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# SURVEY OF U-Bi REACTORS

Reprint of Project Dynamo Memo PD-51


By
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August 5, 1953

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M E M O R A N D U M

# SURVEY OF U-Bi REACTORS

(Reprint of Project Dynamo Memo PD-51)

By J. Chernick

The liquid fuel group is investigating the following reactor type: a graphite structure with channels for a U-Bi solution in the core, and a Th-Bi slurry in the blanket. The core and blanket are externally cooled, a small amount of the U-Bi solution being diverted to the blanket for this purpose. A survey of the nuclear properties of this reactor type as a function of the degree of moderation and enrichment is given in the present memorandum.

## Method

The age $\theta$ in C-Bi mixtures is calculated from the formula

$$\theta = \frac{\theta_C (1 + V_{Bi}/V_C)^2}{\left[1 + \frac{N_{Bi}}{N_C}\left(\frac{\overline{\sigma_{tr}(Bi)}}{\overline{\sigma_{tr}(C)}}\right)\right]\left[1 + \frac{N_{Bi}}{N_C}\,\frac{\xi_{Bi}}{\xi_C}\left(\frac{\overline{\sigma_s(Bi)}}{\overline{\sigma_s(C)}}\right)\right]}$$

where $\theta_C$ is the measured age in graphite and the bars indicate averages over logarithmic energy intervals. The formula should be fairly accurate since the ratio of the mean free paths of bismuth to carbon does not vary greatly with energy. Inelastic scattering in bismuth may reduce these values somewhat, but the experimental data are too meager to attempt such a refinement.

We assume an average reactor temperature of 475°C and take the thermal neutron energy at 0.065 ev. The bismuth density at this temperature is 9.82 and we take the graphite density as 1.6. The latter value is low and it may be useful to repeat the present calculations for a higher graphite density since graphite of density as high as 1.8 is available.

The following absorption cross-sections at 2200 m/s and average thermal scattering cross-section are assumed:

$$\sigma_a(C) = 0.0044 \text{ b} \qquad \bar{\sigma}_s(C) = 4.8 \text{ b}$$

$$\sigma_a(Bi) = 0.032 \qquad \bar{\sigma}_s(Bi) = 9.0$$

$$\sigma_a(U_{233}) = 564 \qquad \bar{\sigma}_s(23) = 8.2$$

With these constants the age formula becomes

$$\theta = \frac{\theta_g(1 + x)^2}{(1 + 0.746\ x)(1 + 0.0426\ x)}$$

with $x = V_{Bi}/V_C$ = bismuth to carbon volume ratio.

The age in graphite $\theta_g = 324$ cm$^2$ was found by adding an increment of 13 cm$^2$ to the measured value of 311 cm$^2$ to indium resonance energy (1.44 ev) for a fission source.

Values of the age as a function of the Bi:C volume ratio are tabulated below, and graphed in Fig. 1.

| $V_{Bi}/V_C$ | $\theta$ |
|---|---|
| 0 | 324 |
| .25 | 422 |
| .50 | 520 |
| 1.0 | 712 |
| 1.5 | 898 |
| 2.0 | 1078 |

## Diffusion Areas

Diffusion areas $L^2$ were determined for 3 possible enrichments of the $U_{233}$-Bi solution. Atom ratios $N_U/N_{Bi}$ of $6 \times 10^{-4}$, $10 \times 10^{-4}$, and $15 \times 10^{-4}$ were used. The absorption cross-sections were averaged for a Maxwell distribution of velocities at 0.065 ev. The diffusion theory formula for a homogeneous reactor,

$$L^2 = 1/3\, \overline{\Sigma}_a\, \overline{\Sigma}_{tr}$$

was used to calculate the diffusion areas. Results are shown in Figs. 2 to 4 and are tabulated below.

| | $N_{23}/N_{Bi} = 6\times10^{-4}$ | | | $N_{23}/N_{Bi} = 10\times10^{-4}$ | | | $N_{23}/N_{Bi} = 15\times10^{-4}$ | |
|---|---|---|---|---|---|---|---|---|
| $V_{Bi}/V_C$ | $L^2$ | $\overline{\Sigma}_a$ | $\overline{\Sigma}_{tr}$ | $L^2$ | $\overline{\Sigma}_a$ | $\overline{\Sigma}_{tr}$ | $L^2$ | $\overline{\Sigma}_a$ |
| 0 | 4699 | $.1952\times10^{-3}$ | .3634 | 4699 | | | 4699 | $.1952\times10^{-3}$ |
| .25 | 742 | 1.316 | .3415 | 482 | 2.023 | .3416 | 336 | 2.906 |
| .50 | 494 | 2.064 | .3270 | 314 | 3.241 | .3270 | 216 | 4.714 |
| 1.0 | 360 | 2.998 | .3088 | 227 | 4.764 | .3088 | 155 | 6.973 |
| 1.5 | 314 | 3.559 | .2978 | 197 | 5.678 | .2979 | 134 | 8.328 |
| 2.0 | 292 | 3.932 | .2906 | 182 | 6.288 | .2906 | 128 | 9.232 |

## Multiplication Factor and Buckling

If the effect of fission and radiative capture of neutrons during the slowing down process can be neglected, then the multiplication factor of the reactor $k_\infty = \eta f$, where f is the thermal utilization of the fuel and $\eta = 2.33 \pm .03$. For the assumed homogeneous mixture, we find

$$f = \frac{4.525\times10^4 (N_U/N_{Bi})x}{1 + x\left[2.567 + 4.525 \times 10^4 (N_U/N_{Bi})\right]}$$

with $x = V_{Bi}/V_g$.

The critical buckling $B^2$ is obtained from the age relation

$$k_\infty = e^{\theta B^2}(1 + L^2B^2) \ .$$

For a reactor with a large fractional neutron leakage, the latter equation is more accurate than the one-group approximation

$$k_\infty = 1 + (\theta + L^2)B^2 \ .$$

Results of the calculations for the three different enrichments are given below.

Case I. $N_U/N_{Bi} = 6 \times 10^{-4}$

| x | f | $k_\infty$ | $B^2$ | $a_c$ |
|---|---|---|---|---|
| .25 | .8053 | 1.876 | $7.93 \times 10^{-4}$ cm$^{-2}$ | 6.34 ft. |
| .50 | .8560 | 1.994 | 7.56 | 6.49 |
| 1.0 | .8839 | 2.059 | 6.99 | 6.75 |
| 1.5 | .8936 | 2.082 | 6.24 | 7.15 |
| 2.0 | .8985 | 2.094 | 5.47 | 7.63 |

Case II. $N_U/N_{Bi} = 10 \times 10^{-4}$

| x | f | $k_\infty$ | $B^2$ | $a_c$ |
|---|---|---|---|---|
| .25 | .8732 | 2.035 | $8.55 \times 10^{-4} cm^{-2}$ | 6.11 ft. |
| .50 | .9083 | 2.116 | 9.42 | 5.82 |
| 1.0 | .9269 | 2.160 | 8.37 | 6.17 |
| 1.5 | .9333 | 2.175 | 7.18 | 6.66 |
| 2.0 | .9365 | 2.182 | 6.24 | 7.14 |

Case III. $N_U/N_{Bi} = 15 \times 10^{-4}$

| x | f | $k_\infty$ | $B^2$ | $a_c$ |
|---|---|---|---|---|
| .25 | .9118 | 2.124 | $10.62 \times 10^{-4} cm^{-2}$ | 5.48 ft. |
| .50 | .9370 | 2.183 | 10.94 | 5.40 |
| 1.0 | .9501 | 2.214 | 9.27 | 5.86 |
| 1.5 | .9546 | 2.224 | 7.80 | 6.39 |
| 2.0 | .9568 | 2.229 | 6.69 | 6.90 |

In addition to the values of f, $k_\infty$, and the buckling $B^2$, we have listed the critical width $a_c$ of a bare cube as an indication of the reactor size.

Reflector Savings

The critical mass of a small reactor is quite sensitive to the assumed reflector savings. For a purely thermal reactor surrounded by a large blanket of similar scattering properties, the reflector savings is, to first order, equal to the diffusion length $L_o$ in the blanket. However, if the great majority of the leakage neutrons are in the fast group, the reflector savings is of order $\sqrt{L_o^2 + \theta_o}$ . The importance of the fast group is indicated by the ratio $\theta/L^2$. If an atom ratio $N_{th}/N_{Bi} \approx 10^{-1}$ is assumed for the blanket, then the age and diffusion areas in the blanket will be similar to those already obtained for the reactor core, and the reflector savings should run from about 20 to 30 cm for the cases considered here. Two group calculations should be carried out for the specific reactors, which are considered desirable on the basis of the present survey. These calculations should also aim at determining the minimum blanket thickness required to reduce neutron leakage from the blanket to negligible proportions. To be conservative, we assume a reflector savings of 20 cm for the present.

Comparison of Reactor Designs

A comparison of the various reactor types will be made on the following points: breeding gain, $U_{233}$ and Bi inventory, reactor core volume, and moderator to fuel atomic ratio.

In order to determine the breeding gain, we need to know the blanket efficiency, i.e., the thermal utilization of the Th in the blanket. The latter is given by the formula:

$$f_{th} = \frac{561.6 \times N_{th}/N_{Bi}}{1 + x(2.567 + 561.6 \frac{N_{th}}{N_{Bi}})}$$

with $x = V_{Bi}/V_C$.

The effect of increasing the atomic ratio $N_{th}/N_{Bi}$ from 0.06 to 0.12 is shown in the following table.

| | $N_{th}/N_{Bi}$ = .06 | $N_{th}/N_{Bi}$ = .12 |
|---|---|---|
| $x = V_{Bi}/V_C$ | $f_{th}$ | $f_{th}$ |
| .25 | .8349 | .9112 |
| .50 | .8807 | .9365 |
| 1.0 | .9044 | .9497 |
| 1.5 | .9126 | .9542 |
| 2.0 | .9167 | .9564 |

The breeding gain can now be obtained as follows. For each $U_{233}$ atom destroyed in the core, $\eta$ neutrons are produced and

$$\eta - \eta\frac{e^{-\theta\Delta}}{1 + L^2\Delta} = \eta(k_\infty - 1/k_\infty)$$

neutrons leak into the fertile blanket. If $f_{th}$ is the blanket efficiency, then the breeding gain is given by

$$\eta f_{th}(\frac{k_\infty - 1}{k_\infty}) - 1 \quad .$$

The breeding gains are listed in Table I for values of $f_{th}$ = 0.9 and 0.95, respectively. The reactor volumes listed in the same table are those of a critical cube with a 20 cm reflector saving. The ratios $N_C/N_U$ are given in order to indicate the degree of thermality of the reactor.

## Table I

### Comparison of U-Bi-C Reactor Possibilities

| $V_{Bi}/V_C$ | Reactor Vol. ($ft^3$) | Mass $U_{233}$ (kg) | Mass Bi (tons) | $N_C/N_U$ | B.G. $f_{th}$ = .9 | B.G. $f_{th}$ = .95 |
|---|---|---|---|---|---|---|
| Case I: $N_U/N_{Bi} = 6 \times 10^{-4}$ | | | | | | |
| .25 | 127 | 4.72 | 7.8 | 18,900 | -.021 | .034 |
| .50 | 114 | 7.08 | 11.6 | 9,440 | .045 | .103 |
| 1.0 | 161 | 15.0 | 24.6 | 4,720 | .077 | .137 |
| 1.5 | 198 | 22.1 | 36.4 | 3,150 | .089 | .150 |
| 2.0 | 253 | 31.3 | 51.5 | 2,360 | .096 | .157 |
| Case II: $N_U/N_{Bi} = 10 \times 10^{-4}$ | | | | | | |
| .25 | 110 | 6.82 | 6.7 | 11,300 | .067 | .127 |
| .50 | 91 | 9.45 | 9.3 | 5,670 | .106 | .168 |
| 1.0 | 115 | 17.8 | 17.6 | 2,830 | .126 | .188 |
| 1.5 | 153 | 28.5 | 28.1 | 1,890 | .133 | .196 |
| 2.0 | 198 | 41.0 | 40.5 | 1,420 | .137 | .200 |
| Case III: $N_U/N_{Bi} = 15 \times 10^{-4}$ | | | | | | |
| .25 | 72 | 6.76 | 4.4 | 7,560 | .109 | .170 |
| .50 | 68 | 10.5 | 6.9 | 3,780 | .137 | .200 |
| 1.0 | 94 | 21.9 | 14.4 | 1,890 | .150 | .214 |
| 1.5 | 131 | 36.5 | 24.0 | 1,260 | .154 | .218 |
| 2.0 | 175 | 54.1 | 35.6 | 940 | .156 | .220 |

The effect of increasing the thorium concentration in the blanket is quite marked. The fuel and bismuth inventory shown refers only to that in the reactor core, several times this amount being required for the complete system. The reactors with bismuth volumes exceeding the graphite volumes have attractive breeding gains but require an excessive amount of fuel. At an atomic ratio of $N_U/N_{Bi} = 10 \times 10^{-4}$, equal bismuth to graphite volumes appears to be the best choice. The choice of $N_U/N_{Bi} = 6 \times 10^{-4}$ also appears satisfactory at a 1:1 bismuth to graphite volume ratio, if a blanket efficiency of 0.95 can be achieved.

The following problems still need to be studied:

1. The neutron leakage and Bi and Th inventory in the blanket as a function of the lateral blanket thickness, graphite density, and bismuth to graphite volume ratio.

2. The end leakage from the reactor. The liquid fuel group has suggested the use of end reflectors containing Th-Bi slurry tubes, but the multiplication factor in this region is still large and there is some question whether a simple tapered duct with no moderator would not prove just as effective.

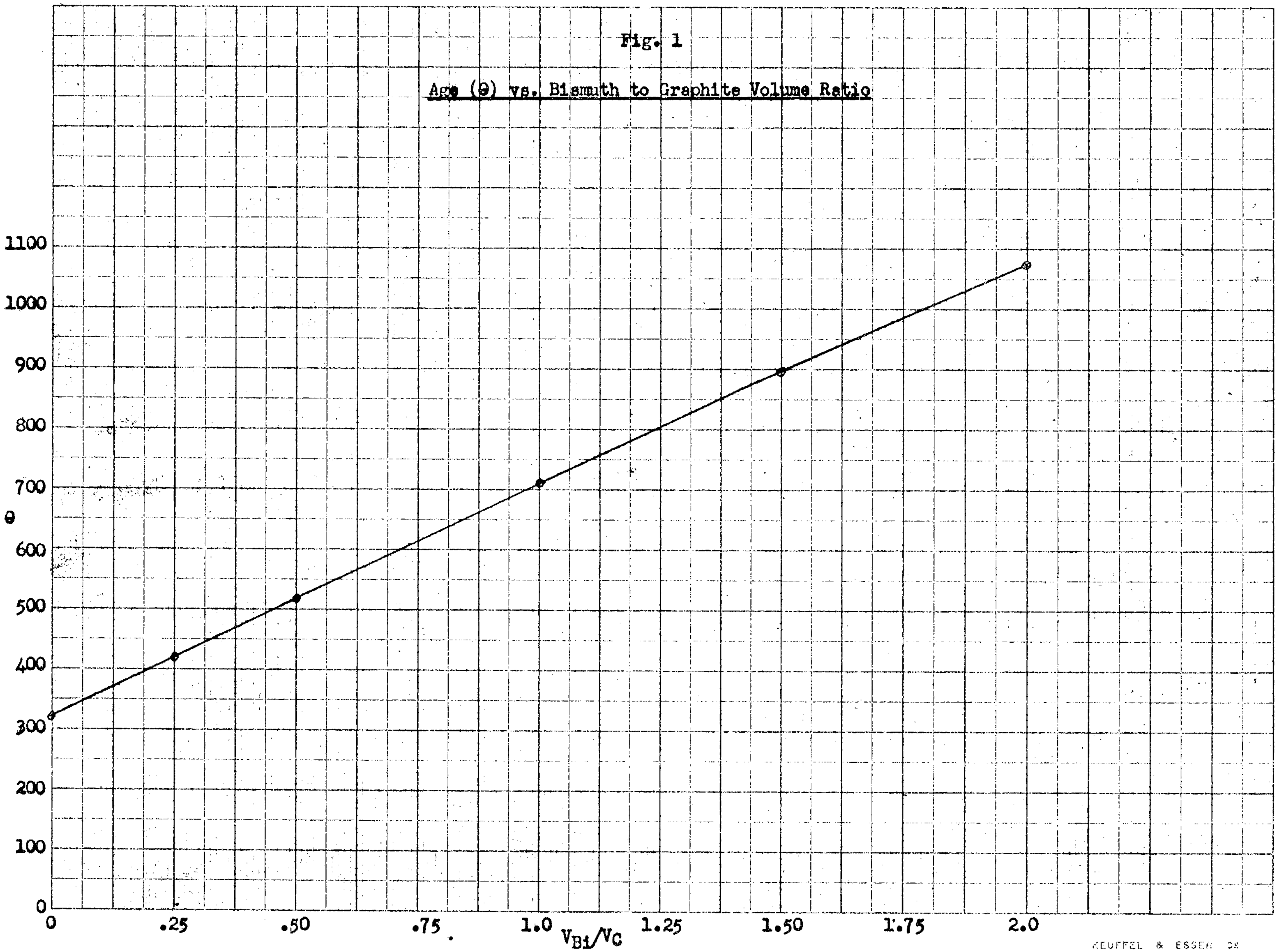
3. Long term changes in the breeding gain and uranium concentration due to the buildup of higher isotopes as a function of the initial breeding gain.

4. The effect of the reactor size on the possible power output of the reactor.

5. The temperature coefficient of the reactor as a function of the bismuth to graphite volume ratio.

Fig. 1

Age (θ) vs. Bismuth to Graphite Volume Ratio



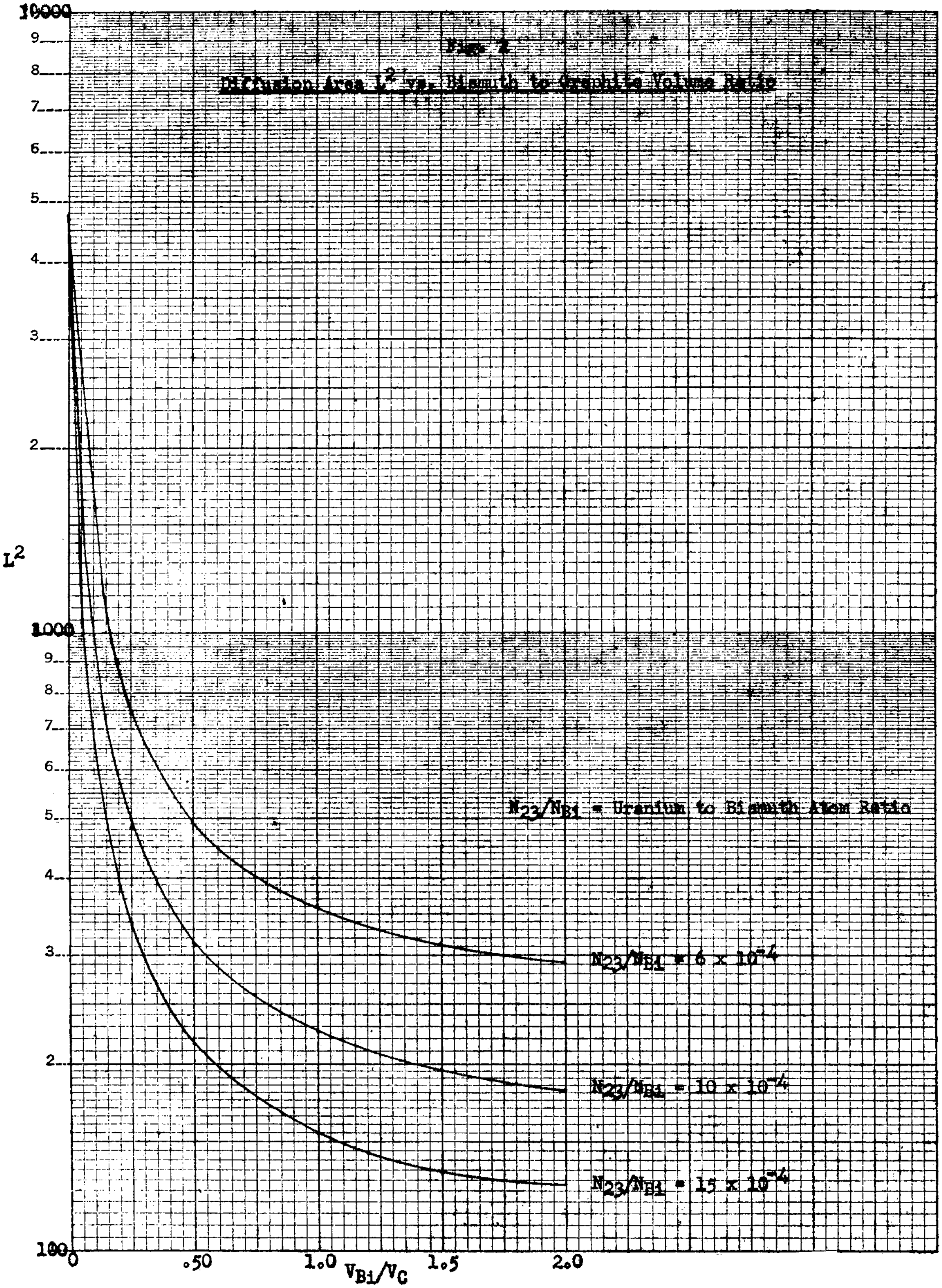



Fig. 2

Diffusion Area $L^2$ vs. Bismuth to Graphite Volume Ratio