly. In this experiment, the total cross section is obtained from fission rates of uranium foils on both sides of a uranium transmission sample while the variation with energy of the uranium fission cross section is obtained by comparison with the boron foil.

C. Fission Cross Sections of Higher Isotopes of Plutonium

Two samples of plutonium of high enrichment in Pu-240 and Pu-241 have been separated by the Radiochemistry Sub-Unit at Hanford. Fission foils of these materials are to be prepared by electro deposition techniques. These two foils will be compared with a foil of low Pu-240 (about 4.6 per cent) concentration in order to make a determination of the energy variations of the fission cross sections of Pu-240 and Pu-241. The difficulties attendant with the higher alpha activity of these foils in the ionization fission chambers remain to be overcome.

D. Monoenergetic Foil Activations

The large monoenergetic fluxes available with the spectrometer, as set up for the measurement of fission cross sections, have enabled the use of the facility for activation measurements in a monoenergetic neutron beam. The facility was used in connection with the calibration of gold and gold-cadmium foils for neutron flux spectra measurements. In this irradiation, a one-inch diameter gold foil 0.002 inches thick and an accompanying background measuring foil were irradiated from about 25 hours in a 0.10 ev neutron beam of about 4.74% energy resolution. The activated Au foil yielded a counting rate of about 7500 c/m for the Au 0.411 Mev photopeak with about 10 per cent resolution. Less than 4 per cent of this counting rate was due to background activation although neither foil was shielded from the room background during the irradiation. The amount of second-order contamination in the beam at 0.10 ev was calculated to be 1.74 ± 0.1 per cent. Similar activations can be made at slightly higher energies where the second-order composition is less and where suitable selective absorbers can be used. Further possibilities of using this technique will be explored.

(5) Leonard, B.R., Hauser, S. M., Seppi, E. J., The Ratio of Plutonium-239 to Uranium-235 Fission Cross Sections from 0.020 to 1.0 Electron Volts, HW-30128, Dec. 1, 1953

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Neutron Flux Spectra Studies

Thermal Neutron Temperatures

As the first step in a program aimed at more detailed knowledge of the flux spectra in cold, clean piles and in hot production piles, a method of obtaining the effective temperature of the pile thermal neutrons inside reactive assemblies is being sought.

A most promising approach results from a new application of the old method of using gold and cadmium, whose cross sections are large and very different functions of energy in the thermal region. Rather than deducing the temperature from gold and cadmium danger coefficients, a thick foil of gold-cadmium alloy (about 80 mils for an alloy of 15 atomic per cent cadmium) is prepared and irradiated with its top and bottom surfaces covered with 1 mil foils of pure gold. The probability that a thermal neutron which passes through the thin gold and enters the alloy foil is not absorbed in the foil is extremely small, due to both the large macroscopic absorption cross section of the alloy and the foil thickness. Therefore, the probability that the neutron energy, E , entering the foil is captured in gold rather than cadmium is just

$$\frac{N_{\text{Au}} \sigma_{\text{Au}}(E)}{N_{\text{Au}} \sigma_{\text{Au}}(E) + N_{\text{Cd}} \sigma_{\text{Cd}}(E)}$$

where N is the number of nuclei per cm^3 and $\sigma(E)$ is the neutron absorption cross section. Since this function decreases drastically with increasing energy in the thermal region, the gold activity in the alloy is very sensitively dependent upon the relative numbers of low energy neutrons. It is the function of the thin gold to measure the flux of neutrons across the surface of the alloy foil. Then, the ratio of the gold activity in the alloy to that in the thin foil is taken as a measure of the effective temperature of the neutron distribution, assumed to be Maxwellian in form. A preliminary calculation has shown that the sensitivity is such that this ratio decreases by almost a factor of 2 for a change in the temperature of a Maxwellian distribution from 300° to 700° K.

This method has several important advantages over other methods, which have been used at other laboratories: 1) it is as easily used in a beam geometry as in an isotropic flux; and with only small corrections (depending upon the thickness of the thin gold which must measure flux) it can be used in a nonisotropic flux, 2) depression of the thermal flux introduces no corrections, 3) there are no large transmission factors (dependent upon angular distribution) to be computed since the foil is totally black to thermal neutrons, 4) it can be employed anywhere that the neutron flux is intense enough to activate the gold sufficiently, as determined by the desired accuracy and counting techniques.

An alloy and thick foils as described above have already been prepared and subjected to irradiation tests and chemical analyses. An irradiation in the E-hole at F-Pile has shown that the alloy foils are homogeneous to at least 1%, the average deviation of the normalized relative counting rates from the mean. The deviations as found, however, seem to be more a function of the irradiation history than position in the original sheet of metal. Since the chemical analyses indicate that selective oxidation of the cadmium during fabrication has lowered the percentage of this metal from that put into the melt and since such analyses

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can not easily give the composition of the alloy to better than about 5%, a mono-energetic irradiation has been performed in the 0.10 ev spectrometer beam at DR-Pile. Though instrumental difficulties invalidated the results, the high counting rates observed ensures us that this method of calibrating the foils is extremely promising. The foils are gamma counted with a 2" x 2" cylindrical NaI crystal and photomultiplier, recording only those pulses which lie in the Au photopeak, centered at 0.411 Mev and having a full width at half-maximum of 0.040 Mev for the crystal and photomultiplier used. The counts in an energy range of about 0.080 Mev which are attributable to the Cd activity are less than 0.1% of the total counts from the Au activity.

The measured linear absorption coefficient of the Au γ ray in the alloy is $2.85 \pm 0.10 \text{ cm}^{-1}$ compared to 3.3 cm^{-1} for pure gold. By using two foils of 39 mils thickness each and counting with the exposed surface toward the crystal, the correction for gamma absorption in the foils is about 8%, which can easily be calculated to within 10%. To reduce edge effects, rings of the alloy are being machined to surround and hold the two thick foils in position. The activity induced by epi-Cd neutrons will be subtracted by measuring that induced in an identical cadmium-covered foil assembly.

C¹² Absorption Cross Section Measurement

A measurement of the absorption cross section of C¹² was initiated several years ago to obtain information required to evaluate the desirability of further purifying the graphite used in graphite moderated reactors. The buildup of C¹³, resulting from the neutron irradiation of a carbon sample enriched in C¹², is to be measured by Dr. M. G. Inghram at ANL. Suitable monitors measure the integrated neutron exposure.

Irradiation of the two carbon samples together with a cobalt monitor was completed on October 7 in the Materials Testing Reactor. The accumulated exposure of these samples totaled 9,294 MWD, equivalent to an integrated thermal neutron flux of about 1.35×10^{22} neutrons/cm² in the L-55 facility. Return of the samples to this site from MTR was completed on November 17.

Mr. Roy Ko of the Chemistry Unit has completed extraction of lithium iodide from the LiI-charcoal mixture which was irradiated with the first cobalt monitor. About 0.6 grams from the original amount of LiI was recovered, which should be more than sufficient for a mass spectrometric determination of the Li⁶ to Li⁷ ratio. As soon as new laboratory space is occupied, the C¹² capsules will be removed from their container and sent together with the LiI to Dr. Inghram. The cobalt flux monitor will be dissolved and counted here in the same manner as the first cobalt sample which was associated with the LiI. This work is reported in HW-28283.

A Multichannel Pulse Height Analyzer

Development has been started on a new type of multichannel pulse height analyzer. In principal, this analyzer converts a pulse height spectrum into a frequency spectrum. Band pass filters then provide the channel placement and channel width. In practice, the pulses, suitably shaped to have a flat top

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about four microseconds long, frequency modulate an oscillator to provide a frequency shift that is a function of the pulse height. The output of this oscillator will feed into a group of band pass filters. The output of each filter is detected and the largest pulse between two adjacent filters is taken by a simple circuit, and fed to suitable register circuits.

The advantages of this circuit lie in the stability of channel position and width, the fast resolving time, the simple calibration equipment and required procedures, and the absence of critical discriminators.

Development has been centered primarily on what appears to be the only major obstacle-- oscillator base-line statistics. Current thought indicates that the oscillator frequency should shift from about 8 megacycles to 16 megacycles or more with a base-line stability of at least 0.1%. While this stability is reasonable with conventional mechanically tuned oscillators, it becomes more difficult to tune electrically with the three requirements of: frequency range of at least 2x, complete frequency shift in less than 1 microsecond, and base-line stability of 0.1% at 10 megacycles. So far, it has been easy to obtain any two of the three requirements, but not the third.

It would be possible to build a frequency correction circuit which would supply the third requirement for a fast shifting, wide range oscillator, but it is felt that the added complication is to be avoided if possible.

A patent application has been submitted on this work.

Photomultiplier Power Supply

Recently a new arrangement was conceived, but not developed, for a high voltage power supply stabilizing unit suitable for photomultipliers. Normally, the reference element is a discharge tube of some sort, or a small battery, often a standard cell. However, discharge tubes tend to drift and inherently have discontinuous "jumps". Standard cells can be made quite stable, but the subsequent error signal requires relatively large DC amplification.

In the new arrangement, a photomultiplier tube, which is powered by the supply to be regulated has at its photo-cathode a constant light source. This source is composed of phosphor and a little radium, or other long half-life radioactive material. The output of this multiplier tube may be fed directly to the grid of the regulating tube as the error signal. The multiplier provides a stable error gain of about 10, which is 10 x 1000 larger than a standard cell error signal. The regulating tube can be a rugged, low gain triode, providing an additional stability margin.

The advantages seem to be high stability for a period of days at least, and simple design.

One disadvantage is the requirement that the multiplier tube must be shielded from changing electric and magnetic fields.

Work has been suspended temporarily to allow development of the multichannel pulse height analyzer.

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IRRADIATION PHYSICS

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Radiation Chemistry

A program for the study of the fundamental processes involved in the radiation induced reactions between graphite and carbon dioxide has been formulated. This program appears to be feasible using the mass spectrometric facilities available at Hanford, and consists of the following experiments:

1. A graphite sample, in high vacuum, is irradiated with electrons accelerated to a controlled low (under 1000 ev) energy and the ionic species formed (together with their appearance potentials) are observed.
2. Ionic reactions between the graphite surface and ions formed in the gas phase (at pressures of about 10^{-3} mm) are observed by analyzing the ionic species formed when a collimated beam of electrons of controlled low energy passes through the gas close to (but not impinging upon or grazing) the graphite surface.
3. Reactions due to ionization of gas adsorbed on the graphite surface are observed by analyzing the ionic species formed when a beam of electrons of controlled low energy impinges on the graphite surface in a carbon dioxide atmosphere.

With the use of isotopically enriched graphite, it is possible to determine directly whether an ionic species originated from the gas molecules (either in the gas phase or adsorbed on the surface) or from reaction of the graphite itself.

It is possible, in principle, to study the non-ionic reactions (due to dissociation or excitation) by repeating (2) and (3), using electrons of energy below the appearance potentials of the ionic species but above the dissociation energy of the gas. (A monochromatic light source of suitable energy could be used instead). A second, isolated, electron beam would be required to analyze the products in the mass spectrometer. The practicability of this scheme is being studied.

PLANT PHYSICSOptimum Flattening

The excess reactivity of Hanford reactors is balanced by full poison columns, which produce a large radial flattening of the flux, and the control rods, which have a longitudinal effect. The flat zone can therefore be approximated by a cylinder of large radius and small altitude. Since the dimensions of this cylindrical flat zone can be varied by the use of columns of mixed fuel and poison slugs, it is of interest to calculate the optimum flat zone dimensions; thus a one-group diffusional treatment was applied to a cylindrical homogeneous model of an H-size reactor. It was found, by integrating the flux over the reactor volume, that Hanford reactors best utilize their fuel when operating with little or no longitudinal flattening; however, if future reactors

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have appreciably greater material buckling, significant gains can be made through careful flattening, longitudinally as well as radially. A more detailed discussion is presented in HW-29853.

Critical Buckling of a Cube with Infinite Reflector

Practical considerations generally require that critical experiments with reflectors be done in a geometry other than spherical. If the dimensions of the core are not appreciably larger than the extrapolation length, the determination of the material buckling of a, say, cubical core from a critical experiment becomes difficult. In order to shed some light on this type of situation, we present here the result of a one-group calculation which gives an upper limit to the ratio of the critical buckling of a cube to that of a sphere of the same volume, for a given ratio of the diffusion length in the reflector to the length of the cube (assuming that the diffusion coefficients in the core and reflector are equal).

Using the notation:

K^2 = critical buckling of cube

c = half-length of cube

λ = inverse diffusion length in reflector,

the mathematical problem is as follows. Find the lowest eigenvalue K^2 of the boundary value problem

$$\nabla^2 u + K^2 u = 0 \quad \text{cube}$$

$$\nabla^2 u - \lambda^2 u = 0 \quad \text{reflector,}$$

$u(\vec{r})$ to be continuous, along with its normal derivative, at the cube-reflector interface; $u(\infty) = 0$.

By a straight-forward application of Green's theorem, this problem may be formulated as an equivalent integral equation

$$4\pi u(\vec{r}) = (K^2 + \lambda^2) \int_{\text{cube}} \frac{\exp(-\lambda |\vec{r} - \vec{r}'|)}{|\vec{r} - \vec{r}'|} u(\vec{r}') d\vec{r}'$$

If $\vec{r} = (\xi, \eta, \zeta)$, then by introducing new independent variables $x = \xi/c$, $y = \eta/c$, $z = \zeta/c$ and making use of the identity

$$\frac{\exp(-\alpha \rho)}{\rho} = \frac{2}{\sqrt{\pi}} \int_0^\infty \exp(-\rho^2 t^2 - \frac{\alpha^2}{4t^2}) dt$$

the integral equation becomes (inverting the order of integration)

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$$4\pi u(x,y,z) = c^2(K^2 + \lambda^2) \frac{2}{\sqrt{\pi}} \int_0^\infty \exp\left(-\frac{c^2 \lambda^2}{4t^2}\right) dt \int_{-1}^1 \int_{-1}^1 \int_{-1}^1 \left\{ \exp\left[-(x-x')^2 - (y-y')^2 - (z-z')^2\right] u(x',y',z') \right\} dx' dy' dz'.$$

The variational method has been applied to the integral equation in this form by forming the expression

$$\sigma = \frac{4\pi \int_{-1}^1 \int_{-1}^1 \int_{-1}^1 v^2(x,y,z) dx dy dz}{\frac{2}{\sqrt{\pi}} \int_0^\infty \exp\left(-\frac{c^2 \lambda^2}{4t^2}\right) dt \int_{-1}^1 \int_{-1}^1 \int_{-1}^1 \int_{-1}^1 \int_{-1}^1 \int_{-1}^1 \exp\left[-(x-x')^2 - (y-y')^2 - (z-z')^2\right] v(x,y,z) v(x',y',z') dx dy dz dx' dy' dz'}$$

with $v(x,y,z) = 1 - a(x^2 + y^2 + z^2)$, and then choosing a so as to make σ a minimum. σ gives an upper bound for and approximation to the true minimum value $c^2(K^2 + \lambda^2)$ for which the integral equation has a solution; thus we obtain an upper bound for and approximation to the critical buckling of the cube,

$$K^2 \approx \frac{\sigma - c^2 \lambda^2}{c^2}.$$

The form of σ as given above has the advantage that all the integrations except that with respect to t may be done analytically.

The results of this calculation are given in dimensionless form in the attached graph (Figure 9). It is seen that for large values of $c\lambda$ (the half-length of the cube in units of the diffusion length in the reflector) the variational trial function used is inadequate; the true value of $K_{\text{cube}}^2/K_{\text{sphere}}^2$ should asymptotically approach 1.155, the ratio of the critical buckling of a bare cube to that of a bare sphere of the same volume. For values of $c\lambda$ below about 3, however, it is felt that the present calculation provides at least a useful upper limit. A more detailed discussion, both of the details of the calculation and of the results will be given in a formal report.

The author wishes to acknowledge the contributions made to the early stages of this problem by Dr. W. A. Horning and Mr. P. M. Anselone, and the valuable aid of Mr. D. D. McCracken who programmed the numerical integrations for the Hanford IBM equipment.

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FIGURE 9

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Conversion Ratio for P-10 Production in DR-10

Because of the large discrepancy (~20%) between observed and calculated tritium production in the H-10 loading, considerable uncertainty exists as to what conversion ratio should be used for accountability purposes in the current DR-10 program. The conversion ratio has been derived from diffusion theory and is given by:

$$C = \frac{1 + S_{25}}{1 + S_L} \left(\frac{\eta f}{1 + S_{25}} - 1 \right)$$

where

$$S_{25} = \frac{\text{Parasitic Absorption in J Slug}}{\text{Absorption in 25}}$$

$$S_L = \frac{\text{Parasitic Absorption in Z Slug}}{\text{Absorption in Li}}$$

$$\eta = \frac{\text{Number of Neutrons Released}}{\text{Thermal Neutron Capture in 25}}$$

$$f = \text{Equivalent thermal utilization for the lattice}$$

This has been recalculated using latest cross-section data and slug constants and was found to be 0.78 which is about 6% lower than the previous calculated value, reducing the discrepancy from 20% to about 15%.

The product yield from four tubes which were discharged in November will be obtained so that a comparison between this and the calculated yield may be made. A report summarizing this work will be issued as soon as the yield data have been received and evaluated.

Critical Mass Problems

Critical mass problems in the form of plant assistance to the separation process are continually arising. The solutions of these problems are obtained mainly by empirical interpretation of experimental critical mass data for plutonium solutions. The following are a few examples of problems which have arisen during the past quarter.

Several critical limits have been estimated for process tanks in the Redox Separation Plant and are being issued in HW-29758. In preparing these limits, the question arose as to the maximum amount of plutonium which could be mixed with natural uranium and remain critically safe in a hydrogen moderated system. Using the effective resonance integral given by Glasstone and Edlund, (6)

$$\int (\sigma_{au}) \text{ eff } \frac{dE}{E} = 3.9 \frac{\sum_s}{N_u} 0.415$$

along with conservative values for the plutonium cross sections of $\sigma_a = 1400$ and $\sigma_f = 946$ with $u = 2.96$, it was calculated that an atomic ratio of 0.0018

(6) The Elements of Nuclear Reactor Theory, Glasstone and Edlund, p. 256

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would be the maximum Pu-239 to U ratio which can never be made critical.

The problem of sample can storage in the 234-5 Building was considered on the basis of critical safety. It was estimated that six sample cans, each containing 400 grams of plutonium, can be safely stored in each cubicle of the storage vault, providing only two cans are placed on each of the three shelves in the cubicle. Considering sample cans each containing up to 1000 grams of plutonium, it was concluded from volume considerations that two of these cans placed together in each cubicle would remain critically safe.

PuF₄ Neutron Source

The neutron emission of a 50.0 gram sample of PuF₄, obtained from W. H. Reas, was measured by comparing its emission with that of a 10 mg RaBe source. The emission obtained was 6.3×10^3 neutrons/sec/gm PuF₄ with counting statistics of 3%.

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