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 Associated With the CTD-HRP In-Pile Loop

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For The Atomic Energy Commission

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1.0 SUMMARY

An estimation of the thermal neutron flux distribution in the Chemical Technology Division - Homogeneous Reactor Project in-pile loop (for the ORR) indicates that the effective flux for a 4-inch spherical core would be 3.5×10^{13} n/cm²/sec for the most probable loading of the ORR and 5.2×10^{13} n/cm²/sec for the most optimistic loading (long side of loading on same side as experiment).

The gamma heat rate was also estimated and found to vary between 3.1 and 1.9 watts/gm over the spherical core assuming that the front of the sphere is placed 2 inches from the lattice edge. The total gamma heat for the spherical vessel would be 3500 watts, with an additional 1200 watts, or more, in the cooling coils necessary to remove this heat. The gamma heat rate for the solution would be 2.5 watts/gm or a total gamma heat in solution of 1930 watts.

The expected plutonium production rate would be approximately 6 mg Pu²³⁹/day. The fission heat rate would be 4.78 kw/liter - % U²³⁵.

2.0 INTRODUCTION

The best approximation to the conditions which may exist in an actual homogeneous plutonium producer blanket can be met either in a small scale test reactor (HRT) or to a lesser degree in an in-pile loop. The loop would be the cheaper of the two. The information that would be obtained from the operation of an in-pile loop would be (1) plutonium production and chemical characteristics, (2) fission product characteristics, (3) radiolytic decomposition and the effects of various recombination catalysts, (4) hydroclone operational characteristics under reactor blanket conditions, and (5) an indication of corrosion conditions and the behavior of the corrosion products in the highly concentrated soup.

The thermal neutron and gamma flux must be estimated before the complete design of an in-pile loop can be made. Since the CTD-HRP loop will be put in the ORR only an estimate of the gamma heat rate and the thermal flux can be made. Since the ORR will approach the MTR in power level, comparisons with the MTR were considered in these estimates. These values of gamma heat rate and thermal neutron flux determine the total heat which must be removed. The gamma rays heat up all components of the loop as well as the solution while the fission process adds heat to the solution. The removal of this heat, both from the solution and the loop components, must be considered in the design of the loop system.

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This memo estimates the effective neutron flux and gamma heat rate and determines the total gamma and fission heat for a typical design. The plutonium production rate is also estimated.

The additional problems of radiolytic decomposition and catalytic recombination as well as shielding of loop components will be considered in later memos.

3.0 ANALYSIS OF PROBLEMS

3.1 Neutron Flux: Using Fig. 1 which is a plot of the expected neutron flux distribution in the CTD-HRP experimental hole the volumetric mean flux can be obtained for a spherical core of 4" ID assuming the outside edge rests 2 inches from face of lattice. Dividing the sphere into two halves, the following average fluxes exist:

	<u>Max Flux</u>	<u>Prob Flux</u>
Front	1.85×10^{14}	1.30×10^{14}
Rear	1.45×10^{14}	0.95×10^{14}
Average	1.65×10^{14}	1.12×10^{14}

These are the unperturbed values of flux and thus they must be corrected for flux depression. These values as given in Fig. 1 were obtained by correcting the flux distribution in the MTR to operation at 20 MW. Plots were then made for the theoretical thermal flux distribution for a plane source in a finite medium for D_2O (60 cm thick reflector), H_2O (60 cm thick reflector), and Be (30 & 60 cm thick reflector). For comparison the actual MTR (~48 cm Be reflector) normalized pure thermal flux distribution was plotted. These comparisons agree fairly well and are shown in Fig. 2. The increase in the thermal flux which occurs at a short distance in the reflector due to thermalization of the fast flux was estimated only by comparison with MTR data.

Since the ORR will be D_2O reflected in the experimental facilities and H_2O reflected in the areas surrounding the experimental facilities, the actual flux distribution in the facility will be somewhat between that for H_2O and for D_2O . In the experimental hole to be used by the Chemical Technology Division, it was estimated that the flux distribution within the facility would be influenced approximately 2/3 by the reflector within the facility and 1/3 by the reflector surrounding the facility. Using this approximation, the plots given in Fig. 2, and the actual MTR flux distribution, the estimated flux distribution for the ORR was obtained.

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Fig. 1. Expected Flux Distribution in CTD-HRP
Experimental Hole in ORR at 20 Mw.

Fig. 2. Normalized Flux Distributions. (Theoretical).

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The distribution of neutron flux in a finite media was obtained by using the following expression¹:

$$\phi = \frac{\sinh K (a-x)}{2KD \cosh Ka}$$

The neutron flux within the experimental core is much less than either 1.65×10^{14} or 1.12×10^{14} . The actual flux is depressed because of all the structural materials and neutron absorbers present. The flux perturbations were determined by using the method of W. B. Lewis². Using this method, three quantities may be obtained. These are F (flux depression), f (total perturbation) and Y (average attenuation in sample). These quantities are a function of the average number of relaxation lengths in a path length through the sample. Also the quantities are related by $F = \frac{f}{Y}$.

For the 1/8" aluminum sleeve these quantities may be calculated as follows:

$$Y = f = 0.987$$

$$F = 1.0$$

Thus the flux through the aluminum would be $0.987 \phi_0$

where ϕ_0 = unperturbed flux

The 1/8" stainless steel liner would depress the flux even more, the corresponding values would be as follows:

$$Y = 0.833$$

$$f = 0.80$$

$$F = 0.96$$

Thus the flux at the surface of the stainless liner would be:

$$\phi = (0.96) (0.987 \phi_0) = 0.947 \phi_0$$

The largest flux depression comes from the titanium vessel itself. It may be assumed that the vessel will be a 4" I.D. sphere that will withstand 2000 psi. Using the standard stress formula for thickness,

$$t = \frac{PR}{2SE - 0.2P}$$

$$t = \frac{2000 \times 2}{2 \times 8000 \times 0.8 - 400} = \frac{4000}{12,400} = 0.323"$$

$$t = \underline{\underline{5/16" \text{ wall thickness}}}$$

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This example will be used to show the details of the flux depression calculation.

where

$$L_{avg} = 4V/S$$

$$= 4 \times 4\pi \times R_{mt}^2 / 4\pi R_o^2$$

$$R_i = \text{inside radius of sphere} = 5.09 \text{ cm}$$

$$R_m = \text{mean radius of sphere} = R_i + 1/2t$$

$$= 5.09 + 0.41 = 5.50 \text{ cm}$$

$$R_o = \text{outside radius of sphere} = R_i + t$$

$$= 5.09 + 0.82 = 5.91 \text{ cm}$$

$$t = \text{thickness} = 0.82 \text{ cm}$$

$$L_{avg} = 4 \times \frac{5.50^2}{5.91^2} \times 0.820 = 2.84 \text{ cm}$$

$$\lambda^{-1} = N \sigma_{aP}/A = 0.6023 \times 5.8 \times \frac{4.5}{47.9} = 0.328 \text{ cm}^{-1}$$

$$b_{avg} = L_{avg} \lambda^{-1} = 2.84 \times 0.328 = 0.932$$

$$\alpha = 1 - e^{-b_{avg}} = 1 - e^{-0.932} = 0.606$$

$$Y = \alpha / b_{avg} = 0.606 / 0.932 = 0.65$$

$$F = 0.84 \text{ from chart}^2$$

$$f = YF = 0.65 \times 0.84 = 0.546$$

Thus the flux at the surface of the core vessel would be:

$$\phi = (0.84) (0.790 \phi_o) = 0.664 \phi_o$$

and the flux at the inside of the core surface would be:

$$\phi = (0.546) (0.790 \phi_o) = \underline{\underline{0.431 \phi_o}}$$

Similarly the total flux perturbation for the slightly enriched (2%) uranyl sulfate solution may be calculated to be 0.90. This value was found by considering the absorption cross-sections for uranium, water and sulfate. Thus the effective flux in the uranyl sulfate would be $(0.90) (0.431 \phi_o) = \underline{\underline{0.388 \phi_o}}$.

The effective flux values would then be $0.388 \times 1.65 \times 10^{14} = \underline{\underline{6.40 \times 10^{13}}}$ n/cm²/sec Max Flux and $0.388 \times 1.12 \times 10^{14} = \underline{\underline{4.35 \times 10^{13}}}$ n/cm²/sec Most Prob. Flux.

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Any material added to the core vessel will depress the flux accordingly and therefore 6.40×10^{13} and 4.35×10^{13} n/cm²/sec thermal fluxes inside the core are probably high. Some additional material will probably be added to the outside of the core in order to remove the gamma absorption heat. An increase in the total metal used for the core vessel (assuming the added metal is also titanium) of 33% would change the values of Y, F and f to 0.574, 0.78, and 0.447 respectively. Thus the total flux perturbation through the titanium wall would be increased by 22% and the effective fluxes reduced to 5.25×10^{13} n/cm²/sec Max. and 3.56×10^{13} n/cm²/sec Most Prob.

The use of a cylindrical core is another possibility for the experimental rig. Investigation of a cylinder 2" I.D., 5" between heads and hemispherical heads indicates that the total perturbation of flux would only be increased about 8% over that for the sphere. However, the average flux would be somewhat less because of the additional length of hole required. The average fluxes may be 5.70×10^{13} Max and 3.87×10^{13} Most Prob. Without correction for additional titanium for gamma heat removal.

These flux values may also be high since the experimental facility does not lie in the mid-plane of the reactor but at some angle to the radius and above the mid-plane. It would be very difficult at this time to evaluate these effects. However, the flux values are probably conservatively high and thus an over design and safety factor is incorporated.

3.2 Gamma Heat: Fig. 3 is a plot which indicates the gamma heat rate in the MTR and estimates the gamma heat rate in the ORR. The gamma heat rate for the ORR was obtained by multiplying the MTR heat rate by 20/30 or the power ratio of the two reactors.

Gamma heating is an important consideration in the design of the core vessel since the majority of the heat external to the main circulating solution must be removed by an additional heat exchange system.

Using the 4" I.D. spherical core as an example, the average gamma heat rate is as follows: Front hemisphere $q_\gamma = 3.1$ watts/gm, Rear hemisphere $q_\gamma = 1.9$ watts/gm.

The metallic volume of a hemisphere would be: $V = 2\pi R_m^2 t = 2\pi \times 5.50^2 \times 0.82$

$$V = 155 \text{ c.c.}$$

$$\text{Mass} = 155 \times 4.5 = \underline{\underline{700 \text{ gms}}}$$

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Fig. 3. Gamma Heat Rate in MTR and ORR.

Thus the gamma heat rate would be as follows:

$$\text{Front Hemisphere} = 700 \times 3.1 = 2170 \text{ watts}$$

$$\text{Rear Hemisphere} = 700 \times 1.9 = \underline{1330} \text{ watts}$$

$$\text{Total } q_\gamma = \underline{\underline{3500}} \text{ watts}$$

This heat will probably have to be removed by external means.

Additional metal, such as cooling coils will probably be added to the outside of the vessel in order to remove this heat. However, one thing must be borne in mind, any additional metal will also receive heat from gamma radiation and this heat must also be removed. If the total metal were increased by 33% by the addition of cooling coils then an additional 1200 watts must be removed from the cooling coils themselves.

The gamma heat in the solution using 2.5 watts/gm would be as follows:

$$q_\gamma = 2.5 \frac{\text{watts}}{\text{gm}} \times 1.4 \frac{\text{gms sol}}{\text{cc}} \times \frac{4}{3} \pi R_i^3$$

$$q_\gamma = 2.5 \times 1.4 \times \frac{4}{3} \pi \times 5.09^3$$

$$q_\gamma = 2.5 \times 1.4 \times \frac{4}{3} \pi \times 131.87 = \underline{\underline{1930}} \text{ watts}$$

This heat can be removed in the main heat exchanger. On the other hand if the cylindrical vessel were used the gamma heat in solution would be only 1300 watts.

Plutonium Production: The plutonium production may be calculated directly from the flux, vessel volume, concentration and U^{238} cross-section. Using an effective cross-section of 12 barns for U^{238} (thermal plus resonance capture) the plutonium production for the most probable conditions would be as follows:

$$\frac{dN_{49}}{dt} = \phi N_{28} \sigma_{28}$$

$$\frac{dN_{49}}{dt} = 3.56 \times 10^{13} \times \frac{V \times C}{238} \times A \times \sigma_{28} \quad (a)$$

$$V = \frac{4}{3} \pi R_i^3 = \frac{4}{3} \pi \times 5.09^3 = 550 \text{ cc}, C = 0.3 \frac{\text{gms}}{\text{cc}}$$

$$\frac{dN_{49}}{dt} = 3.56 \times 10^{13} \times \frac{550 \times 0.3}{238} \times 6.023 \times 10^{23} \times 12 \times 10^{-24} = 17.85 \times 10^{13} \text{ atoms/sec}$$

(a) Most probable flux with titanium cooling coils around core vessel.

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$$\frac{dN_{49}}{dt} = 17.85 \times 10^{13} \times 8.64 \times 10^4 = 15.45 \times 10^{18} \text{ atoms/day}$$

$$\frac{dN_{49}}{dt} = \frac{15.45 \times 10^{18} \times 1000 \times 239}{6.023 \times 10^{23}} = \underline{\underline{6.12 \text{ mg Pu/day}}}$$

This value may vary considerably depending upon the effective cross-section for U^{238} . Twelve barns seems logical for a 2% enriched system, however this value may decrease with lower enrichments.

Fission Heat Rate: The fission heat rate may be calculated in much the same manner as the plutonium production. The fission rate may be calculated from the following equation:

$$\frac{dNF}{dt} = \phi N_{25} \sigma_f (25) \quad (a)$$

$$dNF = 3.56 \times 10^{13} \times \frac{V \times C \times E \times A}{235} \times \sigma_f (25)$$

assume $E = 0.01$ (1%) then other enrichments will give fission rates as multiples of this

$$\frac{dNF}{dt} = 3.56 \times 10^{13} \times \frac{550 \times 0.3}{235} \times 0.01 \times 6.023 \times 10^{23} \times 590 \times 10^{-24}$$

$$\frac{dNF}{dt} = \underline{\underline{8.90 \times 10^{13} \text{ fissions/sec}}}$$

Using 3.38×10^{10} fissions/watt sec as the fission heat rate, the total fission heat production would be:

$$q_f = \frac{8.90 \times 10^{13}}{3.38 \times 10^{10}} = \underline{\underline{2,630 \text{ watts}}}$$

This rate applies for 1% enrichment, for 2% enrichment the heat rate would be 5,260 watts.

This heat will be removed in the main loop heat exchanger. The specific power in the core under these conditions for 1% enriched uranyl sulfate would be

$$\text{Sp. Power} = \frac{2.63}{0.55} = \underline{\underline{4.78 \text{ Kw/liter}}}$$

(a) Most probable flux with titanium cooling coils around core vessel.

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Since a single-region reactor would probably use 2% enriched fuel it would probably be best to design the loop for these conditions as the upper limit. Thus the total fission heat generated should be considered to be 5,260 watts at these conditions. The loop could be operated at lower enrichments and thus the total fission heat generation would be proportionately less.

5.0 References

1. S. Glasstone and M. C. Edlund, The Elements of Nuclear Reactor Theory, D. Van Nostrand, New York, (1952).
2. W. B. Lewis, "Flux Perturbation by Material Under Irradiation", Nucleonics, Vol. 13, No. 10, p. 82-85, Oct. 1955.

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4.0 APPENDIX

Flux Perturbations by Material Under Irradiation

by Wm. Bradley Lewis

Phillips Petroleum Co., Idaho Falls, Idaho

(Material from Nucleonics, Vol. 13 No. 10, pages 82-85, Oct. 1955.
Also in MTRL-54-27 or IDO 16215)

A semi-empirical method can be used to determine the effective flux throughout a sample in a reactor because the perturbation factor is a function of the ratio of average chord to relaxation length. The justification of the method lies in the results, which have proved applicable and useful to a high degree. Data were gathered at the Materials Testing Reactor (NU, April '54, p. 20), and the results apply strictly to the MTR. But similar curves could be obtained for other reactors.

Material irradiated in a neutron atmosphere absorbs neutrons at a rate proportional to the neutron flux. To calculate this rate, the flux must be known at each small element of the sample's volume. This flux differs from the flux that would exist if the absorptive material were absent.

The sample perturbs the flux by: (a) attenuating it inward from the surface of the sample, and (b) depressing it near the sample. Attenuation results from progressive absorption of neutrons as they penetrate the sample. Depression is caused by the flux gradient that comes into existence when a neutron sink is introduced into the neutron field.

The usual analytical treatment involving the diffusion equation presupposes symmetry and boundary conditions that are not even roughly approximated in practical situations. So this attack has been abandoned in favor of a semi-empirical approach.

DEVELOPMENT

Two rigorous equations describe the rate at which neutrons are captured by a sample in terms of physical and geometric parameters. One may be termed the

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"surface equation"

$$x = p_{avg} F \phi S/4 \quad (1)$$

and the other, the "volume equation"

$$x = f \phi V/\lambda \quad (2)$$

In general, it is not possible to calculate p_{avg} . The quantity $p(b)$ is a function of the path length through the sample. This length is a function of four parameters, two of which are required to locate a surface element in terms of arbitrary surface coordinates, and two of which are required to determine the direction of the path relative to the surface element. For an arbitrary shaped surface, the analytical expression for the average of $p(b)$ cannot be evaluated.

As an approximation we will substitute α for p_{avg} , where α is the probability of capture along the average path

$$L_{avg} = 4V/S \quad (3)$$

This expression is derived from general geometric considerations and holds for arbitrary shapes. However, we will limit consideration to surfaces pierced in no more than two points by an arbitrary straight line.

To return to the desired probability, the expression

$$p(b) = 1 - e^{-b} \quad (4)$$

gives the probability that a neutron is captured in traversing a path whose length is $L = b$.

We have approximated p by α . Actually,

$$p_{avg} \leq p(b_{avg}) = \alpha = 1 - e^{-b_{avg}} \quad (5)$$

The two quantities are equal when b_{avg} is zero, and both p_{avg} and α approach unity as b_{avg} becomes large. The approximation is good throughout the range of b_{avg} . The inequality can be derived from the well-known inequality that the geometric mean does not exceed the arithmetic mean.

So from Eq. 1, it follows that

$$x = (S/4) p_{avg} F \phi \approx (S/4) \alpha F \phi \quad (6)$$

and from Eqs. 6 and 2 it follows that

$$(f/F) = (S/4) (\lambda/V) p_{avg} = (p_{avg}/b_{avg}) \approx (\alpha/b_{avg}) = Y \quad (7)$$

The symbol Y is introduced for convenience.

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DISCUSSION

It now remains to determine the values of f or F ; in most cases f will be found. Samples of known dimensions and compositions are irradiated in regions of known unperturbed flux. The amount of product is determined experimentally, and f is determined directly from Eq. 2.

Since the parameter b_{avg} involves averaging a path length over the sample, it might be expected that shape and symmetry factors would be eliminated and that b_{avg} would be the significant parameter. To what extent this is so is shown in Fig. 1 where the curve f is fitted to the experimental points obtained from samples of various shapes and sizes as shown in the table.

Along with f are plotted the theoretical curve Y , and the derived curve F . The Y -curve serves as a helpful guide as the f -curve must lie below it and the F -curve above it. Both Y and f vanish for b_{avg} infinite; F approaches a non-vanishing positive quantity. These are aids in extrapolating beyond the range of data.

The vanishing of f for infinite values of b_{avg} is implied in Eq. 2. Unless f vanishes, the capture rate would become infinite. Similarly the nonvanishing of F follows from Eq. 1. As b_{avg} becomes infinite, p_{avg} approaches unity, and the rate does not vanish; so F cannot vanish.

The physical significance of the quantities is now clear: α , which is very nearly equal to p_{avg} , measures the "greyiness" of the sample. For a perfect absorber, or black sample, α is unity; for a perfectly transparent sample, α is zero. The symbol F measures the flux depression; f measures the total perturbation. Thus Y is a measure of the average attenuation in the sample.

Throughout this discussion it has been tacitly assumed that there is no "coupling" between the sample and the neutron source. A direct implication of this assumption is that as the greyiness of the sample is increased, the rate of capture of neutrons continually increases. In other words, there can be no extremum in the rate of capture. This implies that the product αF is monotonely increasing with α . This can be true only if $F(1) \geq -(dF/d\alpha)\alpha=1$, which makes $F(1) \geq 1/2$.

These conditions, viz: that the function $\alpha F(\alpha)$ is monotonely increasing, and that $F(1) \geq 1/2$, set fairly rigid limits on the shape of the F vs α curve. The smallest value of $F(1)$ consistent with the experimental data is $F(1) = 0.6$. This

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Fig. 1. Flux Depression (F), Total Perturbation (f), and Average Attenuation in Sample (Y).

Fig. 2. Flux Depression (F), Total Perturbation (f), and Average Attenuation in Sample (Y).

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determines the shape of the F-curve, and hence the f-curve, very closely. Figure 2 shows the three curves plotted against α .

APPLICATION

Since the data points were all based on samples of simple geometric shapes, an example will be chosen that extends the method to a fairly complicated geometry. An array of cobalt disks, 2 cm diameter and 1 mm thick are set on opposite faces of a square graphite prism.

Analysis is based on the simple and reasonable assumptions that (a) the disk's two faces are exposed to different fluxes, an outer flux ϕ_o and an inner flux ϕ_i ; (b) the inner flux consists of neutrons that enter the graphite (nonabsorptive medium) directly, plus those that are attenuated by traversing the cobalt disks. Thus

$$\phi_i = \phi_o(1 - y) + \phi_o y e^{-b_{\text{avg}}} \quad (8)$$

$$\begin{aligned} \phi_{\text{eff}} &= (1/2) (\phi_o + \phi_i) = \phi_o(1 - \alpha y/2) \\ &= \phi F(1 - \alpha y/2) \end{aligned} \quad (9)$$

We shall use this value of effective flux in Eq. 1 to calculate the specific activity of the cobalt. Interaction between disks of the assembly, other than those here considered, is assumed to be negligible.

For this array, we find: $\lambda^{-1} = 3.07 \text{ cm}^{-1}$, $L_{\text{avg}} = 0.1818 \text{ cm}$, $b_{\text{avg}} = 0.558$, $y = (1/2) (\pi d^2/4d^2) = (\pi/8) = 0.3927$, mass of disk = 2.75 grams, $\alpha = 0.425$, $F = 0.90$, $t_{1/2} = 1.67 \times 10^8 \text{ sec}$.

Since each capture produces 1 atom of Co^{60} , we have $n = (S/4)\alpha F(1 - \alpha y/2) \phi t (0.864 \times 10^{19})$, where t is time in days, and $\phi \times 10^{14}$ is neutron flux in neutron $\text{cm}^{-2} \text{ sec}^{-1}$. Disintegrations per second are $n(\ln 2/t_{1/2}) = 4.15 \times 10^{-9} n$. Thus, the total curies is $A = n(4.15 \times 10^{-9}/3.7 \times 10^{10}) = 1.12 \times 10^{-19} n$, and the specific activity, $a = (A/2.75) \text{ curies/gram}$. So, $a = 0.0970 \phi t$.

For the sample taken, $\phi t = 287$; the measured specific activity was 31 c/gm. So we compare the calculated a of 28 c/gm with the measured a of 31 c/gm, and conclude that, in view of the inherent errors in the assumed values for flux, the agreement is excellent.

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EXPERIMENTAL DATA

The actual activities in the table were determined by a number of means -- samples 1 and 2 by fission-product analysis; samples 5, 15, 16 by comparison with a standard cobalt source; samples 3, 4, 6-13 were measured by standard counting techniques; sample 14 was analysed by observing the rate of production of fission heat. The chief source of error in determining ϕ lies in the assumed value of the unperturbed flux. Although extensive flux measurements have been made in the MTR, there have been only a few indirect measurements of nvt (flux averaged over an extended time interval). The probable accuracy of the values used is about $\pm 20\%$.

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Data for Various Samples Irradiated in MTR

No.	Sample	Diameter (in.)	Length (in.)	L _{avg} (in.)	$\frac{-1}{(in.^{-1})}$	b _{avg}	α	f
1.	U ²³⁵	0.050	5	0.050	79.5	4.01	0.982	0.122
2.	U _{nat}	0.188	1.00	0.176	0.97	0.201	0.182	0.97
3.	Au	0.040	*	0.040	14.2	0.566	0.432	0.75
4.	Co	0.040	*	0.040	7.82	0.330	0.26	0.87
5.	Co	0.295	1.18	0.262	7.82	2.05	0.872	0.30
6.	Co	0.020	0.017	0.0126	7.82	0.0985	0.097	0.960
7.	Co	0.040	0.036	0.0257	7.82	0.203	0.183	0.854
8.	Co	0.119	0.118	0.079	7.82	0.617	0.450	0.654
9.	Co	0.186	0.188	0.124	7.82	0.975	0.620	0.537
10.	Co	0.020	0.250	0.0199	7.82	0.155	0.144	0.920
11.	Co	0.040	0.250	0.0397	7.82	0.311	0.268	0.887
12.	Co	0.119	0.250	0.0961	7.82	0.752	0.528	0.613
13.	Co	0.186	0.250	0.136	7.82	1.062	0.653	0.530
14.	U _{nat}	1.5	*	1.5	0.85	1.28	0.722	0.30
15.	Co	0.312	3.0	0.269	7.82	2.31	0.899	0.238
16.	Co	0.240	12.0	0.240	7.82	1.88	0.848	0.30

*Length not known, but large compared to diameter.

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SYMBOLS

- A = atomic weight of sample
- b = L/λ = number of relaxation lengths in distance L
- F = ratio of effective flux to unperturbed flux at sample's surface
- f = experimentally determined factor to make Eq. 2 rigorous
- L = path length through sample
- P_{avg} = $[p(b)]_{avg}$ = probability of capture
- $p(b)$ = probability of capture along a path whose length is λb
- S = surface area of sample
- U_{avg} = weighed average of u taken over all paths through sample; weighing factor is cosine of angle between path and normal to surface at point of entry
- V = volume of sample
- x = rate of neutron capture by sample
- Y = α/b_{avg} = average attenuation in sample
- y = fraction of graphite surface covered by cobalt
- α = $p(b_{avg}) = 1 - e^{-b_{avg}} =$ probability of capture along path whose length is $L_{avg} = b_{avg} \lambda$
- λ = $(N \sigma_p / A)^{-1}$ = relaxation length for sample material
- ρ = density of sample
- σ = absorptive cross section of sample
- ϕ = unperturbed value of neutron flux

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