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QUARTERLY PROGRESS REPORT

FOR PERIOD ENDING DECEMBER 20, 1952

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Please make the following corrections in your copy of ORNL-1477, Physics Division Quarterly Progress Report for Period Ending December 20, 1952:

Fig. 1.6, page 7 -- Substitute the attached figure.

Figs. 1.9 and 1.10, pages 10 and 11 -- (1) In Configurations 1, 2, and 3 delete the term "cm of" in every case. As is explained on text page 4,

Configuration 1: 11 Fe plus 10 H_2O

indicates that there were 11 evenly spaced 2.22 cm thick iron slabs separated by 10 nearly equal 3.05-cm layers of water. In Configurations 2 and 3 the number of slabs used was reduced.

(2) In Fig. 1.9 substitute the number "7" for the number "4" in Configuration 2. In Fig. 1.10 substitute "7" for "2." That is, in both cases, it should read

Configuration 2: 7 Fe plus 6 H_2O

A. H. Snell, Director
Physics Division

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PHYSICS DIVISION

QUARTERLY PROGRESS REPORT

for Period Ending December 20, 1952

A. H. Snell, Director
E. O. Wollan, Associate Director

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PHYSICS DIVISION QUARTERLY PROGRESS REPORT

1. SHIELDING RESEARCH

E. P. Blizard J. L. Meem

Shielding research is being pursued in two distinct but contiguous channels. One approach is phenomenological and is aimed at gathering a catalogue of general data for design problems; the other is more basic and has the purpose of giving a better understanding of attenuation. Fitting into the first category is the determination of the effective removal cross sections of various elements and materials in order to specify the attenuation resulting from a slab of material introduced into a water shield. The measurement of fast-neutron spectra by means of the proton-recoil counter is typical of the work in the second category.

Design work is in progress on the new Tower Shielding Facility that will be used to measure full-scale models of divided shields for aircraft.

BULK SHIELDING REACTOR

J. L. Meem

R. G. Cochran E. B. Johnson
M. P. Haydon J. K. Leslie
K. M. Henry F. C. Maienschein
H. E. Hungerford G. M. McCammon
T. A. Love

Shielding Experiments. During the past quarter, the gamma ray spectral measurements on the divided-shield mockup have been completed,[1] and fast-neutron spectral measurements have been started.

A new set of air-scattering experiments with the reactor operating at 100 kw has been performed, and the data are now being analyzed. Irradiation of monkeys for the study of eye cataracts is continuing during weekends.

Details of the above experiments are included in the current ANP quarterly reports.

Determination of the Energy per Fission (J. L. Meem, L. B. Holland, G. M. McCammon). As mentioned in the last quarterly report,[2] a special fuel element was constructed for a determination of the energy per fission. All the experimental data have been obtained, and the results are now being analyzed. By using the fission foil "catcher" technique, the fission rate per gram at the midpoint of the special element has been determined.

The first step was to calibrate the fission foils in a known thermal-neutron flux. The flux in the standard pile[3] was too weak to give a good counting rate on the fission foils; so a 30-curie polonium-beryllium source was put in a 6- by 6- by 10-ft graphite stack (the so-called "Pinky" pile). The neutron flux in the Pinky pile was calibrated against that in the standard pile through the use of indium foils.

In this known thermal neutron flux, it was calculated that the fission rate in the uranium foils should be 3.625×10^4 fissions/sec·gram. The mean thermal-neutron temperature in the Pinky pile, the self absorption of the uranium, and the cadmium ratio were taken into account.

A number of 10-hr exposures were made with the fission foils in the Pinky pile and, after a wait time of 30 min, the activity was observed on a scintillation counter of known geometry. The fission foils were then exposed for an identical time in the reactor at a power of 0.1 watt, and the same counting technique was used. The activity of the fission foils in the Pinky pile was 0.01599 times that in the reactor when it was operating at a power of 0.1 watt. The results were reproducible to within 1%. At the specified reactor power, the fission rate at the center of the special element was therefore

$$\frac{3.625 \times 10^4}{0.01599} = 2.267 \times 10^6 \frac{\text{fissions}}{\text{sec}\cdot\text{g}} .$$

By using gold foils, the thermal-neutron flux at this point in the reactor has been determined to be

(1) F. C. Maienschein and T. A. Love, *Gamma-ray Spectral Measurements with the Divided Shield Mockup, Part IV*, ORNL CF-52-11-124 (Nov. 17, 1952).

(2) J. L. Meem *et al.*, *Phys. Quar. Prog. Rep. Sept. 20, 1952*, ORNL-1421, p. 4.

(3) E. D. Klema, R. H. Ritchie, and G. McCammon, *Recalibration of the X-10 Standard Graphite Pile*, ORNL-1398 (Oct. 17, 1952).

$$1.67 \times 10^6 \frac{\text{neutrons}^{(4)}}{\text{cm}^2 \cdot \text{sec}} .$$

Accordingly, the fission rate should be

$$1.67 \times 10^6 \text{ cm}^{-2}\text{sec}^{-1} \times 545 \times 10^{-24} \text{ cm}^2 \times 6.03 \times 10^{23} \times \frac{1}{235} = 2.335 \times 10^6 \frac{\text{fissions}}{\text{sec} \cdot \text{g}} .$$

This agreement is very gratifying since the calculation of the fission rate in the reactor from the thermal flux as determined by gold foils has always been subject to uncertainties such as the mean neutron temperature in the reactor, the number of fissions caused by neutrons above thermal energies, and the self absorption of the gold foils.

Further calculations toward the determination of the energy released per fission are continuing.

Proton-Recoil Type Fast Neutron Spectrometer (R. G. Cochran, K. M. Henry). The development of a triple coincidence type of proton-recoil spectrometer at the Bulk Shielding Facility is essentially complete. The instrument is composed of a triple-section proportional counter with suitable radiator-absorber wheels, amplifiers, electronic pulse-height selectors, and a triple coincidence circuit (Fig. 1.1). Sufficient energy resolution is obtained for shielding work. Considerable effort has been made to obtain maximum sensitivity by optimizing the available parameters.

The spectrometer energy and intensity calculations have been checked with known neutron energies from the ORNL Van de Graaff generator, and excellent agreement was obtained (Fig. 1.2). Several polonium-beryllium and polonium-boron

(4) J. L. Meem and E. B. Johnson, *Determination of the Power Bulk Shielding Reactor, Part II, Neutron Flux Measurements in Several Beryllium-Oxide Reflected Assemblies*, ORNL-1438 (to be published).

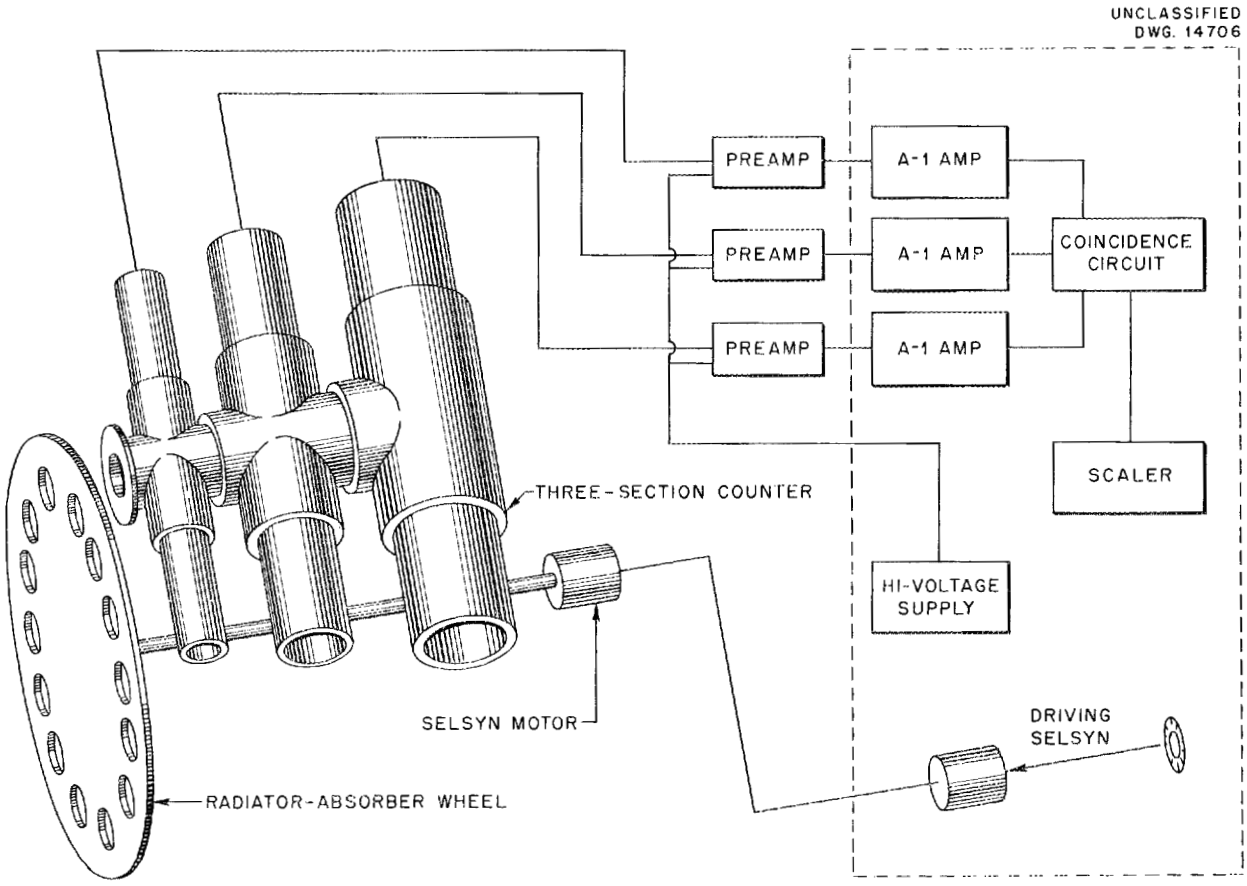

Fig. 1.1. Proton Recoil Neutron Spectrometer.

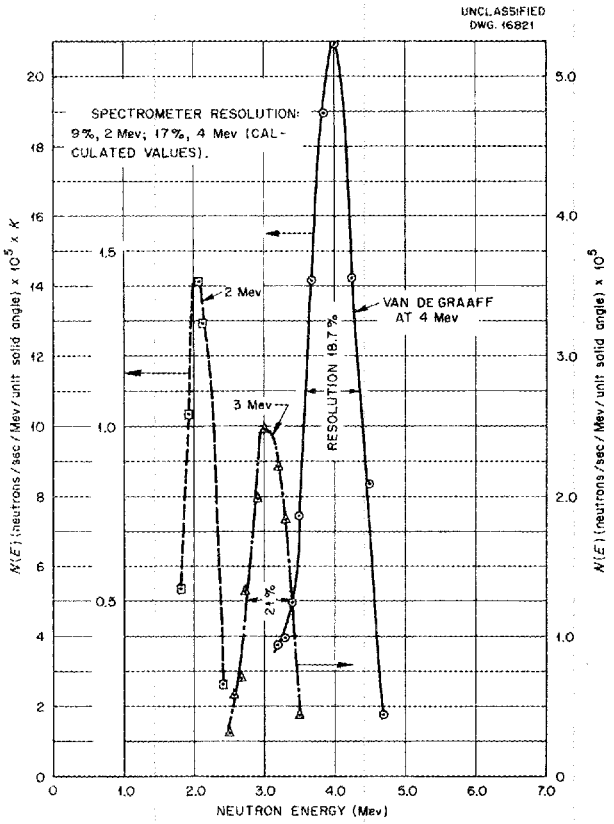

Fig. 1.2. Fast Neutron Spectrometer Data.

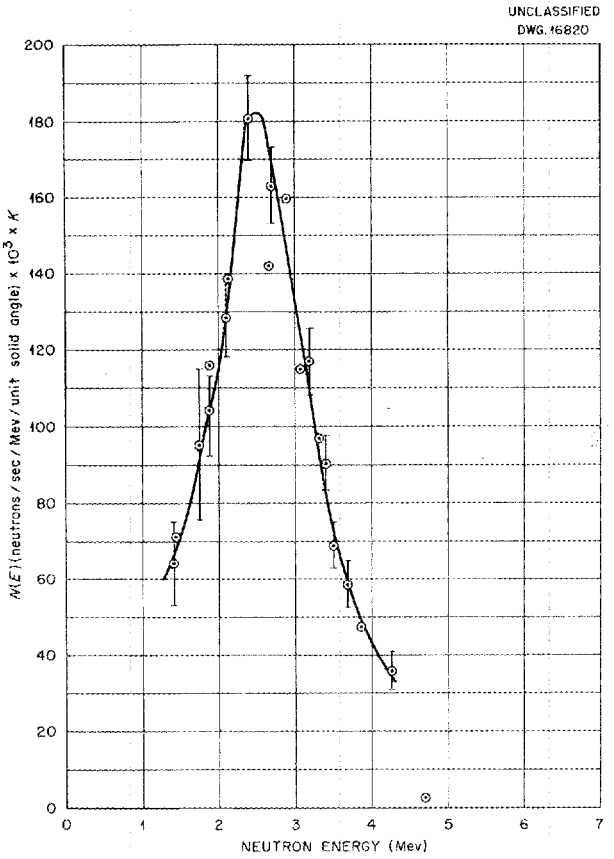

Fig. 1.3. Polonium-Boron Spectrum of Source PB-281. Data taken with BSF Fast Neutron Spectrometer.

source spectra have been measured with the spectrometer. Figure 1.3 is an example of the type of data being obtained. A complete report on this instrument is in preparation.

Other Experiments. Because of other work of higher priority, no progress has been made on measuring capture gamma rays by use of the thermal column. The experiment to determine the O^{16} and O^{17} activation in water passing through the reactor has been performed, but some additional experimental data on the fast-neutron flux in the reactor are needed before final calculations can be made. This experiment will be continued as soon as reactor time is available.

TOWER SHIELDING FACILITY

C. E. Clifford T. V. Blosser
J. Y. Estabrook

The design of the tower is in the hands of the architect engineers, Knappen, Tippetts, Abbett and McCarthy, of New York, who are charged with translating the ideas of the Laboratory into engineering-wise, feasible structures. The present plan is to use six separate hoists so that the position of cockpit shield and reactor can be adjusted in any of several degrees of freedom. The choice of reactor controls is being made primarily from the aspect of safety because of the unusual location of the reactor with respect to the operator, who will be in a nearby underground building.

LID TANK FACILITY

J. D. Flynn J. M. Miller
G. T. Chapman F. N. Watson

During the past three months, work has been continuing on the measurement of the effective removal cross sections of various elements and materials. This work will continue on new materials as time between major experiments permits.

The most recent work completed in the Lid Tank Facility is the mockup of the CP-6 iron-water thermal shield for Du Pont. Work is now starting on a repeat mockup of the SIR top plug experiment in which the iron will be replaced with type 347 stainless steel.

Effective Removal Cross Sections. In a predominantly hydrogen shield, it is possible, with good accuracy, to represent the effect on the fast-neutron attenuation of other elements in the shield by a simple exponential, just as if the only effect were one of absorption. The cross section that best fits the experimental data is called "the effective removal cross section." In order to avoid transition effects resulting from spectral changes, the extraneous element is always introduced near the source, and the effect is observed at large water thicknesses.

The experimental data from which the cross sections are determined are two sets of neutron measurements, either with a thermal detector[5] or with a neutron dosimeter. The first set of measurements is taken in plain water along the center line of the source, and the second set is a repeat of the first after a not-too-thick slab of the material to be tested has been placed in the water close to the source.

The tungsten was supported in a Unichrome plastic matrix that was cast in a set of iron frames. The background measurement, corresponding to the pure water measurement, was obtained by means of a dummy setup in which the plastic matrix contained no tungsten powder.

The effective neutron removal cross section was obtained by solution of[6]

$$\frac{D_1(z)}{D_2(z+t)} = \frac{\lambda_1}{\lambda_2}\left(\frac{z+t}{z}\right)^2 \left(1 - \frac{a^2 t}{4\lambda z(z+t) - a^2 z}\right) e^{\Sigma t} ,$$

where:

$D_1(z)$ and $D_2(z+t)$ are the two neutron measurements,

z is the thickness of water between source and detector,

t is the thickness of sample,

λ_1, λ_2 are relaxation lengths of the two sets of data at z and $z+t$, respectively,

λ is an average of λ_1 and λ_2,

a is the source radius (35.6 cm), and

Σ is the macroscopic effective removal cross section.

The thermal data obtained thus far are shown in Figs. 1.4 through 1.8, and the corresponding cross sections are given in Table 1.1.

TABLE 1.1. EFFECTIVE FAST-NEUTRON REMOVAL CROSS SECTIONS

MATERIAL	CROSS SECTION (barns/atom)
Aluminum	1.19
Tungsten	3.08
Beryllium	1.12
Lithium fluoride	1.40
Carbon (AGHT)	0.84

CP-6 Thermal Shield. The CP-6 reactors are shielded by concrete, the inside layer of which is protected from excess heating by an iron-water thermal shield. The large investment involved made it essential to test the Du Pont design to determine the adequacy of this protection. Consequently, a mockup that has nearly the same volume ratio of iron to water was installed in the Lid Tank Facility, and iron slabs on hand were used. Although the mockup actually had about 5% excess water over that called for in the Du Pont design, the results will be quite applicable to the problem.

There was 0.9 cm of water between the source and the first iron plate. Following this were 11 evenly spaced 2.22-cm-thick iron slabs, separated by 10 nearly equal 3.05-cm layers of water. The total distance from the front of the tank to the back of the last iron slab was thus 55.8 cm. The gamma dose in mr/hr in the water behind this configuration was then measured (Fig. 1.9), as was the thermal flux in the same position (Fig. 1.10). All measurements were made on the axis of the 28-in.-dia source.

In order to extend the experiment to one that was more generally applicable, the shield thickness was reduced in steps, and the effect on the radiation was measured each time the thickness was reduced.

[5] In a thick water shield, the thermal neutrons are a good indication of the fast flux in the vicinity; see E. P. Blizard, *Introduction to Shield Design, Part II*, ORNL CF-51-10-70 (March 7, 1952), p. 107.

[6] Derivation by E. P. Blizard.

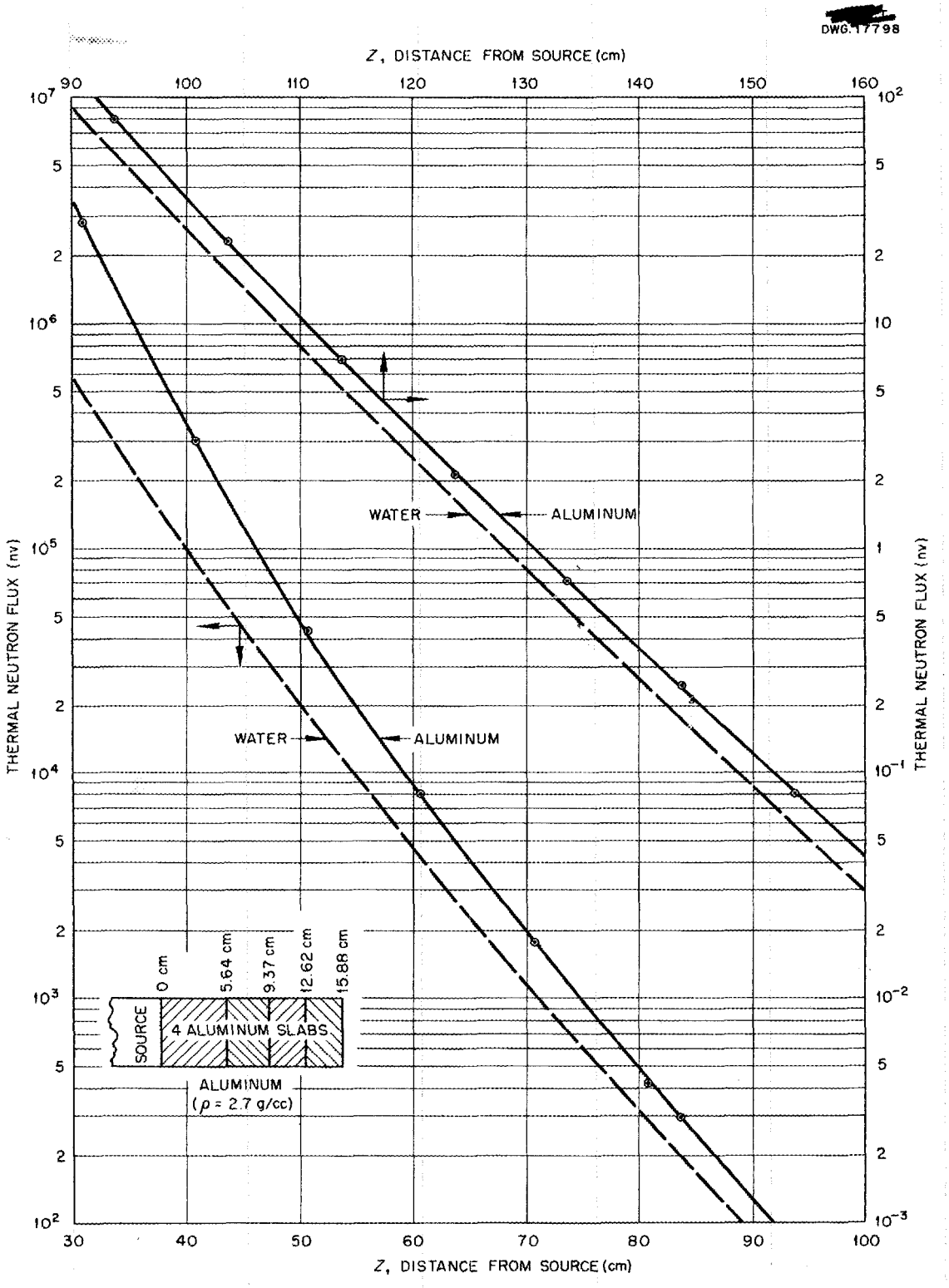

Fig. 1.4. Thermal Neutron Distribution beyond Aluminum.

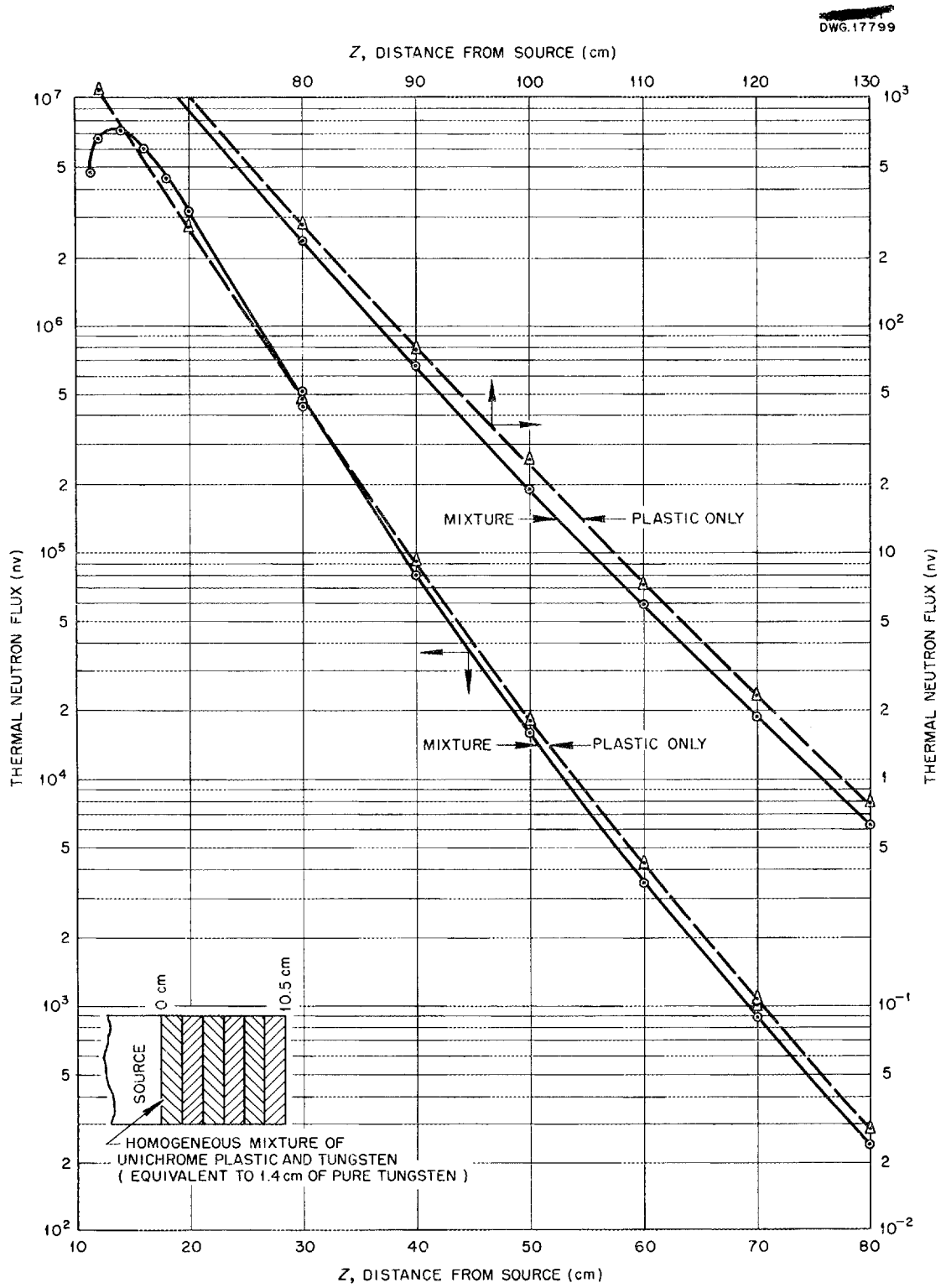

Fig. 1.5. Thermal Neutron Distribution beyond Tungsten.

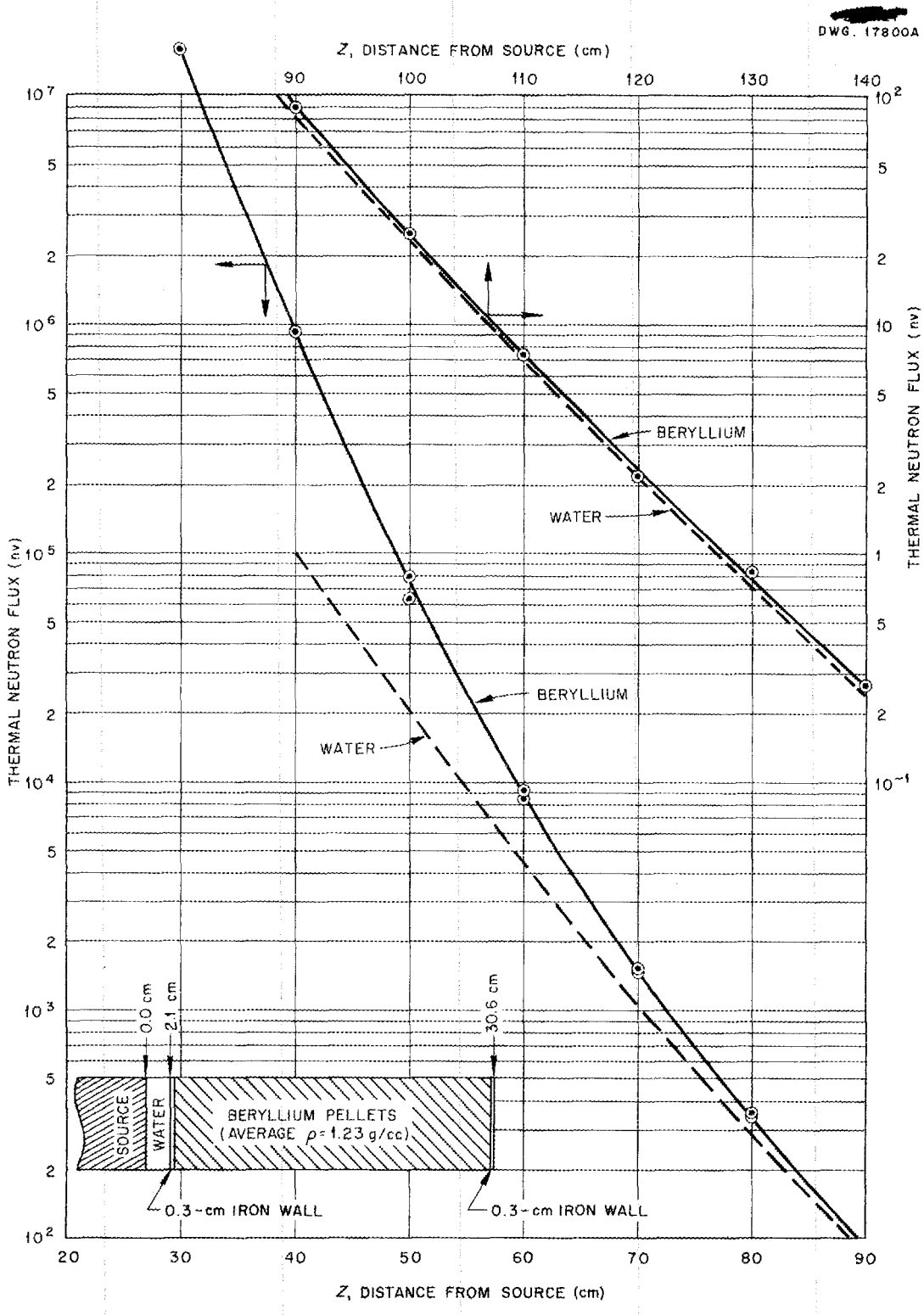

Fig. 1.6. Thermal Neutron Distribution Beyond Beryllium.

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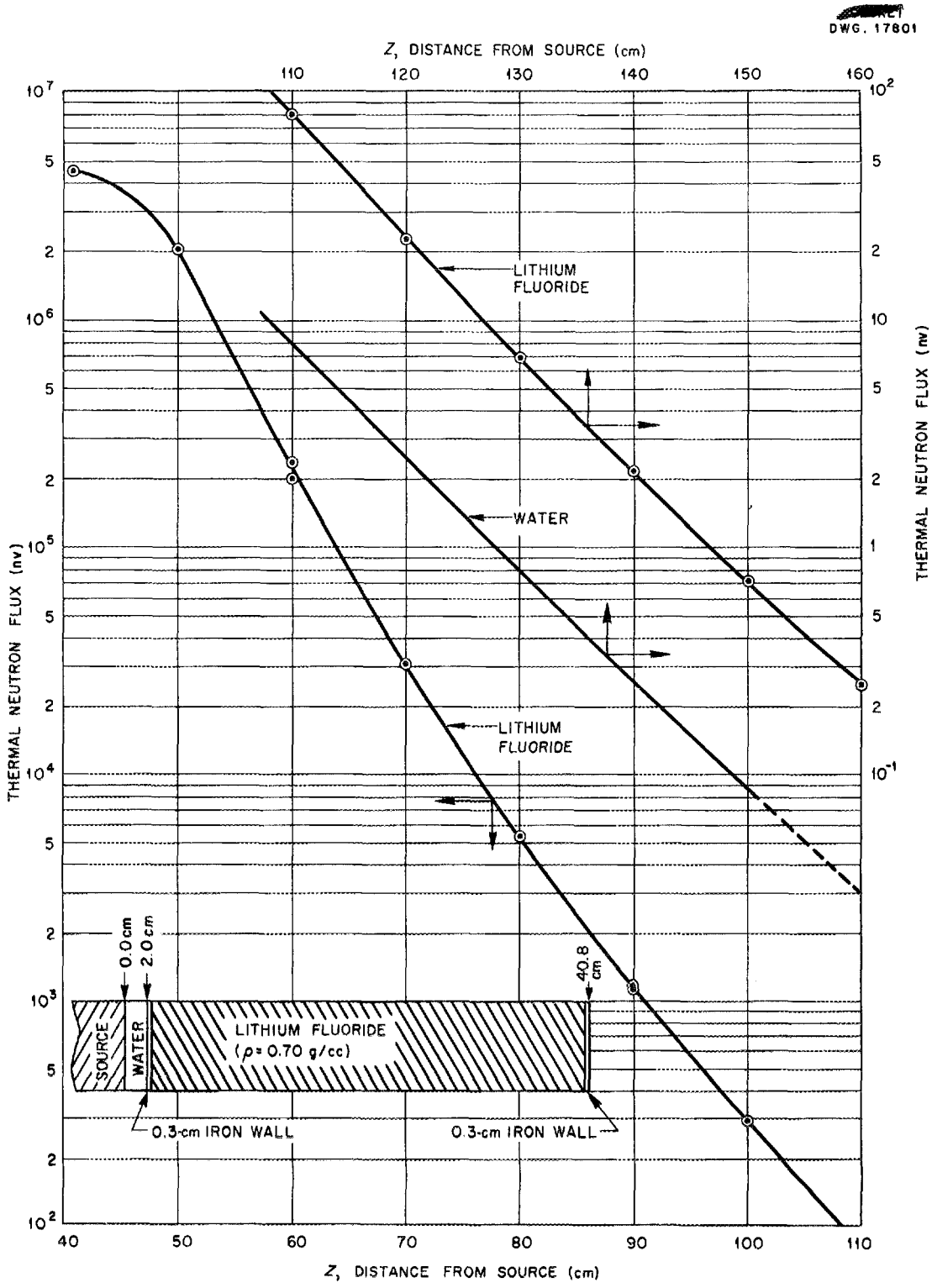

Fig. 1.7. Thermal Neutron Distribution beyond Lithium Fluoride.

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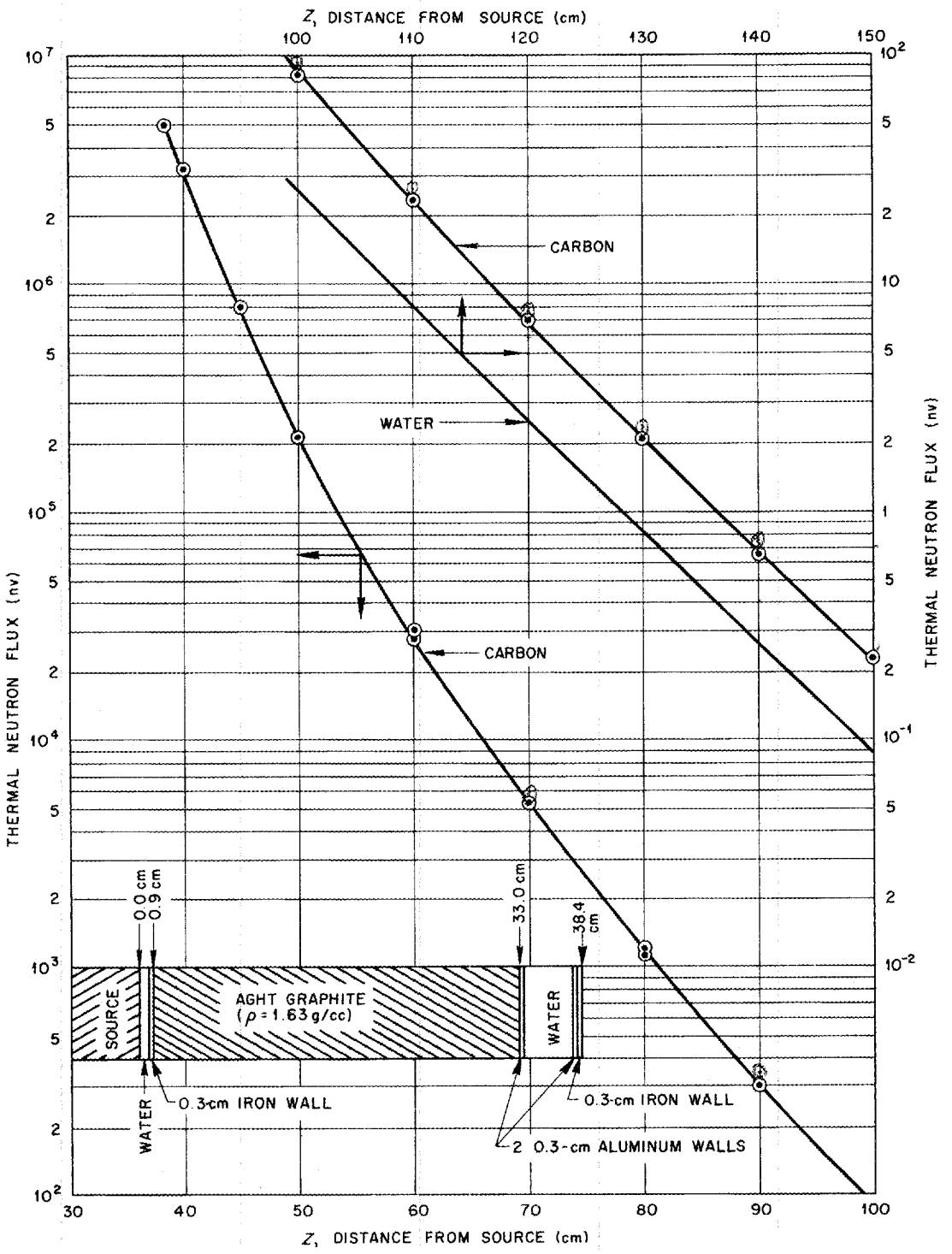

Fig. 1.8. Thermal Neutron Distribution beyond Carbon.

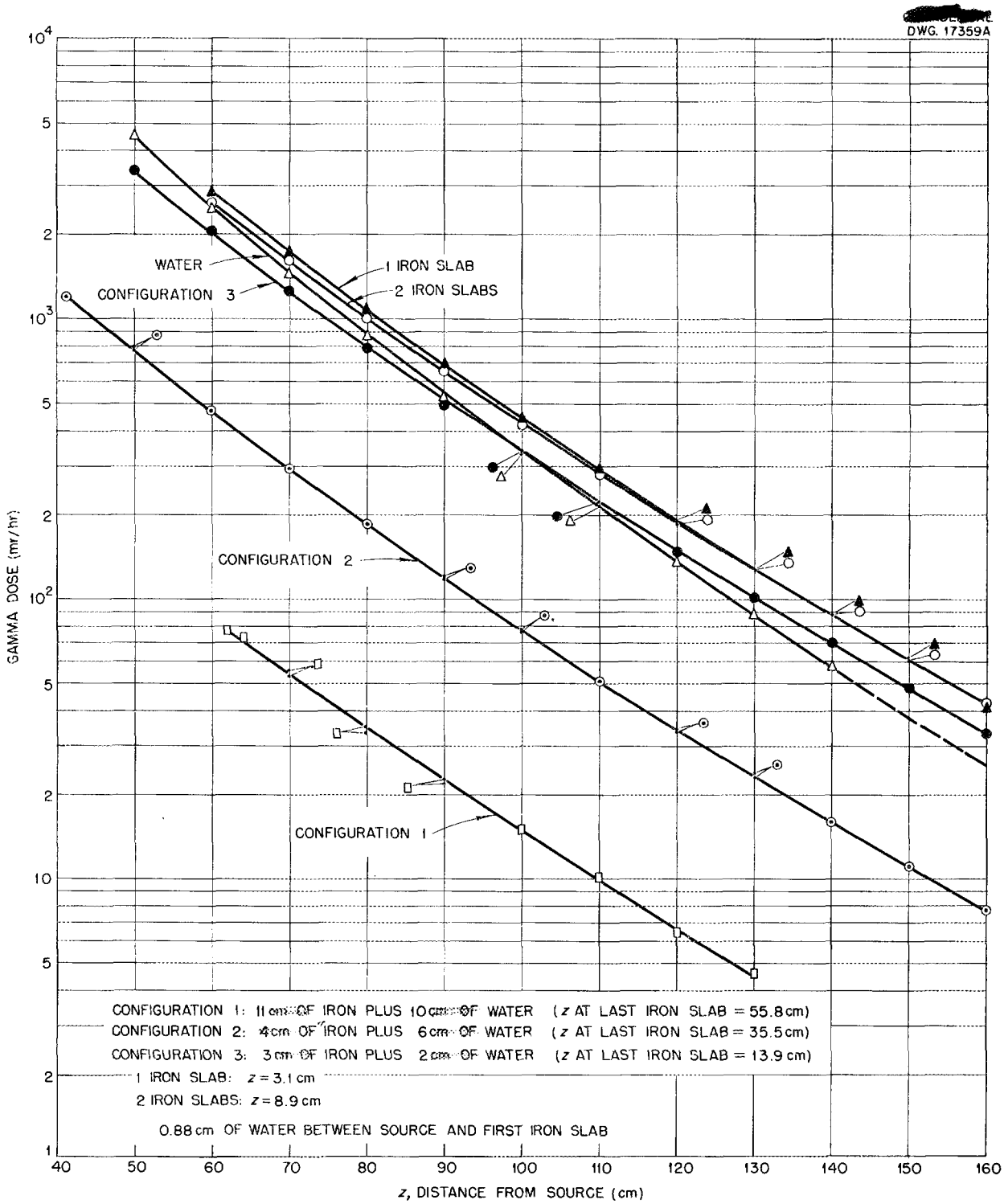

Fig. 1.9. Gamma Dose Rate in Water beyond Iron-Water Thermal Shield.

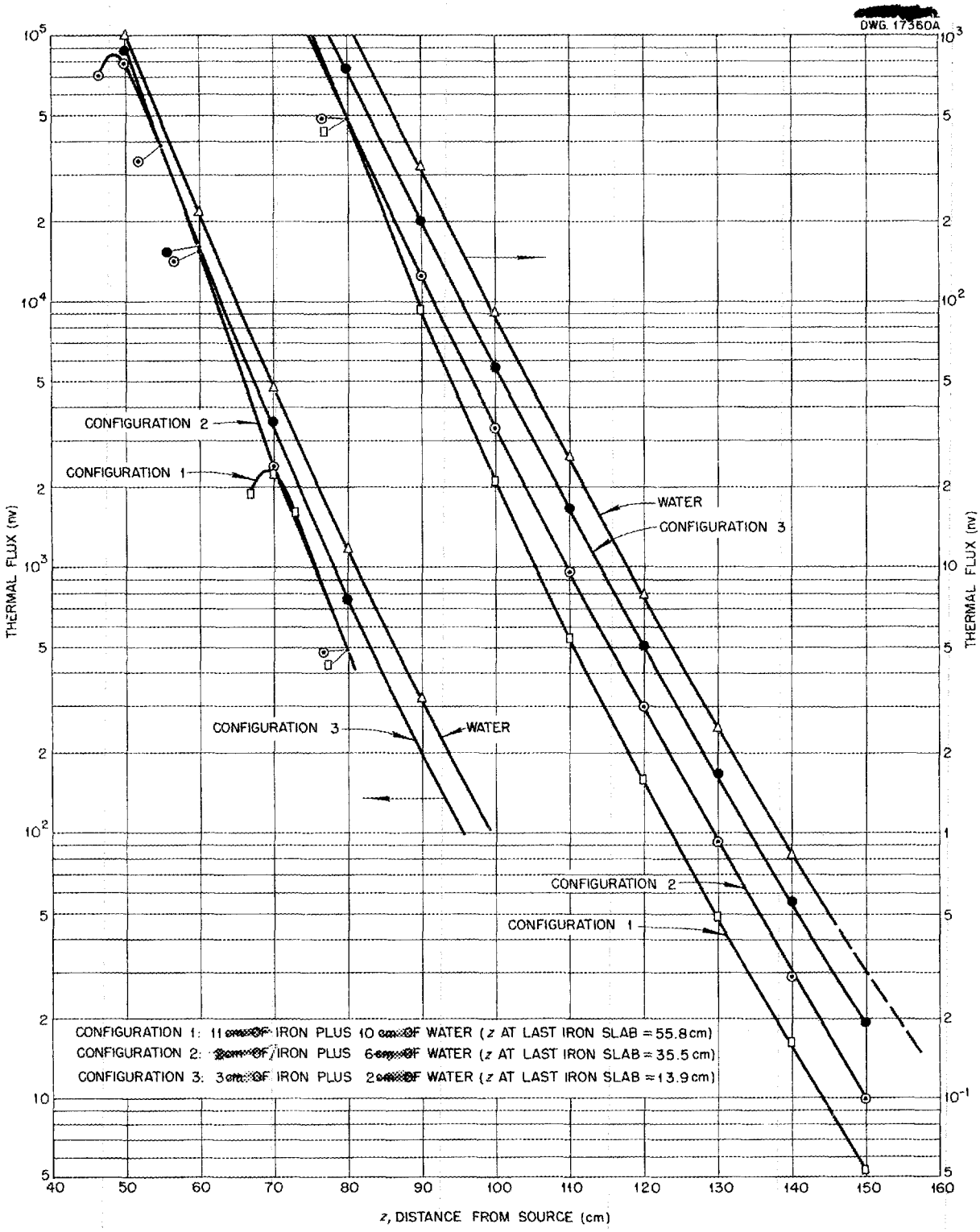

Fig. 1.10. Thermal Flux in Water beyond Iron-Water Thermal Shield.

2. BASIC REACTOR STUDIES

A. D. Callihan

The program of basic reactor studies of the Physics Division covers two topics, each concerned with chain-reacting assemblies of fissionable materials. One topic is intended, primarily, to obtain data on which design criteria for fissionable-material processing plants can be based; in the other, investigations are made of the nuclear characteristics of proposed reactors.

CRITICAL EXPERIMENTS

D. F. Cronin — L. W. Gilley
J. K. Fox — J. W. Morfitt[1]
R. A. Francis, Jr. — E. R. Rohrer

Uranium-235. The first part of the study[2] of the critical conditions of uranium enriched to 5% U^{235} has been completed. In these experiments, U_3O_8 has been mixed with increasing amounts of an organic compound to provide a moderator. The hydrogen content has been varied from zero to an $H:U^{235}$ atomic ratio of 750, and determinations of critical masses and volumes have been made throughout that range. The minimum mass thus far measured is 3.5 kg of U^{235}, which occurred, with a complete water reflector, at an $H:U^{235}$ ratio of about 350. It should be pointed out that these critical assemblies consist of the oxide mixture packed in essentially cubical arrays of aluminum boxes. The structural materials thus built into the assemblies make the measured values of the critical masses greater than they would be if the uranium and hydrogeneous materials were homogeneous.

The minimum mass measured in the same manner, but without the reflector, was 6.5 kg. The minimum volumes observed were about 78 and 175 liters, with and without reflector, respectively, and occurred at an $H:U^{235}$ ratio of 300.

It is obvious that with 5%-U^{235}-enriched uranium, it is impossible to prepare an aqueous solution at an $H:U^{235}$ ratio of less than about 500. In order to evaluate the error in the above quantities that is due to inhomogeneities and structural materials, the uranium is being converted into UO_2F_2, from which an aqueous solution will be made for additional studies at the higher moderations.

The results thus far also indicate that an infinitely long 10-in.-dia cylinder or a slab 6 in. thick of infinite extent cannot be made critical with 5%-enriched U^{235} of the density used in these experiments.

Uranium-233. A limited quantity of almost pure U^{233} has been available as an aqueous solution of uranyl nitrate for some critical experiments. The test system was described in a previous report.[3] Measurements have been made with solution concentrations varying from $H:U^{233}$ ratios of 40 to 750 in paraffin-reflected cylinders that have diameters of 5 to 12 inches. The minimum critical mass observed is 600 g at an $H:U^{233}$ ratio of 400. A sphere 10 in. in diameter has been fabricated, and one comparison of the critical mass in spherical vs. cylindrical geometry showed the former to be smaller by about 5%.

The U^{233} is now being converted to the oxyfluoride in order to evaluate the nitrate poison and to enable a direct comparison of the nuclear properties of U^{233} and U^{235}.

REACTOR STUDIES

Homogeneous Reactors (D. F. Cronin, L. W. Gilley, J. A. Marable, E. R. Rohrer, J. W. Morfitt,[1] R. B. Gallaher[4]). In cooperation with the HRP, equipment for the study of the nuclear stability of aqueous slurries of enriched uranium oxide has been designed and constructed. This equipment consists of a loop that contains a bank of 5-in.-dia storage cylinders and a pump through which the slurry will be continuously circulated. A lead from the discharge of the pump will connect to a cylindrical reactor vessel, either 10 or 12 in. in diameter. These vessels are equipped with 3-in. quick-opening valves for the drainage of the slurry in the event of an inadvertent high-radiation

[1] Y-12 personnel.

[2] J. W. Morfitt, J. K. Fox, D. F. Cronin, E. R. Rohrer, and P. E. Wilkinson, *Phys. Quar. Prog. Rep. Dec. 20, 1952*, ORNL-1216, p. 24.

[3] D. F. Cronin, J. K. Fox, J. W. Morfitt, and E. R. Rohrer, *Phys. Quar. Prog. Rep. June 20, 1952*, ORNL-1374, p. 8.

[4] HRP personnel.

level. A stirrer, driven by a variable speed motor, is located in the cylinder. The apparatus has been installed and is being leak and pressure tested. It is proposed to bring the system to critical while the slurry is well stirred and then to note reactivity changes as the agitation is reduced and settling begins.

Aircraft Reactor Studies (D. Scott,[1] D. V. P. Williams, E. L. Zimmerman, R. C. Keen, J. J. Lynn). A series of experiments at room temperature has been performed on a mockup of the ARE. The reactor is to have beryllium oxide as the reflector and moderator and a fluid mixture of several fluorides, including UF_4, at high temperature as the fuel and the primary heat transfer medium. In the mockup reported here, the reflector and moderator were beryllium oxide in the form of a right cylinder 36 in. in height and 47 in. in diameter. The fuel was a powder mixture of enriched UF_4, ZrO_2, C, and NaF packed in $1\frac{1}{4}$-in.-dia stainless steel tubes. These tubes, in turn, were spaced in vertical holes in the beryllium oxide. In the first loading, the fuel density was 0.163 g of U^{235} per cc, and 61 tubes that contained 5.8 kg of U^{235} were required for criticality. In a second loading, the density was 0.214 g of U^{235} per cc, and 5.2 kg in 42 tubes was critical. The latter loading permitted the addition of sufficient excess fuel and reactivity to calibrate some of the control and safety rods that have been fabricated for the ARE reactor. These tests have shown the rods to be adequate for their designed use.

Detailed comparisons have been made of the importance of increments of fuel and of various structural materials at a number of positions in the reactor. These data have been reported in detail elsewhere.[5] In another type of reactor that has received preliminary consideration, the fluid fuel flows in an approximately spherical shell between a neutron-reflecting island and a conventional external reflector. In one model that has been examined, the central island is a 12-in. cube of metallic beryllium, the fuel is a 3-in.-thick lattice of uranium and sodium metals, and the surrounding reflector is a 12-in.-thick layer of beryllium backed by graphite. The system has been made critical with about 15 kg of U^{235}. Some neutron-flux and fission-rate distributions have been measured that show the expected high thermal-neutron flux in the central beryllium. About 70% of the fissions is caused by thermal neutrons. Details of these preliminary measurements have been published elsewhere.[6]

[5] D. Scott and C. B. Mills, *ANP Quar. Prog. Rep. Sept. 10, 1952*, ORNL-1375, p. 43; D. Scott, C. B. Mills, J. F. Ellis, and D. V. P. Williams, *ANP Quar. Prog. Rep. Dec. 10, 1952*, ORNL-1439, p. 52.

[6] D. V. P. Williams, J. J. Lynn, R. C. Keen, D. Scott, and C. B. Mills, *ANP Quar. Prog. Rep. Dec. 10, 1952*, ORNL-1439, p. 48.

3. TIME-OF-FLIGHT SPECTROMETER

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The transmission of samples of U_3O_8, enriched in U^{236}, and of the oxide of depleted uranium has been observed, and the results are presented in Figs. 3.1 and 3.2.

Uranium-236. A mass analysis supplied with the sample of U_3O_8, enriched in U^{236}, showed the following isotopic composition:

U^{234}	0.05%
U^{235}	3.09%
U^{236}	96.65%
U^{238}	0.21%

The transmission of the 0.602 g/cm^2 sample was first measured with a resolution of 0.71 μsec/meter in the energy range 1.6 to 1000 ev. The only resonance apparent was at 5.27 ev. During observations of the "sample in" counting rate in the energy range 0.3 to 5.0 ev, no strong resonances were seen. The energy range of 4.0 to 50 ev was then examined with a resolution of 0.5 μsec/meter. The results of both runs are shown in Fig. 3.1.

A thicker sample is now on hand and is being run in a search for the next higher energy levels and to evaluate the strength of the 5.3-ev level.

Uranium-238. The transmission of a 6.87 g/cm^2 sample of U_3O_8, depleted to less than 7 ppm of U^{235}, is shown in Fig. 3.2. The energy range of 2.0 to 5000 ev was covered, with a resolution of 0.5 μsec/meter. There is a possibility that the weak level at 5.14 ev is due to an impurity; therefore the sample is being analyzed. There are resonances evident at 5.1, 6.65, 20.4, 36.5, 67.1, and 110 ev, and probably at least two at 184 and 244 ev.

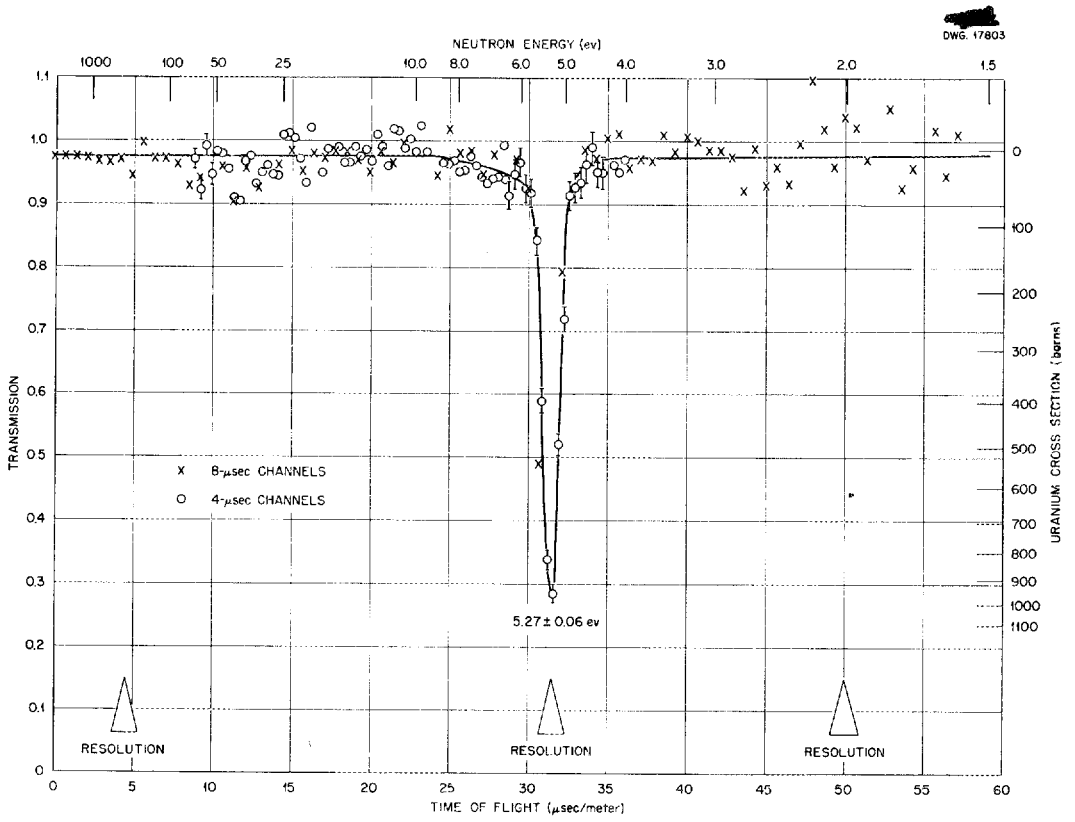

Fig. 3.1. Transmission of U^{236}. Enriched $U_3^{236}O_8$ – 0.602 g/cm^2.

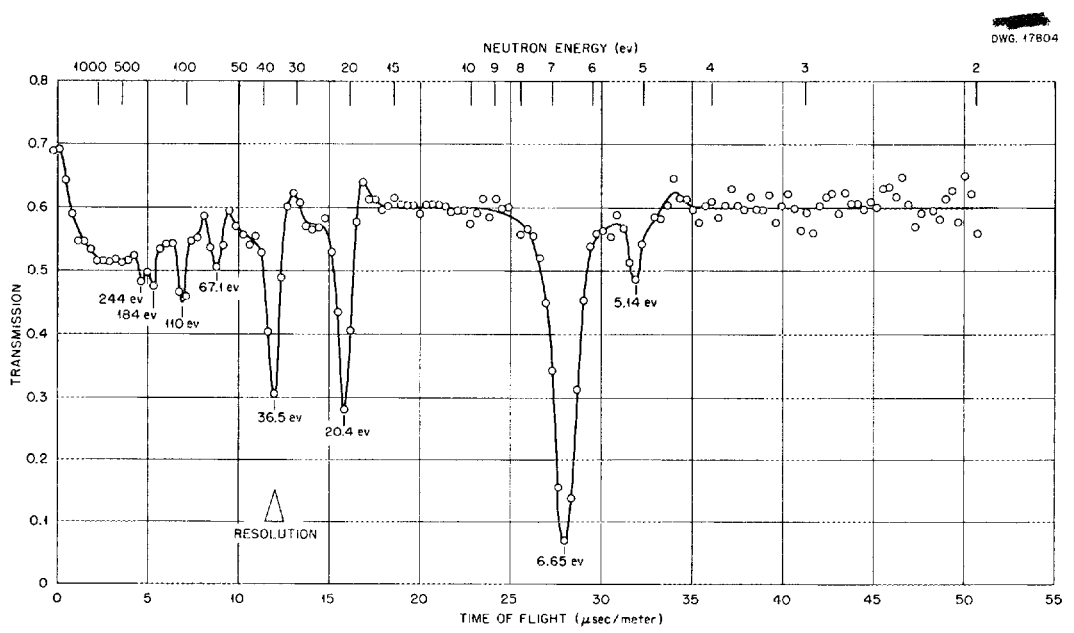

Fig. 3.2. Transmission of U^{238}. Depleted U_3O_8 – 6.87 g/cm^2.