MASTER

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

UNION CARBIDE

ORNL-TM-399

COPY NO. - 50

DATE - October 17, 1962

A FACILITY OF HIGH THERMAL NEUTRON FLUX IN THE ABSENCE OF FAST NEUTRONS AND GAMMA RAYS

R. van der Walt* and A. L. Colomb

ABSTRACT

A facility capable of yielding a high thermal neutron flux in the absence of fast neutrons and gamma rays has been studied. This facility, if used with a 1-Mw pool-type reactor, will yield at the target position:

a thermal flux of 9×10^{11} $n \cdot cm^{-2} \cdot sec^{-1}$.

a thermal flux to fast neutron dose ratio of 4×10^{11} $n \cdot cm^{-2} \cdot rem^{-1}$.

a $0 \rightarrow 10$ kev flux to 10 kev $\rightarrow$ 10 Mev neutron dose ratio of 9×10^{12} $n \cdot cm^{-2} \cdot rem^{-1}$.

a thermal flux to gamma dose ratio of 2.9×10^{11} $n \cdot cm^{-2} \cdot rad^{-1}$.

These results were obtained using the Twenty Grand Code[1] to compute the fast and thermal neutron flux distributions and the Nightmare Code[2] to calculate the gamma dose rates.

*On loan from the South African Atomic Energy Board, Pretoria, Republic of South Africa.

A FACILITY OF HIGH THERMAL NEUTRON FLUX IN THE ABSENCE OF FAST NEUTRONS AND GAMMA RAYS

INTRODUCTION

Experiments often require the accumulation of a high thermal neutron dose during a short irradiation time and, at the same time, a limited fast neutron and gamma-ray dose.

These requirements cannot be met by any existing facility of the ORR and LITR reactors where a high thermal neutron flux is always accompanied by a high fast neutron flux and high gamma-ray dose rates. Lack of space and flexibility prevents the introduction of large quantities of material close to the cores of these reactors, and little can be done to improve the thermal-to-fast neutron ratio and thermal neutron to gamma dose rate ratio in any facility without serious reduction of the thermal neutron flux itself.

Such flexibility, however, does exist at the Bulk Shielding Reactor (BSR). The average thermal neutron flux of the BSR (5 x 5 element loading) at a power of 1 Mw is of the order of 5.6×10^{12} $n \cdot cm^{-2} \cdot sec^{-1}$. It should thus be possible to obtain a thermal neutron flux of the order of 10^{12} $n \cdot cm^{-2} \cdot sec^{-1}$ at large distances from the reactor core if a suitable reflector material is used. The gamma dose rate at the core face of the BSR is of the order of 10^{8} r/hr at 1 Mw so that a material with suitable gamma ray attenuation properties will have to be introduced between the core and the irradiation position.

As a reflector, a material with good moderating properties and low thermal neutron absorption will have to be used. This will lead to a high thermal-to-fast neutron ratio at positions far from the core without serious reduction of the thermal neutron flux. Greater experimental flexibility would be obtained by using a liquid moderator. Heavy water (D_2O) meets these requirements more closely than any other moderator material.

Bismuth was chosen as a gamma-ray attenuating material rather than lead because of its smaller absorbtion cross section for thermal neutrons. This not only reduces thermal neutron losses but also decreases production of capture gamma rays.

PROCEDURE FOR COMPUTATION

Flux Distribution

Various geometries were analyzed for neutron and gamma fluxes at the target location. The bismuth shield, a slab having the same surface area as the reactor and a thickness of 10 inches, was located at various positions between the reactor and the target. Bismuth was also homogeneously mixed with D_2O. These calculations did not yield results as good as those obtained with the geometry illustrated in Figure 1 where the bismuth shield surrounds the target position. The results presented in this report were obtained with this latter geometry. The thermal and fast neutron flux distributions were computed by making use of the Twenty Grand Code[1], which solves the neutron diffusion equations in two dimensions for up to six neutron groups.

Cylindrical geometry was assumed, and a 3 x 7 BSR-element core was equated to a cylindrical core on the basis of equal buckling. This was done to insure equal leakage in the computed case and in the actual case.

Four neutron groups with energies lying in the ranges of 10 Mev to 100 kev, 100 kev to 10 kev, 10 kev to 0.14 ev, and thermal (Maxwellian distribution with an upper energy of 0.14 ev) were chosen.

The few group constants for the four neutron groups were computed by using the Modric-Code[3]. This code provides cross sections as a function of energy and solves the neutron diffusion equation in one dimension for up to 50 energy groups. The other two dimensions were taken care of by using buckling factors. The amount of various materials composing the reactor core was computed from the geometry of the fuel elements and homogenized over the core volume. In the reflector region, heavy water was assumed to contain 0.3% H_2O.[4]

The thermal neutron flux distribution along the centerline of the reactor, as computed by the Twenty Grand Code, is shown in Figure 2.

UNCLASSIFIED
ORNL-LR-DWG 74451

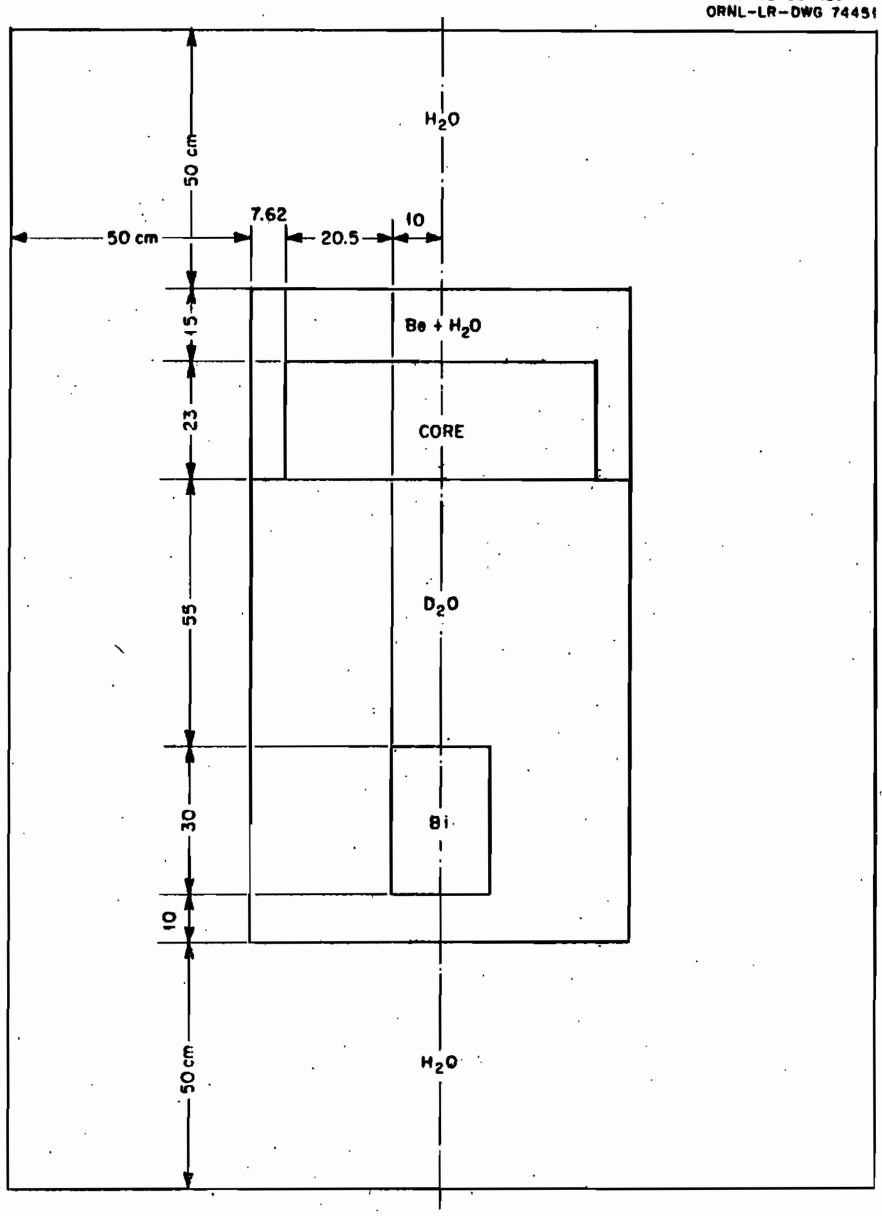

Figure 1. Reactor Geometry Used in Computations.

UNCLASSIFIED
ORNL-LR-DWG 74452

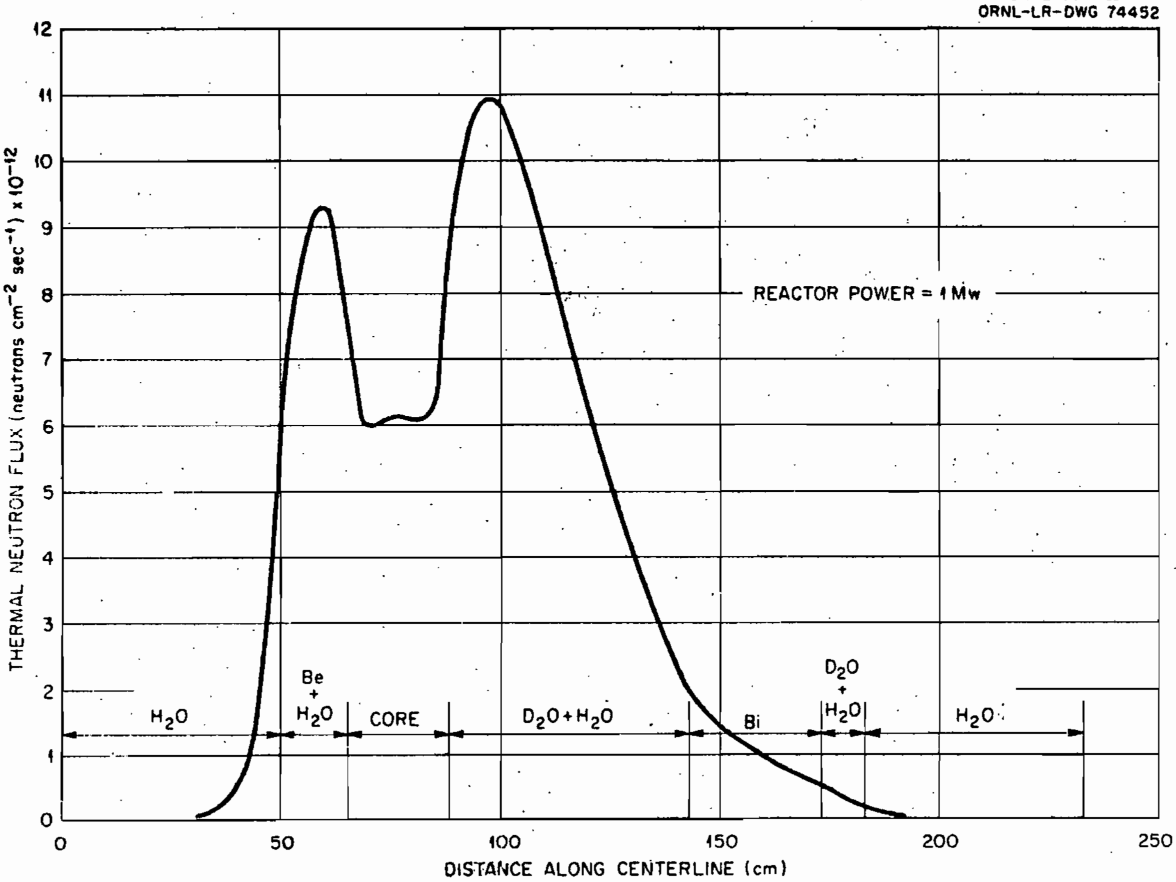

Figure 2. Thermal Neutron Flux at Reactor Center Line.

The average reactor flux is equal to 6.25 10^{12} $n\cdot cm^{-2}\cdot sec^{-1}$, and the ratio of reflector peak flux in D_2O to reactor average flux is 1.71.

Figure 3 shows the variation of the thermal neutron flux along the axis of the bismuth cylinder, and it can be seen that a thermal flux greater than 10^{12} $n\cdot cm^{-2}\cdot sec^{-1}$ can be obtained up to 15 cm into the bismuth.

Figure 4 indicates that the radial thermal flux distribution in the bismuth is almost flat, thus insuring evenly distributed irradiation of the sample.

Figure 5 shows the neutron spectrum as computed with the Modric Code, using 34 neutron groups, at the chosen irradiation position. The flux at this position is very well thermalized, and the ratio of thermal neutron flux to fast neutron dose rate is of the order of 4 x 10^{11} $n\cdot cm^{-2}\cdot rem^{-1}$. Fast neutron dose rates were calculated by using energy-averaged flux to dose-rate conversion factors obtained from the <u>Reactor Shielding Design Manual</u>.[5] In this ratio all neutrons with energy above 0.14 ev are considered as fast. The ratio of 0 $\rightarrow$ 10 kev neutron flux to 10 kev $\rightarrow$ 10 Mev neutron dose rate is of the order of 9 x 10^{12} $n\cdot cm^{-2}\cdot rem^{-1}$. These ratios, as computed from the Twenty Grand and Modric Codes, agree.

Gamma-Ray Dose Rate

The gamma-ray dose rate at several points along the axis of the bismuth cylinder was computed by the use of the Nightmare Code.[2] This code computes the gamma dose rate at any point in or near a reactor by means of the NDA buildup-factor method.[6] The source distribution for this calculation was computed from the flux distribution obtained by the Twenty Grand Code. Six gamma energy groups were used.

The gamma-ray spectrum used for computing the dose rate due to the core (fission plus capture) is compared in Figure 6 with the gamma-ray spectrum measured at the BSR.[7] The energy distribution of the various capture gamma produced by the materials surrounding the reactor core was taken from Table B.7 of Reference 2.

Figure 7 shows the contribution to the gamma-ray dose rate at different positions in the bismuth due to the various materials of the facility. At a certain distance from the bismuth face the total gamma

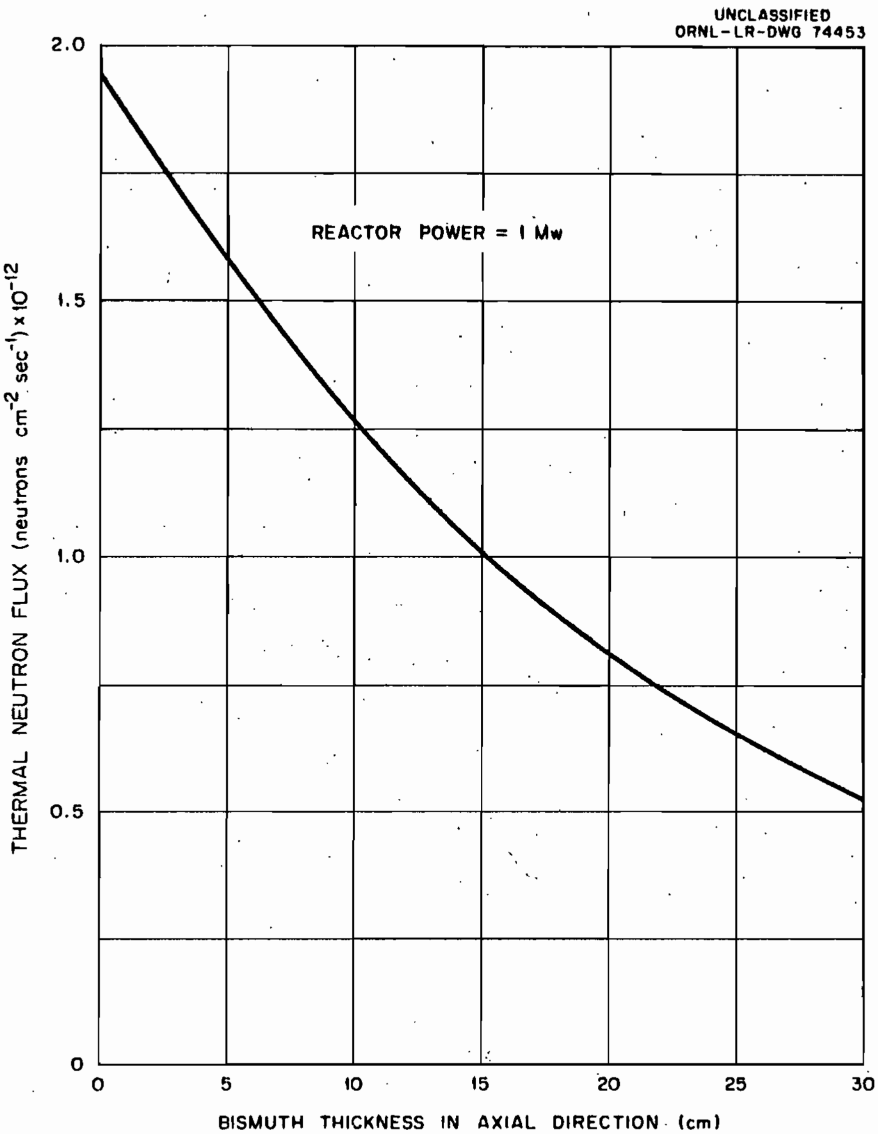

Figure 3. Thermal Neutron Flux in Bismuth in the Axial Direction.

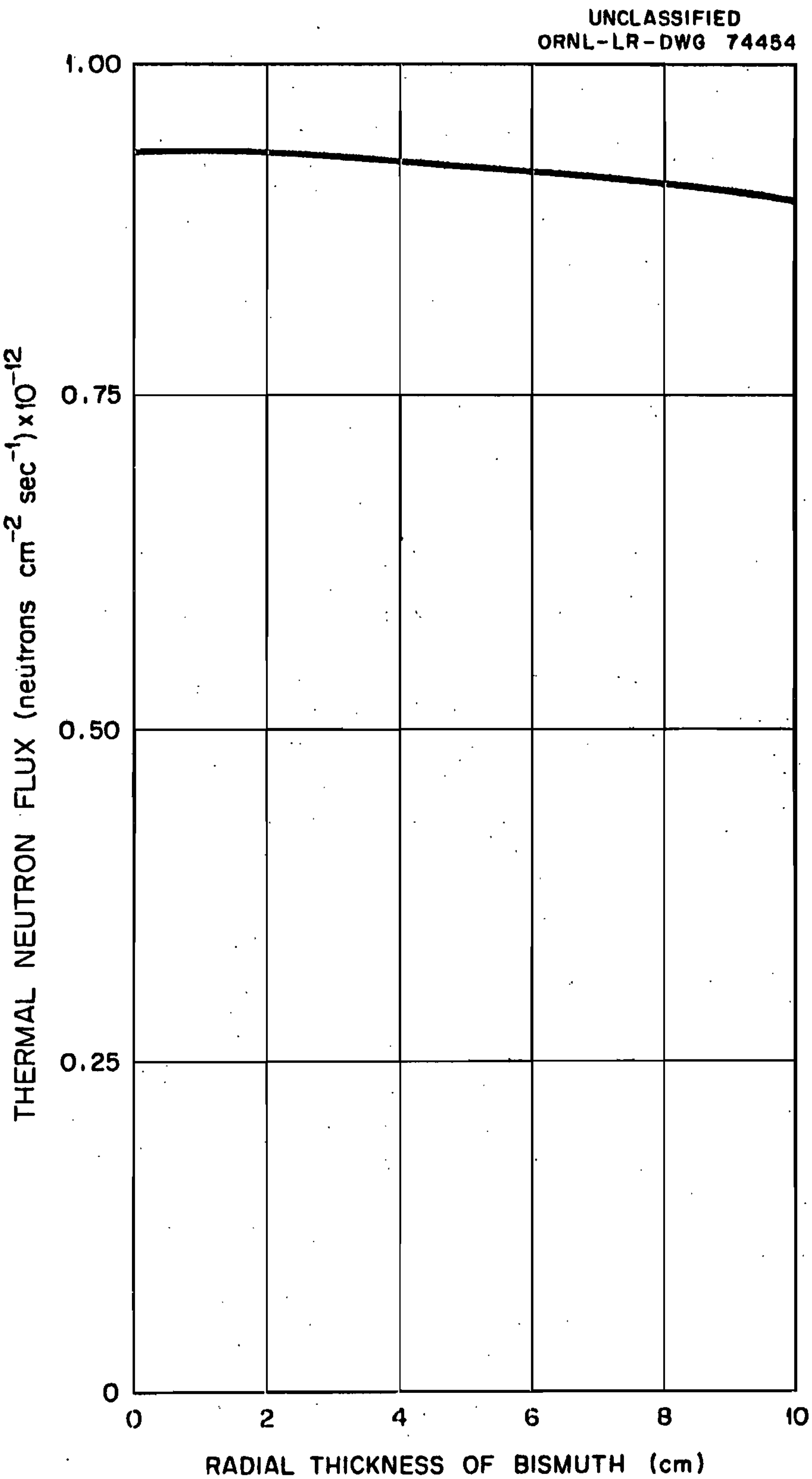

Figure 4. Thermal Neutron Flux in Bismuth in the Radial Direction from the Chosen Sample Position.

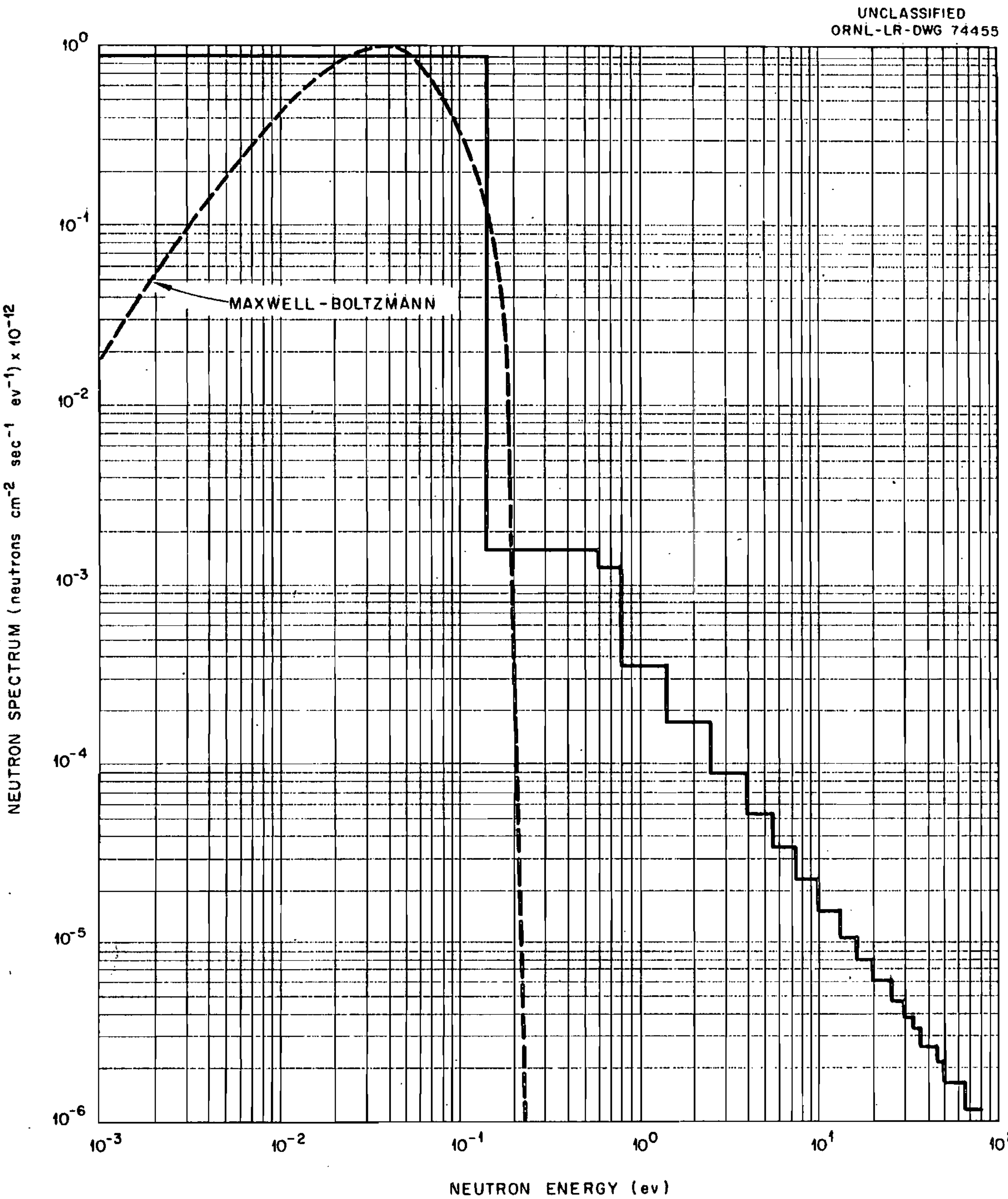

Figure 5. Neutron Spectrum at Sample Position.

UNCLASSIFIED
ORNL-LR-DWG 74456

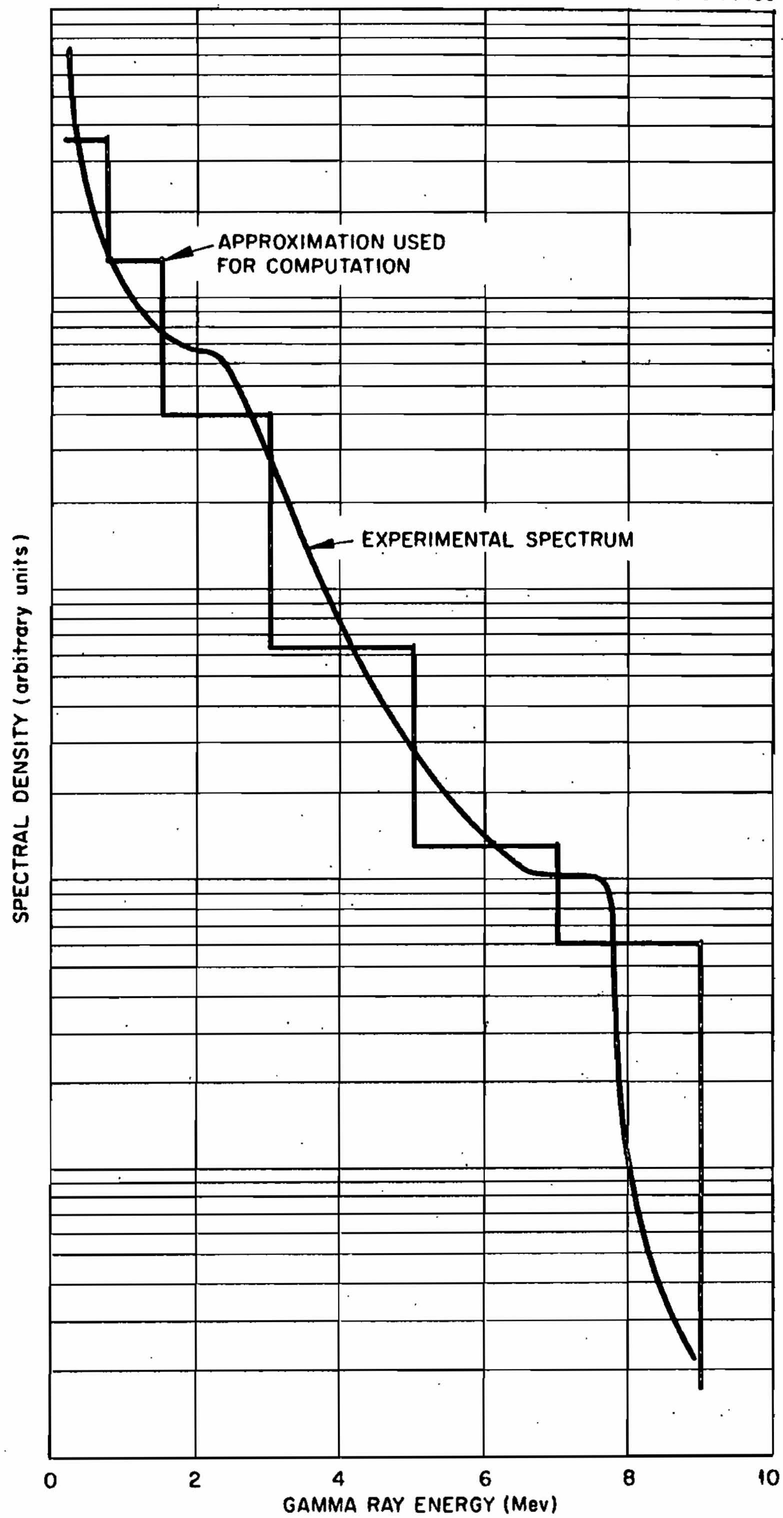

Figure 6. Gamma-Ray Spectrum Used for Computation Compared to the Experimentally Determined Spectrum.

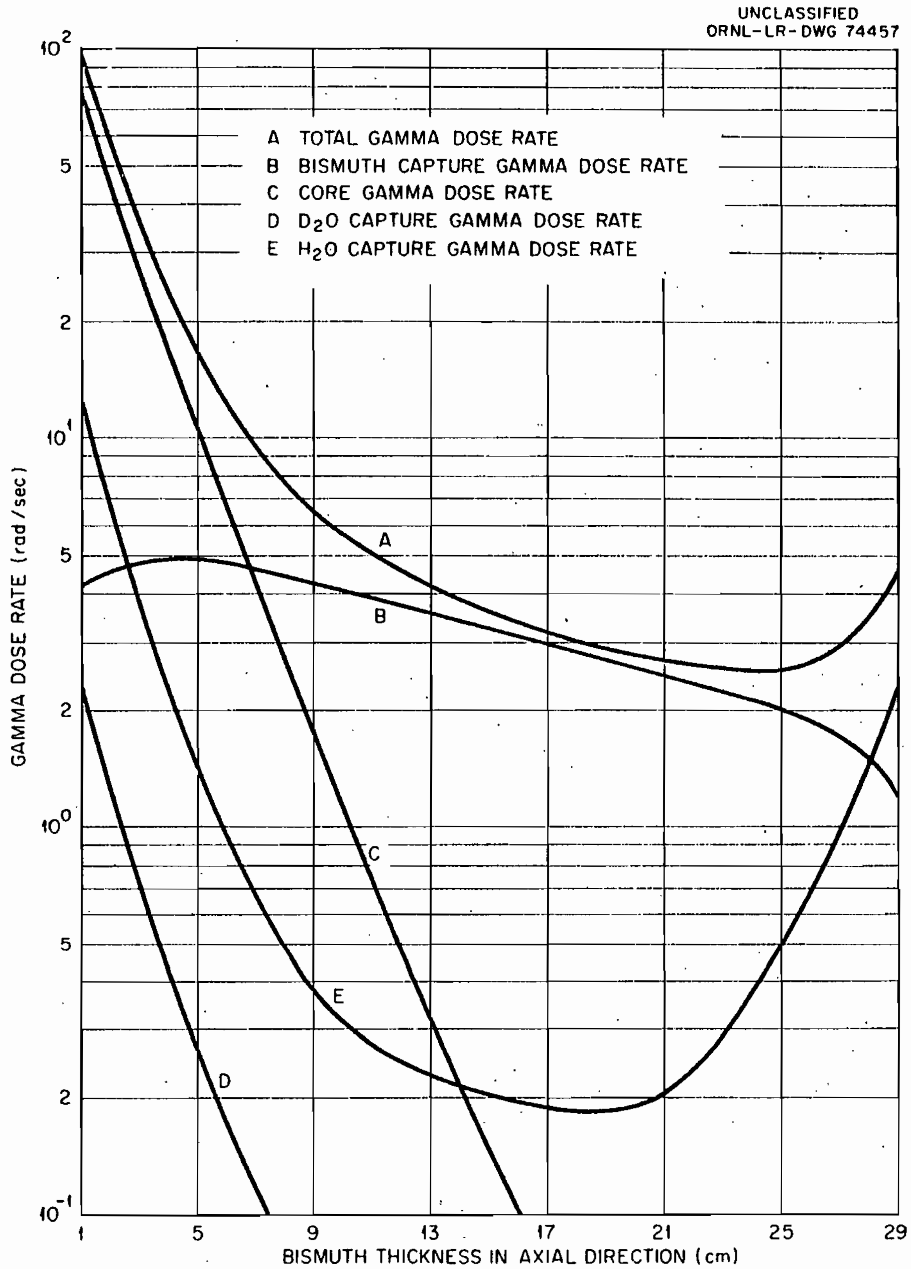

Figure 7. Contribution to Gamma Dose Rate in Bismuth from Various Regions of the Facility.

dose is almost the same as the dose produced by thermal neutron captures in the bismuth only. This indicates that the thermal neutron to gamma dose ratio cannot be improved by either increasing the bismuth thickness or the thermal neutron flux.

Figure 8 shows the variation of the ratio of thermal neutron flux to gamma dose rate with position in the bismuth. It can be seen that this ratio is fairly constant over a certain portion of the bismuth cylinder and reaches a value of 2.9×10^{11} at the chosen irradiation position.

CONCLUSION

It thus seems possible, by making use of the BSR reactor operating at 1 Mw, to obtain an irradiation position with the following characteristics:

Thermal neutron flux: 9×10^{11} $n \cdot cm^{-2} \cdot sec^{-1}$.

Thermal neutron flux to fast neutron dose rate ratio:
4×10^{11} $n \cdot cm^{-2} \cdot rem^{-1}$.

$0 \rightarrow 10$ kev flux to 10 kev $\rightarrow 10$ Mev neutron dose rate ratio:
9×10^{12} $n \cdot cm^{-2} \cdot rem^{-1}$.

Thermal neutron flux to gamma dose rate ratio:
2.9×10^{11} $n \cdot cm^{-2} \cdot rad^{-1}$.

PROPOSED FACILITY

In order to achieve the radiation conditions described above, the following facility could be built. It would consist of an aluminum box 60 cm by 70 cm by 95 cm. A bismuth cylinder of 10 cm radius and 30 cm length would be located close to one end of the box. The bismuth weight would be approximately 92 kg. The bismuth shield should be movable along the box axis, and a flexible penetration pipe should connect the irradiation position in the bismuth to the box wall. This is to allow the experimenter a choice in the value of thermal flux to be used.

Three hundred and eighty liters of heavy water would be required to fill the facility. At $15 per pound the cost of heavy water would be $14,000. The aluminum box and the bismuth shield would cost approximately $10,000 to construct.

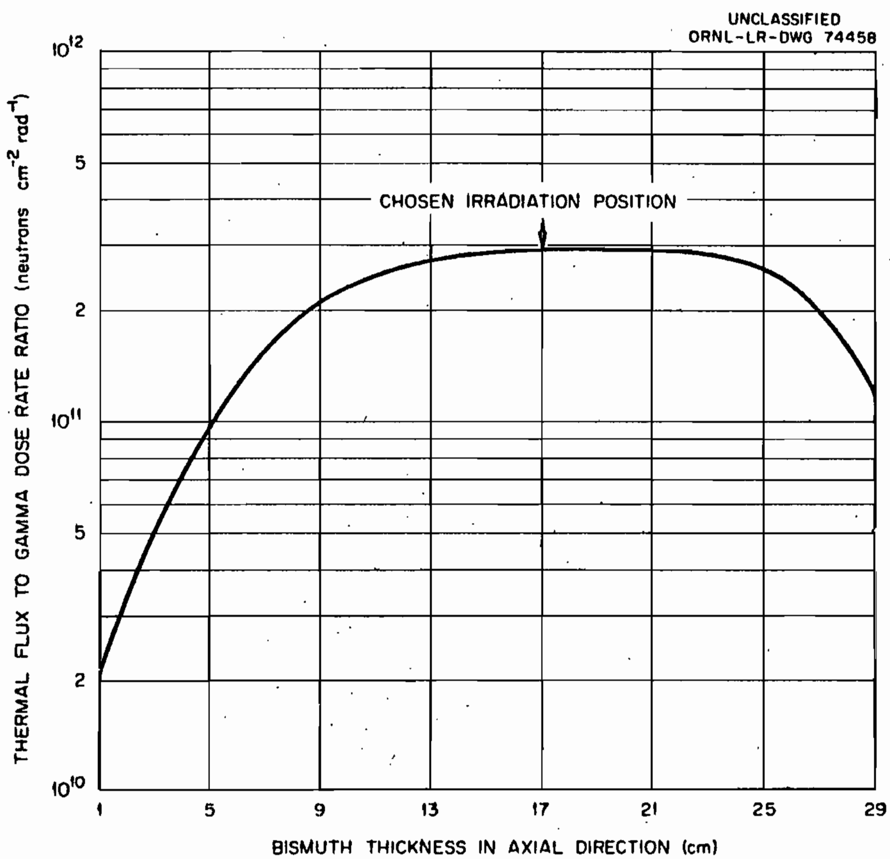

Figure 8. Ratio of Thermal Neutron Flux to Gamma Dose Rate at Various Positions in Bismuth.

REFERENCES

1. M. L. Tobias and T. B. Fowler, The Twenty Grand Program for the Numerical Solution of Few-Group Neutron Diffusion Equations in Two Dimensions, ORNL-3200 (1962).
2. M. L. Tobias, D. R. Vondy, and M. P. Lietzke, Nightmare--An IBM 7090 Code for the Calculation of Gamma Heating in Cylindrical Geometry, ORNL-3198 (1962).
3. "Modric--A One-Dimension Neutron Diffusion Code for the IBM-7090," To be published.
4. H. Etherington, Nuclear Engineering Handbook, p 12-107, McGraw-Hill, New York, 1956.
5. T. Rockwell, III, Reactor Shielding Design Manual, p 22, D. van Nostrand, New York, 1956.
6. H. Goldstein and J. E. Wiklins, Report NYO-3075, 1954.
7. F. Maienschein and T. Love, "Gamma-Ray Spectrum of the Bulk Shielding Reactor," Nucleonics. 12(5), 6 (May, 1954).

DISTIRUBTION

1. F. T. Binford
2. T. E. Cole
3-12. A. L. Colomb
13. J. A. Cox
14. E. P. Epler
15. T. B. Fowler
16. K. M. Henry
17. A. Hollaender
18. J. D. Kington
19. F. C. Maienschein
20. A. M. Perry
21. J. A. Swartout
22-31. R. van der Walt
32-41. R. C. von Borstel
42. D. R. Vondy
43. A. M. Weinberg
44-45. Central Research Library
46. Document Reference Section
47-49. Laboratory Records (LRD)
50. Laboratory Records--Record Copy
51. ORNL Patent Office
52-66. Division of Technical Information Extension
67. Research and Development Division (ORO)