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SEMIANNUAL PROGRESS REPORT

FOR PERIOD ENDING MARCH 10, 1954

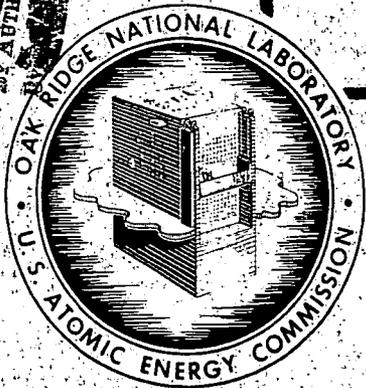
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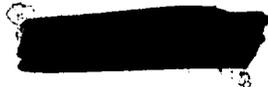
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for Period Ending March 10, 1954

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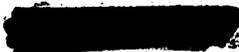
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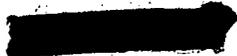
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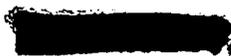
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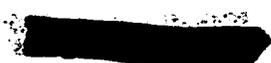


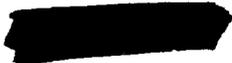
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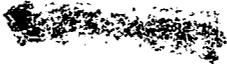
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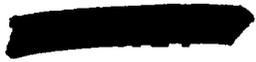


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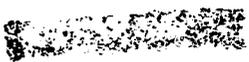
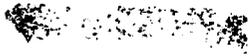
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## PREFACE

In general, the classified work of the Physics Division is carried out in support of applied projects having to do mainly with reactor technology. Since much of the reactor shielding research is carried out for the ANP Project, it is reported in the quarterly publications of that project. Where the results are also of general reactor interest, they are additionally reported in this publication in a somewhat briefer form. Some of the shielding research not directly applicable to ANP, and not supported by ANP funds, is reported exclusively here. Similarly, the work of the Critical Experiments Group is, in general, reported more fully in the ANP quarterly reports and elsewhere, and the Semiannual Report will be the primary vehicle for presenting the results of basic investigations.



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## PHYSICS DIVISION SEMIANNUAL PROGRESS REPORT

### SUMMARY

#### SHIELDING RESEARCH

In the general search for analytic calculational methods for large attenuations, the continuum theory of nuclear interactions has been used to calculate, rather satisfactorily, the effective neutron removal cross sections for a number of elements. In addition, a number of short-time projects have been carried out in support of the experimental program; one of these was an investigation of the fast-neutron albedo of water, a value which is of interest in scattering calculations as well as duct theory. Three other calculations are reported.

At the Lid Tank Shielding Facility a series of experiments designed to study the penetration of fast neutrons through slant thicknesses of a material have been initiated; tentative results indicate that most of the dose comes from neutrons penetrating the shield along the slant (undeviated) path. Air-duct experimentation has also been resumed, and a new method of inserting a boron layer in a shield has been investigated. Removal-cross-section measurements have been continued.

With the exception of some intercomparison of instruments, all the work at the Bulk Shielding Facility has been for the ANP Project. A summary of gamma-ray and neutron spectral measurements is given.

The Tower Shielding Facility has been completed and the first experiment has been started. A tentative experimental program has been outlined.

#### CRITICAL EXPERIMENTS

Basic reactor studies at the Critical Experiments Facility included measurement of the value of  $\eta$  for  $U^{233}$  of 2.33 over the energy range of 0.026 to 0.032 ev; the observed decrease in  $\eta$  over this energy range was approximately 0.01. The reac-

tivity temperature coefficients of  $U^{233}$  and  $U^{235}$  were also measured. In addition, a number of experiments have been performed to determine the critical parameters of aqueous solutions of  $U^{233}$ , the effect of boron poisoning in a critical slab of aqueous  $UO_2F_2$  enriched with  $U^{235}$ , the minimum number of uranium-aluminum alloy slugs required for criticality as a function of lattice cell spacing, and the multiplication of neutrons in a subcritical solution of  $UF_6$  enriched with  $U^{235}$  such as might be encountered in adjacent process tubes.

Specific aircraft-reactor studies have included experiments on preliminary assemblies of the GE-ANP AC-1, GE-ANP AC-100-A, Reflector-Moderated, and Supercritical Water Reactors. Estimates of the critical mass for both these arrangements were considerably lower than the measured values. Further experimentation is planned on the Reflector-Moderated Reactor to provide more fundamental information, which should also aid in evaluating previous critical assemblies. For the criticality tests on the Supercritical Water Reactor, designed core dimensions and core composition are being approximated by furfural ( $C_5H_4O_2$ ) liquid surrounding stainless steel tubes filled with an aqueous  $UO_2F_2$  solution.

#### HIGH-VOLTAGE PHYSICS

An investigation has been made of the number of prompt neutrons per fission as a function of the energy of the neutrons which produce the fission over the energy range of 0 to 5 Mev, with an additional measurement at 14 Mev.

#### NEUTRON TIME-OF-FLIGHT SPECTROMETER

The measurements of the total cross section of  $Xe^{135}$  have been extended to energies as low as 0.01 ev.

## 1. SHIELDING RESEARCH

E. P. Blizzard

## SHIELDING ANALYSIS

E. P. Blizzard      F. H. Murray  
J. E. Faulkner      H. E. Stern<sup>1</sup>

## Some Estimates of Removal Cross Sections Based on the Continuum Theory of the Scattering of Neutrons from Nuclei

F. H. Murray

In a previous progress report,<sup>2</sup> the asymptotic character of the attenuation of neutrons through large thicknesses of material, as calculated using the continuum theory, was expressed in the form  $\exp(-\sigma x/\lambda)$  where  $\lambda$  is the largest eigenvalue of an infinite matrix. The results can be applied to the calculation of the neutron removal cross section of a material between a fission source and a large thickness of water. The details of such an application have been published in a recent report.<sup>3</sup> Several of the calculated removal cross sections are compared with those determined from experimental measurements as follows:

Material	$\Sigma$ (barns/atom)	
	From Experiment	From Calculation
Al	1.31	1.23
Fe	1.95	1.95
Cu	2.08	2.21
Pb	3.4 (estimated)	3.36
Bi	3.43	3.32

The calculations were for 10 or 15 cm of material. For thin sections with small values of the product of removal cross section and thickness, the calculations required only the value of  $\sigma_{\text{tot}}/\lambda$  for the energy corresponding to the maximum uncollided flux corresponding to the given thickness of water.

<sup>1</sup>Consolidated Vultee Aircraft Corporation.

<sup>2</sup>F. H. Murray, *Phys. Semiann. Prog. Rep. Sept. 10, 1953*, ORNL-1630, p 6.

<sup>3</sup>F. H. Murray, *Some Estimates of Removal Cross Sections Based on the Continuum Theory of the Scattering of Neutrons from Nuclei*, ORNL CF-53-12-60 (Dec. 11, 1953); also reported in *ANP Quar. Prog. Rep. March 1, 1954*, ORNL-1692, p 115.

## Analysis of the Response Characteristics of the Fast-Neutron Dosimeter Integrating Circuit

J. E. Faulkner

A study<sup>4</sup> has been made of the response characteristics of the Hurst fast-neutron dosimeter,<sup>5</sup> with emphasis on the limits of the binary integrating circuit method.

It should be pointed out that the operation of the dosimeter cannot be handled in a routine manner and that the integrating circuit is not completely flexible. Errors of considerable magnitude can be introduced when the integrating circuit is used under conditions of wide variation in the measured spectrum from that of the calibrating spectrum. Most of the error in the binary-integrating-circuit method results from cutting off the lower energy pulses (at 6 v).

If it is assumed that the pulse-height distribution is of the form  $e^{-\lambda V}$ , where  $\lambda$  is the slope of the curve and  $V$  is the voltage, and if the slope<sup>6</sup> of the calibrating source is  $0.1 \text{ v}^{-1}$ , then the percentage error as a function of slope is as follows:

$\lambda$	Percentage Error
0.100	0
0.102	0.057
0.110	2.22
0.130	7.00
0.150	11.98
0.200	24.46

Although the pulse-height distribution does not vary appreciably for high-energy spectra, there is a rapid variation for spectra where the peak energy is below 1 Mev (i.e., beyond solid iron shields or other low-energy sources). Further, if the integrator is used in conjunction with the anthracene crystal to measure gamma dosages, great caution

<sup>4</sup>J. E. Faulkner, *Analysis of the Response Characteristics of the Hurst Fast-Neutron Dosimeter Integrating Circuit*, ORNL CF-54-4-17 (to be issued).

<sup>5</sup>F. M. Glass and G. S. Hurst, *Rev. Sci. Instr.* 23, 67 (1952).

<sup>6</sup>This is the approximate slope for three-chamber Lid Tank flow dosimeter for a Po-Be source.

must be used because wider variation is to be expected.

Recently, Hurst reported, in a personal communication, that provision has been made in the integrating circuit to extrapolate the integral to zero pulse size on the basis of the lowest pulses which are measured. The extrapolation is linear and should give a very good estimate of the actual distribution.

### Visible Light from a Nuclear Power Plant

J. E. Faulkner

The results of some experiments carried out at Los Alamos on the visible light from a polonium source<sup>7</sup> have been used to obtain an estimate of the ionizing radiation that would have to be present around the aircraft reactor shield to make it just visible. If the airplane were approximately 200 ft from the observer, the ionizing radiation near the shield would have to be about  $5 \times 10^6$  r/hr, whereas if the airplane were very far from the observer, say 50,000 ft, the ionizing radiation would have to be about  $10^9$  r/hr. These estimates will be compared with measurements made on the BSR.

### Neutron Reflection Coefficient for Water

H. E. Stern<sup>8</sup>

An estimate of the neutron reflection coefficient for water has been obtained by using a simple model having a coincident source and a nondirectional receiver located above the water surface. If isotropic oxygen scatter and attenuation governed by the removal cross sections are assumed, the reflected flux  $F$  at height  $b$  for source strength  $N_0$  is

$$F = \frac{N_0}{48\pi b^2} \left( \frac{\Sigma_{\text{scatter}}}{\Sigma_{\text{removal}}} \right).$$

The albedo approach gives

$$F = \frac{N_0 \alpha}{10\pi b^2}$$

for a re-emission distribution between isotropic and cosine, where  $\alpha$  is the reflection coefficient.

<sup>7</sup>J. G. Hoffman, *Radiation Doses in the Pajarito Accident of May 21, 1946*, LA-687 (May 26, 1948).

<sup>8</sup>Consolidated Vultee Aircraft Corporation.

The value of  $\alpha = 0.085$  thus obtained is in fair agreement with the experimental value of 0.08.

### Estimate of Age in NaZrF<sub>5</sub>

J. E. Faulkner

For the fused-salt reflector-moderated reactor reactivity calculations it is necessary to know the age of fission neutrons in the fused salt which is used as a vehicle for the fuel. In preparation for an experiment to measure the age, a rough calculation of this quantity has been carried out below.

The age  $\tau$  to energy  $E_0$  for a point source of fission neutrons in a medium for which it is possible to apply Fermi age theory is given by

$$\tau = \int_{V(E_0)}^{\infty} \left\{ \frac{2 dV G(V)}{VG[V(E_0)]} \right\} \left[ \frac{\lambda^2}{3\xi(1 - \overline{\cos \theta})} \right],$$

where

$E$  = energy (Mev),

$V = \sqrt{2E}$ ,

$\xi$  = average loss in logarithm of energy upon collision,

$\overline{\cos \theta}$  = average cosine of angle between neutron direction before and after collision in laboratory system,

$\lambda$  = mean free path,

$$G(V) = e^{-(V-1)^2/2} [1 - F(1 - V)]$$

$$- e^{-(1/2)(V+1)^2} [1 - F(V + 1)] + \sqrt{2\pi}$$

for  $0 \leq V \leq 1$ ,

$$= e^{-(V-1)^2/2} [1 + F(V - 1)]$$

$$- e^{-(1/2)(V+1)^2} [1 - F(V + 1)]$$

for  $V \geq 1$ ,

$$F(x) = e^{x^2/2} \int_x^{\infty} e^{-t^2/2} dt.$$

$G(V)$  is the integral of the analytic fit to the fission spectrum. By this method the age to indium resonance for fission neutrons was calculated for NaZrF<sub>5</sub> of density 4 g/cc. The method was tested by calculating the age for graphite with a density

## SHIELDING RESEARCH

of 1.6 g/cc. The age for graphite was found to be 311 sq cm, which differs by 6% from the experimental value of 330 sq cm. In the case of  $\text{NaZrF}_5$  it was estimated to be 433 sq cm  $\pm$  25%.

### LID TANK FACILITY

C. L. Storrs<sup>9</sup>  
G. T. Chapman      J. M. Miller  
D. K. Trubey

### Slant Penetration of Neutrons in Water

G. T. Chapman

A series of experiments designed to study the penetration of fast neutrons through slant thicknesses of a material have been in progress at the Lid Tank. Basically, the experiments consist of an air-filled aluminum duct placed against the source plate and tilted from the normal to the source plate to reduce the background of fast neutrons from the ORNL Graphite Reactor. In this manner the duct is used as a collimated "point" source, and fast-neutron measurements are made in the water beyond with various distances between the detector and the end of the duct (Fig. 1.1).

For the first experiments the duct was 100 cm long and 5 cm in diameter. By integrating the dose along planes through the radiation field, the dose penetrating different thicknesses of water at various angles was determined. Although this information must be regarded as tentative until the tests have been repeated with more care, the results indicate that most of the dose comes from neutrons penetrating the shield along the slant path. These results are similar to those obtained for gammas for comparable attenuations.<sup>10</sup>

The observed relaxation lengths, calculated by using the normal shield thickness, ranged from 4.9 to 6.3 cm. These values are smaller than those obtained with thick shields. Such small values would be expected since a fission spectrum was used as a source, and since the shields were quite thin and there was no attenuation hardening of the spectrum.

In a second phase of the experiments the 100-cm duct was replaced by a duct 90 cm long and 7.3 cm

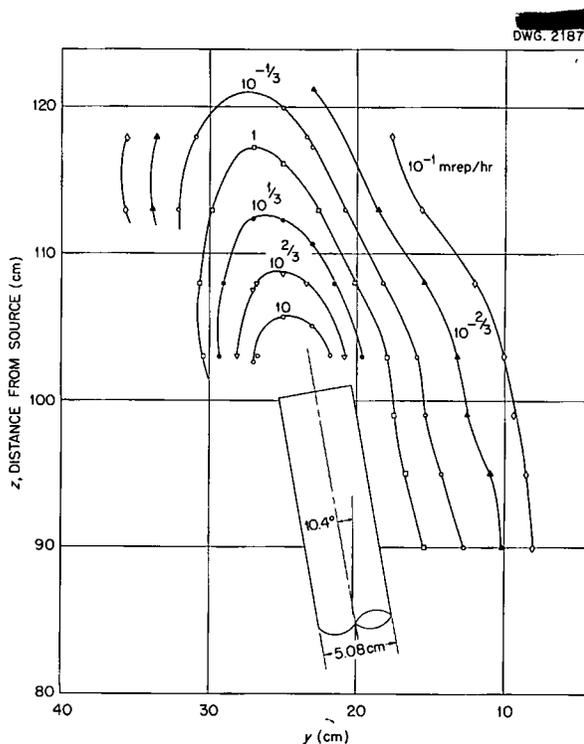


Fig. 1.1. Fast-Neutron Dose Distribution off the End of a Duct 101.6 cm Long and 5.08 cm in Diameter.

in diameter. This was done in order to increase the flux in the water beyond the duct for measurements with greater slant thicknesses (up to 40 cm) of water between the duct and detector. These data have not yet been analyzed.

### Neutron Streaming Through Air Ducts

J. M. Miller

The air-duct experimentation has been resumed in the Lid Tank as a study of both the interference between adjacent ducts and the neutron streaming through a single duct. Each duct consists of one to three straight cylindrical sections (22 in. long,  $3\frac{15}{16}$ -in.-dia thin aluminum walls) joined at angles of 45 deg. These experiments have been described in an ANP progress report<sup>11</sup> and will be reported in detail.<sup>12</sup>

<sup>9</sup>GE-ANP.

<sup>10</sup>F. S. Kirn, R. J. Kennedy, and H. O. Wyckoff, *Oblique Attenuation of Gamma-Rays from Cobalt-60 and Cesium-137 in Polyethylene, Concrete and Lead*, NBS-2125 (Dec. 23, 1952).

<sup>11</sup>J. M. Miller, *ANP Quar. Prog. Rep. March 1, 1954*, ORNL-1692, p 117.

<sup>12</sup>C. L. Storrs and J. M. Miller, *Some Neutron Measurements Around Air Ducts*, ORNL CF-54-2.93 (to be issued).

### Effect of Thin $B_2O_3$ Coating on an Iron Gamma Shield

D. K. Trubey

A new method for inserting a boron layer in a shield was investigated; in this method a ceramic coating of  $B_2O_3$  (0.011 g of boron per square centimeter) was applied to the source side of a  $1\frac{5}{8}$ -in. iron gamma shield. A 15-in. tank of borated water containing 1.1% boron was placed behind this configuration in order to reduce the capture gammas which would result from the high thermal flux behind the iron.

The coating was sufficiently durable for the experiment and reduced the gamma dose behind the entire configuration by about 20% (see Fig. 1.2).

#### Removal Cross Sections

The effective fast-neutron removal cross section of  $C_7F_{16}$  has been measured with the use of 15 in. of the liquid instead of the 28 in. used in a pre-

vious measurement. Despite the large difference in thickness, the measured values of the molecular removal cross section agreed to within 5%, which is within the limits of the experimental error, and thus provide a check on the validity of the geometrical corrections that were made. The revised value for carbon, 0.80 barn/atom instead of 0.84 barn/atom, when used with the latest  $C_7F_{16}$  measurements, gives a removal cross section of 1.31 barns/atom for fluorine as compared with the value of 1.36 barns/atom published in the previous report.

A check on removal-cross-section measurements has also been provided by an experiment in which solid  $B_2O_3$  was used in the usual large-slab geometry. This gave a value of  $4.4 \pm 0.14$  barns for the removal cross section of  $B_2O_3$ , as compared with 4.56 barns calculated on the basis of 0.87 barn for boron and 0.94 barn for oxygen.

The value for tungsten, erroneously stated in the last progress report as  $2.06 \pm 0.2$ , should have been  $2.6 \pm 0.2$ .

#### BULK SHIELDING FACILITY

R. G. Cochran

G. Estabrook	K. M. Henry
J. D. Flynn	H. E. Hungerford
M. P. Haydon <sup>13</sup>	E. B. Johnson

The major effort of the Bulk Shielding Facility has been devoted to measuring mockups of two bulk shields for General Electric ANP. The first shield was a mockup of a duct system and shield to be used at the G-E Idaho reactor test facility. The second mockup consisted of two sections of the reactor shield of the R-1 ANP divided shield which is to be tested this summer at the Tower Shielding Facility. Since the data from these tests have been reported in a recent ANP quarterly progress report,<sup>14</sup> they will not be given here.

#### Intercomparison of Radiation Dosimeters

H. E. Hungerford

A group under the auspices of the Health Physics Division recently made comparative radiation dose measurements of various monoenergetic and other neutron and gamma-ray sources with several dif-

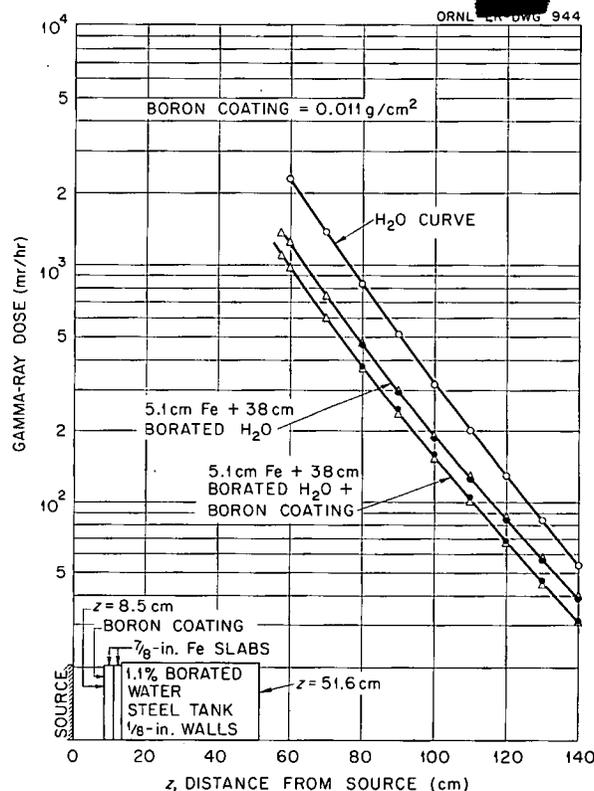


Fig. 1.2. Gamma-Ray Dose Beyond Iron Shield With and Without  $B_2O_3$  Coating.

<sup>13</sup>Part time.

<sup>14</sup>R. G. Cochran et al., ANP Quar. Prog. Rep. March 1, 1954, ORNL-1692, p 123.

## SHIELDING RESEARCH

ferent types of instruments in order to intercalibrate the various methods of measuring nuclear doses. In addition, it is hoped that the data will furnish reliable information on the gamma-ray yield of various neutron sources, as well as on the neutron response of gamma-ray film and CO<sub>2</sub> ionization chambers.

The study was made by using Po-B and Po-Be neutron sources, a Co<sup>60</sup> gamma-ray source, and beams of monoenergetic neutrons from the Van de Graaff and Cockcroft-Walton machines with energies which ranged from 0.1 to 14 Mev. Certain parts of the experiment are still in progress and there is nothing definite to report at this time. Preliminary inspection of the raw data indicates that the various calibration techniques yield results that are in good agreement.

The instruments used in the experiment were fast-neutron standard dosimeter, fast-neutron phantom type of dosimeter, CO<sub>2</sub> gamma-ray ionization chamber, tissue-equivalent total dose ionization chamber, and radiation-sensitive films.

### Spectrum of Gamma Rays Emitted by the BSR

F. C. Maienschein      T. A. Love

A measurement of the spectrum of gamma rays from the unshielded BSR was made as a continuation of the studies of the gamma-ray energy spectra and angular distributions through lead and water.<sup>15</sup> The shape of the spectrum shows the usual broad peaks at 2.2 Mev that are due to water capture and those at 7 to 8 Mev that are due to capture in aluminum and possibly other materials. The cause of a 0.4-Mev peak, which appears to be real, has not been ascertained. A 0.49-Mev peak, however, may be ascribed to gamma rays from the 0.478-Mev level in the Li<sup>7</sup> formed by neutron capture in the boron liner of the spectrometer and to annihilation radiation.

The experiment is described in an ANP progress report<sup>16</sup> and in a scientific journal.<sup>17</sup>

<sup>15</sup>F. C. Maienschein, *Gamma-Ray Spectral Measurements with the Divided Shield Mockup, Part I*, ORNL CF-52-3-1 (March 3, 1952); *Part II*, ORNL CF-52-7-71 (July 8, 1952); *Part III*, ORNL CF-52-8-38 (Aug. 8, 1952); and T. A. Love, *Part IV*, ORNL CF-52-11-124 (Nov. 17, 1952).

<sup>16</sup>R. G. Cochran et al., *ANP Quar. Prog. Rep. Dec. 10, 1953*, ORNL-1649, p 116.

<sup>17</sup>F. Maienschein and T. Love, *Nucleonics* 12, No. 5 (in press).

### Fast-Neutron Spectra from the BSR

R. G. Cochran      K. M. Henry

Spectral measurements have been made with the proton-recoil spectrometer<sup>16</sup> of the fast neutrons emerging from the BSR for attenuations in the surrounding water of from 0 to 70 cm. In the measurements at 70 cm, 4½ in. of lead was interposed between the reactor and the spectrometer. In addition to the spectral measurements, a brief study was made of the angular distribution of emergent neutrons at 10 cm from the reactor surface. The detailed spectra are given in the ANP quarterly report.<sup>16</sup>

### Intercalibration of Recoil-Proton Fast-Neutron Dosimeters

The recoil-proton fast-neutron dosimeters used at the BSF have been of two types: one with two sections filled with 2 atm of methane and the other with three sections filled with 1 atm of ethylene. A series of measurements carried out to obtain intercalibration factors showed that there was a constant relationship between them. Similar measurements by a third type of dosimeter, having three sections filled with 2 atm of methane, also were in agreement.

The measurements with the various counters are plotted in Fig. 1.3.

### TOWER SHIELDING FACILITY

C. E. Clifford

T. V. Blosser      J. L. Hull  
L. B. Holland      F. N. Watson

The construction of the Tower Shielding Facility (see Fig. 1.4) was completed during February, and acceptance tests were successfully run on the tower structure and hoists. The reactor, the mechanical components of which were constructed by the ORNL Research Shops, was loaded to criticality the first week in March, and the first experiment was started.

A tentative experimental program, the instrumentation, and estimates of some of the radiation levels to be expected are given below.

### Experimental Program

C. E. Clifford

**Critical Experiments.** The initial critical experiment with the TSR was performed primarily for the

ORNL-RR-DWG 945

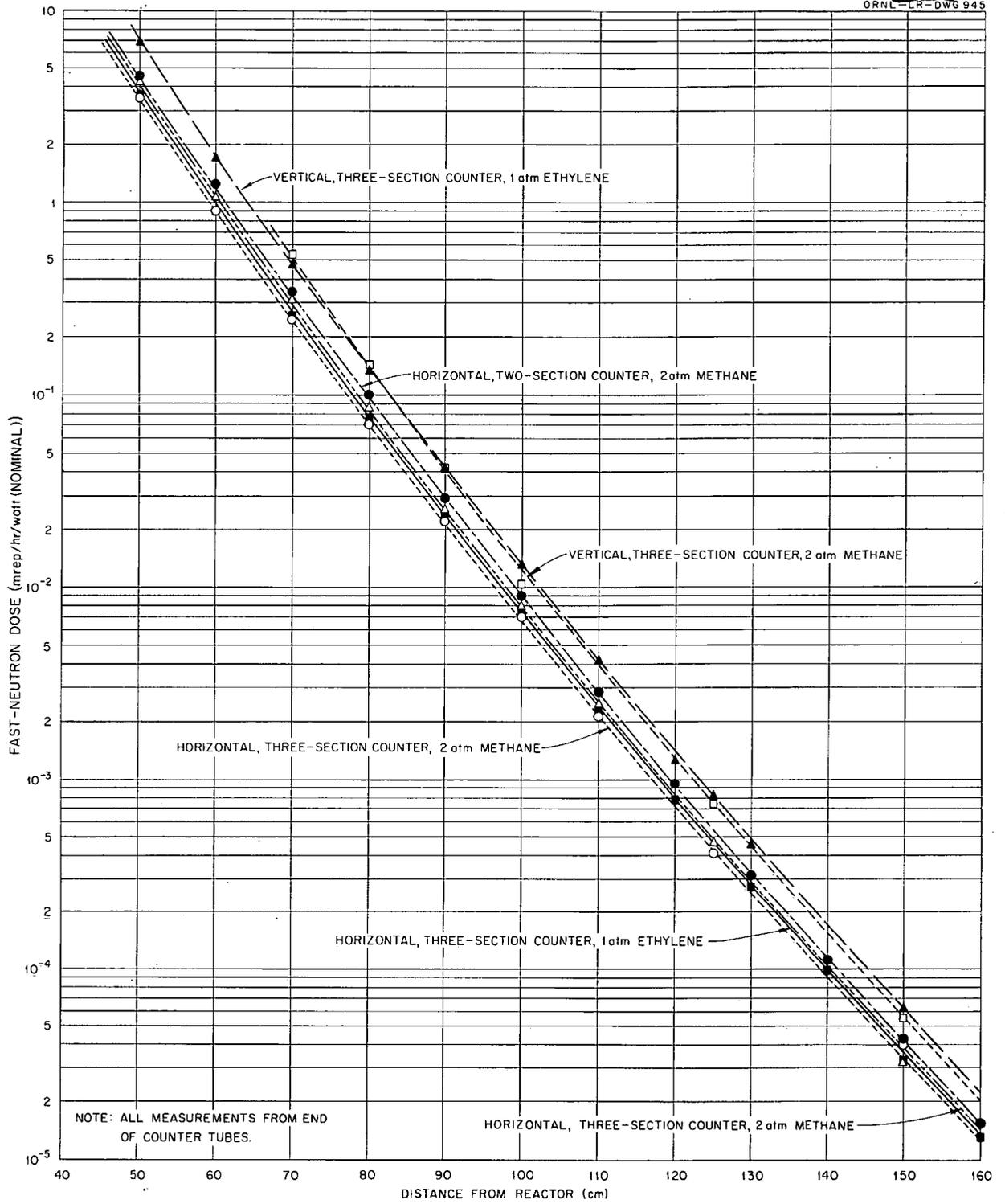


Fig. 1.3. Fast-Neutron Dosimeter Tests at the BSF.

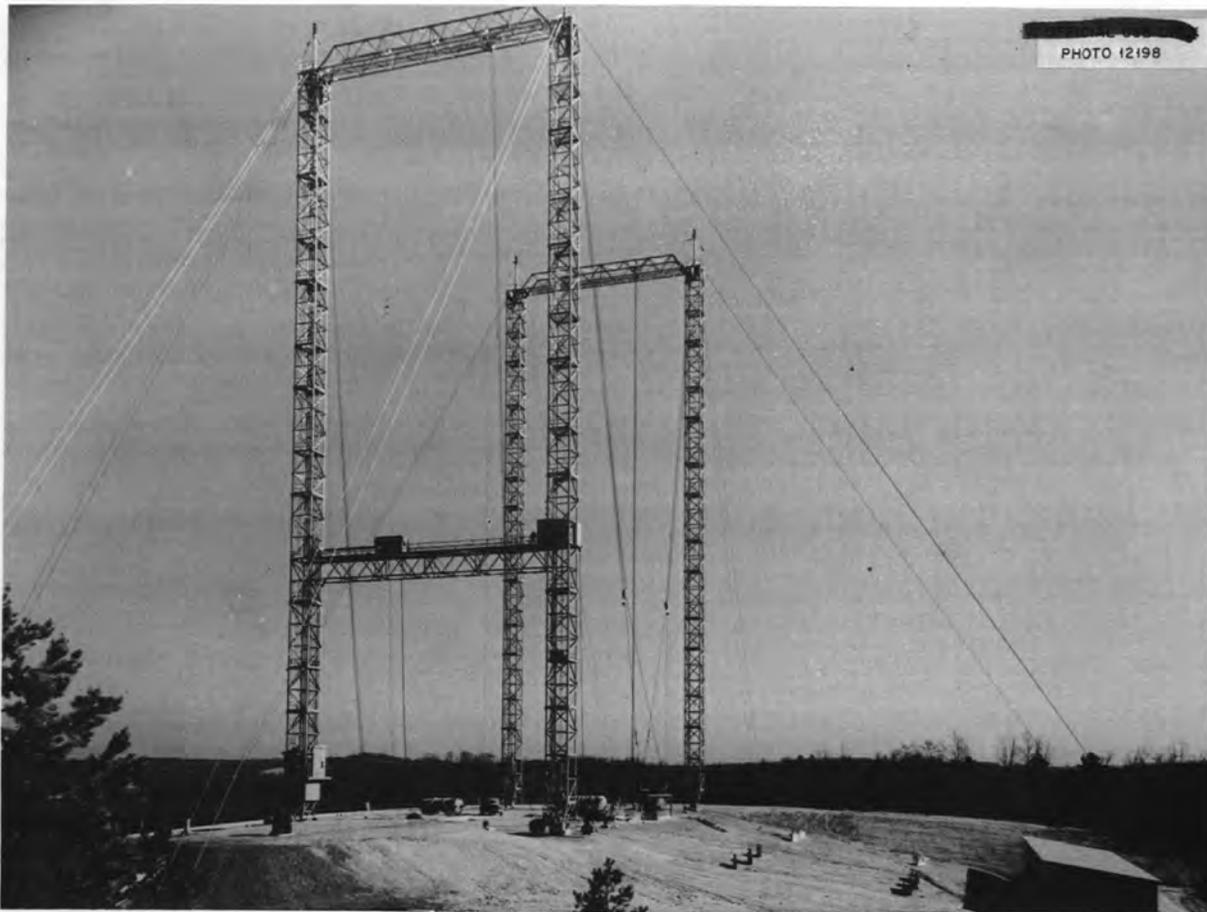


Fig. 1.4. Tower Shielding Facility.

purpose of checking the reactor instrumentation. The elements are MTR-type with a 140-g uranium loading per element. The loading used for the experiment had been previously checked by a critical experiment at the BSF. It consisted of 3 half elements (control) and 23 full elements, making a 5 by 5 array with an additional element in the center of the back face. The indications were that this array was critical with a three-quarter element in the center of the front row. During the course of the experiment the reactor power was raised sufficiently high ( $\sim 50$  watts) to check the safety chambers.

Other immediate critical experiments will be limited to the 5 by 6 loading necessary to assure that the correct excess  $k$  is present to permit operation of the reactor at 100 kw with a water temperature of 140°F. Additional critical experi-

ments will, of course, be required whenever the reactor is installed in a new mockup or is being subjected to other unusual circumstances.

**Preliminary Power-Level Determinations.** A preliminary requirement for experimental work at the facility will be an accurate determination of the reactor power, since all dosage measurements must be reported in terms of that power. This determination will be accomplished primarily by a comparison of thermal-neutron flux levels within the reactor with those in the BSR. The reactor will be loaded with a configuration that has been previously measured at the BSR. The relative thermal-neutron flux within the TSR will be mapped by scanning cobalt wires which have been inserted the full length of the fuel elements, and the absolute flux at several points within the reactor will be obtained with the use of gold or uranium foils.

A further check on the power will be obtained from neutron- and gamma-dose measurements in a water shield surrounding the reactor.

In addition, the effect of the lead above the fuel elements on the gamma dose above the reactor before and after shutdown will be determined and compared with BSF measurements where no lead was present.

**Background Determinations.** The contribution of ground-scattered radiation to the total dose will be determined by a series of dosage measurements for various configurations of reactor and crew compartment. The effect of altitude of the reactor tank on the dose at the crew position will be measured for several reactor positions within the tank. Location of the detector at various positions within the crew compartment water tank will be necessary to indicate the crew compartment penetration of this ground-scattered background.

The contribution of radiation scattered from the tower structure will be determined by an experiment in which the distance from a tower leg of both the crew compartment and the reactor is varied simultaneously. This experiment must be performed at an altitude of 160 ft because of limitations inherent in the hoisting system.

**GE-R1 Divided-Shield Mockup Experiment.** A mockup of the G-E divided shield for the R1 reactor (subsequently replaced) will be the first aircraft shield design tested at the TSF. The reactor shield for this experiment was designed and built by GE. All shield thicknesses were maintained to the designed thickness, with the over-all mockup being scaled down to wrap closely around the TSR. Air ducts, which can be flooded to permit measurement of an unperforated shield, were included. As a result of scaling down, some minor uncertainties involving geometrical calculations will be introduced in the unperforated shield measurements. Somewhat more serious perturbations will be introduced into the transmission of radiation through the air ducts because the transmission is very strongly dependent on the geometry of the ducts. However, these corrections should be calculable on the basis of ducting experiments at the LTSF and BSF.

The crew shield for this experiment is being supplied by ORNL and is also scaled down to the extent of reducing the cavity inside to a cylindrical volume 3 ft in diameter and 6 ft long. Shielding thicknesses are maintained according to G-E de-

sign by means of two concentric aluminum tanks which will be filled with water. The lead gamma shielding will be installed to permit variation in the thicknesses at all positions within the crew shield.

The various detectors will be mounted within the crew compartment with sufficient mobility to permit longitudinal scanning at several radial distances within the compartment. This will permit a determination of the dose levels for all regions within the compartment. The experiment will also include mapping of the dose distribution in air for the various regions considered most important for the air-scattering calculations.

**Differential Shielding Experiments.** An extended program of differential shield experiments is planned in order to obtain the variation of dose within the crew compartment of a divided shield for a given variation of any portion of the shield. From these data an attempt will be made to obtain an empirical relationship sufficiently accurate and general to enable the machine calculation of similar divided shields by a perturbation method. Some parts of this experiment will be carried out in the process of shakedown tests of the reactor and instruments prior to the G-E mockup experiment.

**Biological Program.** Since the TSF offers the unique opportunity of exposing large numbers of animals simultaneously to the high-level dosages of both neutron and gamma rays, several groups are planning to initiate biological exposures at the facility. This information on radiation effects will be of interest for military reactor applications as well as for the determination of civil effects from nuclear explosions.

So far, in addition to the Biology Division, only the U.S. Air Force Special Weapons Group and the AEC Division of Biology and Medicine have been contacted in regard to the biological program. However, it seems, from the interest generated from these contacts, that a large program could be readily envisioned. As presently conceived, the biological exposures will take place intermittently between the aircraft shielding experiments.

**Optimized Shield Mockup.** On the basis of the differential shield measurements and subsequent calculations of the reactor shield in a given reactor and crew compartment, a new shield mockup is to be built. This mockup, which is to be optimized

## SHIELDING RESEARCH

as nearly as possible with respect to weight for a given separation distance, dose level, and reactor configuration, will be tested at the facility, probably early in 1955.

It will be possible to introduce perturbations in the mockup to effect the next step in the optimization. The mockup will also be necessary to obtain the normalization factors required for integration of differential shield measurements in order to predict the total dose in the crew compartment.

### Radiation Detection Equipment for the TSF

T. V. Blosser

During the past year a complete set of instruments has been collected and checked out for use at the TSF for experimental dose and energy measurements. Some of the instruments are standard and have been in general use at both the LTSF and BSF. Other instruments have been developed or modified to meet the requirements imposed by the facility.

A specially designed lead absorber wheel varying in thickness from 0 to 0.7 in. will be used in conjunction with the  $\frac{1}{2}$ -in. anthracene crystal to show the effect of adding small thicknesses of lead to a shield. The anthracene crystal will be properly shielded so that only the dose entering through a small circular opening in one side will be detected and the absorber wheel can be rotated to cover the opening.

A wire (cobalt or gold) scanner also has been specially developed to determine the flux distribution within each fuel element of the reactor. A single wire can be inserted along the full length of the center fuel plate and, upon withdrawal, can be surveyed with a NaI crystal and its activity recorded automatically.

An isodose plotter will be used with most of the TSF instruments to automatically plot the intensity as a function of position in the water tank which represents the aircraft crew compartment. It can, of course, also be used to record the loci of other pertinent data.

### Estimate of Neutron and Gamma Radiation Expected at the TSF

C. E. Clifford

In anticipation of a biological program in which mass exposures of large animals will be undertaken, estimates of some of the neutron and gamma doses which can be achieved at the TSF have been made. The dose rates, which were calculated on the basis of measurements of radiation at the BSF and LTSF, are given in Table 1. Details of the calculation have been presented in a separate report.<sup>18</sup>

<sup>18</sup>C. E. Clifford, *Estimate of Neutron and Gamma Radiation Expected at Tower Shielding Facility*, ORNL CF-53-12-23 Revised (Dec. 10, 1953).

TABLE 1. ESTIMATE OF RADIATION DOSES AT THE TSF FOR A REACTOR POWER OF 100 kw

POINT OF MEASUREMENT	NEUTRON DOSE (rep/hr)	GAMMA DOSE (r/hr)
18-cm Pure-Water Shield		
At surface	100,000	1,600,000
123 ft from surface	2.08	41.0
250 ft from surface	0.33	10.0
600 ft from surface	0.018	1.0
18-cm Lead-Borated-Water Shield (11.4 cm lead)		
At surface	480,000	80,000
123 ft from surface	10	2.0
228 ft from surface	2.0	0.57
294 ft from surface	1.0	0.27

## 2. CRITICAL EXPERIMENTS

A. D. Callihan

## BASIC REACTOR STUDIES

Measurements of  $\eta$  for  $U^{233}$ 

J. T. Thomas      J. K. Fox

Aqueous solutions of  $U^{233}$  and  $U^{235}$  oxyfluoride salts ( $UO_2F_2$ ) were compared in two spherical critical assemblies over the temperature range 25 to 100°C. The critical spheres were 10.4 and 12.6 in. in diameter and had effectively infinite water reflectors.

With the hypotheses that the reported nuclear constants for  $U^{235}$  were reliable and that the leakage spectra for  $U^{233}$  and  $U^{235}$  were about equal when their solutions were critical in the same water-reflected sphere, the value of  $\eta(233)$  over the energy range 0.026 to 0.032 eV was determined to be  $2.33 \pm 0.03$ .

It is desirable ultimately to determine  $\eta(233)$  as a function of neutron energy. However, the probable error for an individual measurement is very large because of the uncertainty in the absorption cross section of  $U^{233}$  on which the determination of  $\eta(233)$  from these experiments depends. Plans are currently being made to measure the ratio  $\sigma_a(233)/\sigma_a(235)$  as a function of neutron energy with the velocity selector. When this information is available we hope to be able to analyze these critical experiment data to obtain  $\eta(233)$  as a function of neutron energy.

The variations of critical mass for  $U^{233}$  and  $U^{235}$  as a function of temperature in both the 10.4- and the 12.6-in.-dia reactors are shown in Figs. 2.1 and 2.2. The reactivity temperature coefficients of  $U^{233}$  and  $U^{235}$  in the 12.6-in.-dia reactor over the temperature range are  $-3.54 \times 10^{-5}/^\circ\text{C}$  and  $-1.16 \times 10^{-4}/^\circ\text{C}$ , respectively. Period measurements made in the 12.6-in.-dia reactor indicate that when equal amounts of excess reactivity are added to just critical assemblies of  $U^{233}$  and  $U^{235}$ , the period observed in  $U^{235}$  is approximately three times that observed in  $U^{233}$ .

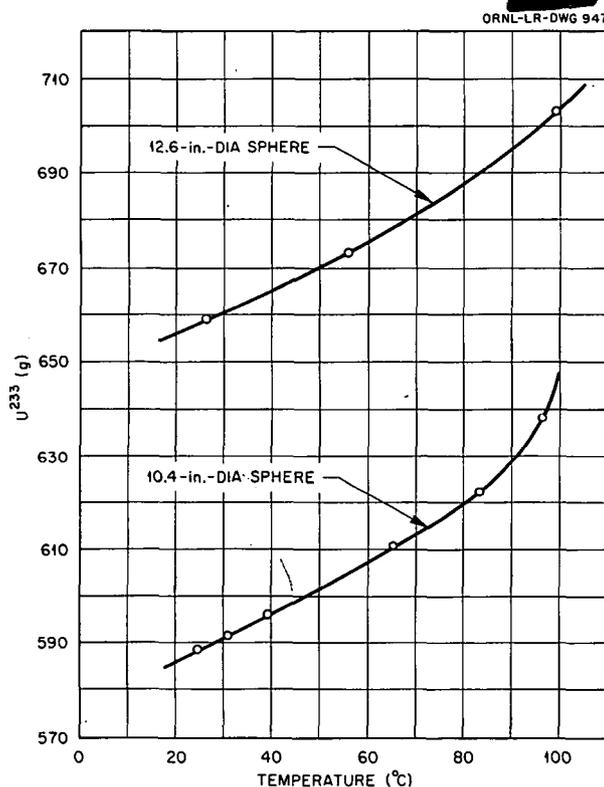


Fig. 2.1. The Critical Mass of  $U^{233}$  in Water-Reflected Spherical Reactors.

Critical Mass Studies of Aqueous Solutions of  $U^{233}$ J. K. Fox      J. T. Thomas  
E. R. Rohrer

A series of experiments has been performed to determine the critical parameters of water solutions of uranium (98.7%  $U^{233}$ ) as uranyl oxyfluoride ( $UO_2F_2$ ) and uranyl nitrate [ $UO_2(NO_3)_2$ ] salts. Spheres, square cylinders, and effectively infinite cylinders were investigated. Most reactors had effectively infinite paraffin reflectors, and a comparison made between infinite paraffin and infinite water reflectors showed that the former was slightly more efficient.

The  $U^{233}$  concentration range for these experiments extended from 30 to 600 g/liter. The minimum experimentally determined critical mass was 589.5 g of  $U^{233}$  at an H: $U^{233}$  atomic ratio of 419 in a sphere 10.4 in. in diameter. The minimum experimentally determined critical volume was

## CRITICAL EXPERIMENTS

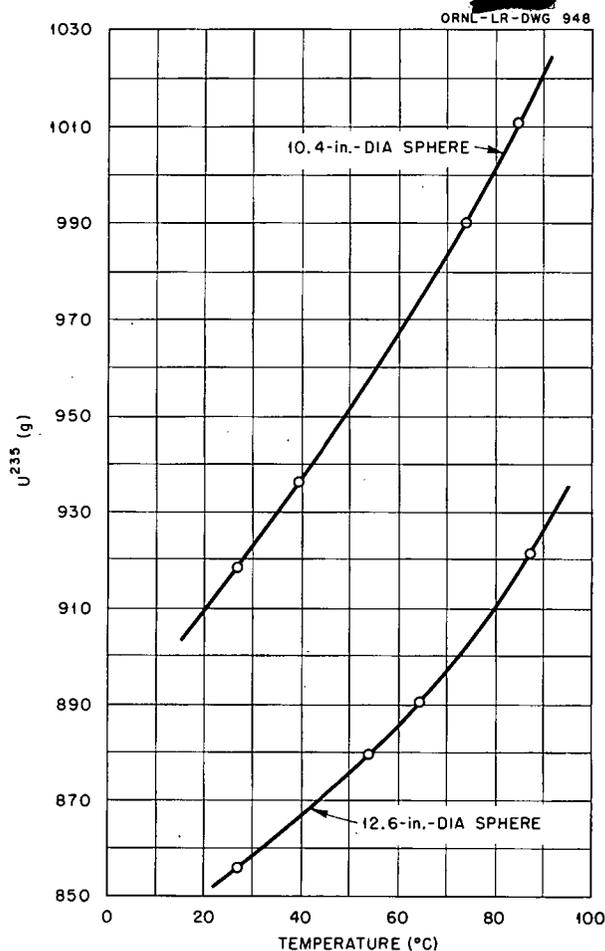


Fig. 2.2. The Critical Mass of  $U^{235}$  in Water-Reflected Spherical Reactors.

3.66 liters in a 6.7-in. square cylinder at an H: $U^{233}$  atomic ratio of 39.4.

Extrapolated multiplication data indicate that a 5-in.-dia pipe can be made critical if reflected but that a 4.5-in.-dia reflected pipe would be subcritical at all moderations. The number of measurements possible without a neutron reflector was limited by the available  $U^{233}$  inventory. It was found, however, that 2.02 kg of  $U^{233}$  contained in a 10-in. square cylinder was critical at an H: $U^{233}$  atomic ratio of 154, whereas a sphere 12.5 in. in diameter was critical with 1.14 kg of  $U^{233}$  at an H: $U^{233}$  ratio of 381.

The effect of the nitrate ion, as compared with that of the oxyfluoride ion, was found to be mainly that of greater hydrogen displacement at high concentrations. Hence the nitrate ion increases

the minimum infinite cylinder critical diameter to more than 5 in. but has very little effect on the minimum critical mass. Application of the data to the calculation of geometries other than those used experimentally was made by the method described by G. I. Bell.<sup>1</sup>

### Boron Poisoning in Critical Slabs

L. W. Gilley                      D. F. Cronin  
V. G. Harness

A series of experiments has been initiated to determine the effect of partitioning with boron slabs on the criticality of a slab of solution. The multiplying media being studied are bare, as well as partially water-reflected, aqueous solutions of  $UO_2F_2$  in which the uranium has been enriched 93.3% with  $U^{235}$ .

Because of the inherent experimental difficulties in providing a neutron reflector above the solution, the reflected slabs being studied are only half-reflected; that is, the level of the reflector water is brought to approximately the same height as the center of mass of the fuel solution. The poison partition sheets used have been, for the most part,  $\frac{1}{4}$ -in. boral sheets clad in stainless steel. However, one set of experiments was run from which a comparison could be made between  $\frac{1}{4}$ -in. boral,  $\frac{1}{2}$ -in. boral, and a stainless steel-covered slab of boron carbide. The  $\frac{1}{4}$ -in. boral sheet contains about 300 mg of boron per square centimeter.

In a typical experiment the fuel solution was added to an aluminum container with a 30 × 60 in. rectangular base. A number of equally spaced poison partition sheets had been placed vertically in this container parallel to the longer base dimension. The critical height of the fuel solution level corresponding to this number of poison sheets, that is, to the compartment width, was then measured. At present all experiments have been performed with a solution concentration corresponding to an H: $U^{235}$  atomic ratio of 78.7. Very preliminary data thus far obtained are summarized in Figs. 2.3 and 2.4 where the critical heights and critical masses have been plotted as a function of the width (inside measurement) of the compartment formed by the boral sheets. The curves for the completely reflected slab were obtained from the data by assum-

<sup>1</sup>G. I. Bell, *A Simple Method of Calculating Critical Masses of Proton Moderated Assemblies*, LA-1548 (May 1953).

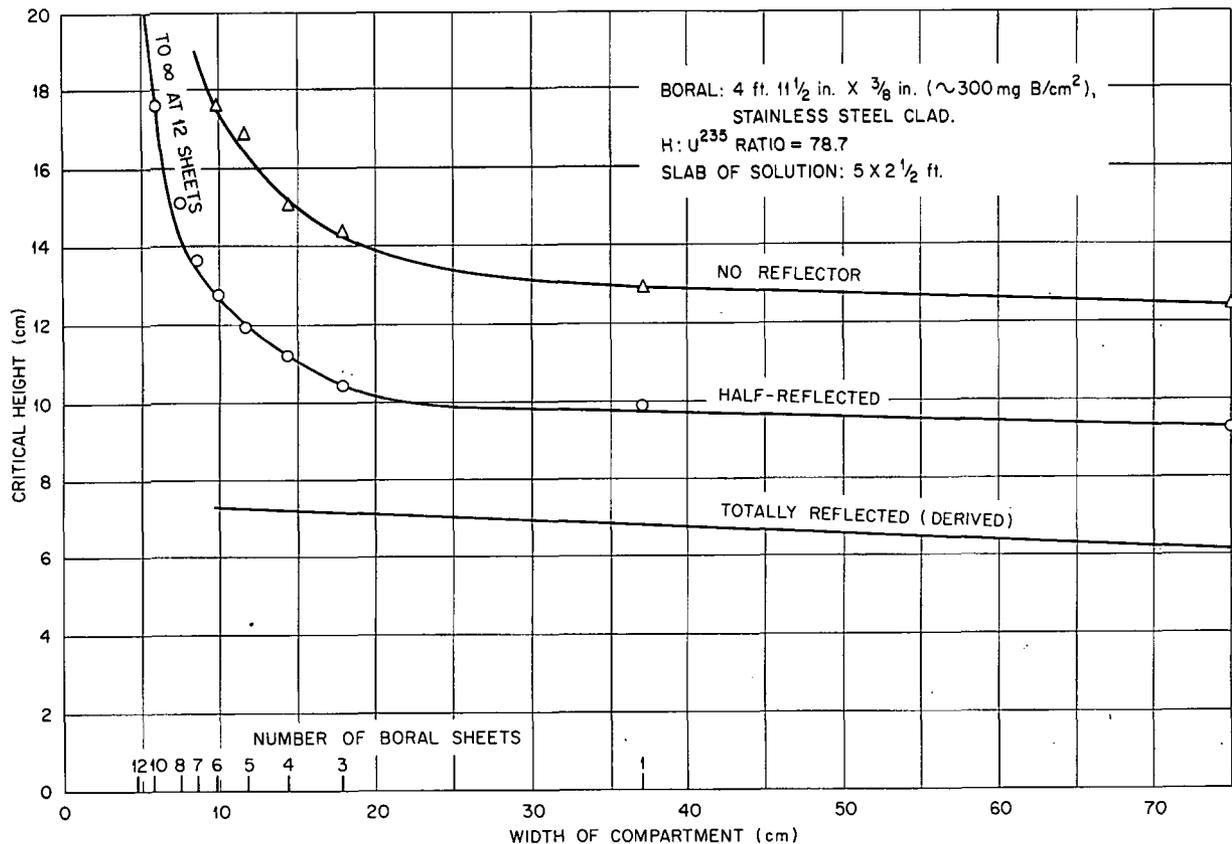


Fig. 2.3. Effect of Boron Poisoning on Critical Height of a Slab of Uranium Solution.

ing the reflector savings, which would be incurred by adding a water reflector above the solution, to be equal to those resulting when the reflector was added to the bottom of the slab.

Doubling the boron thickness in the partitions increased the critical mass slightly. A  $\frac{3}{8}$ -in. layer of stainless steel placed below the slab of solution serves as a partial reflector; when the steel separates the slab from a thick water reflector, it increases the critical mass. The results are in agreement with those previously reported.<sup>2</sup> The experiments will be continued with solutions of lower concentration.

#### Criticality Data for Uranium-Aluminum Slug Lattices in Water

J. K. Fox

J. H. Marable

Regular lattices of uranium-aluminum alloy slugs were flooded with an effectively infinite water reflector in order to determine the number of slugs

required for criticality as a function of the lattice spacing. These slugs are 1.015 in. in diameter and 12.00 in. in length and consist of an alloy of aluminum and 93.3% enriched  $U^{235}$ . The mass of  $U^{235}$  in each slug is 22.3 g, which is 5% of the total slug mass. These experiments are similar to those of a more extensive program reported previously.<sup>3</sup>

The experimental arrays were made up of single tiers of either square or hexagonal lattice cells uniformly spaced. The minimum number of slugs which could be made critical was 134, arranged in an approximately cylindrical array of hexagonal cells, with an edge-to-edge slug spacing of 0.6 in. Prior calculations of a semiempirical nature, based

<sup>2</sup>D. Callihan, D. F. Cronin, J. K. Fox, and J. W. Morfitt, *Critical Mass Studies, Part V*, K-643 (June 30, 1950).

<sup>3</sup>D. Callihan *et al.*, *Critical Mass Studies, Part VI*, Y-801 (Aug. 8, 1951).

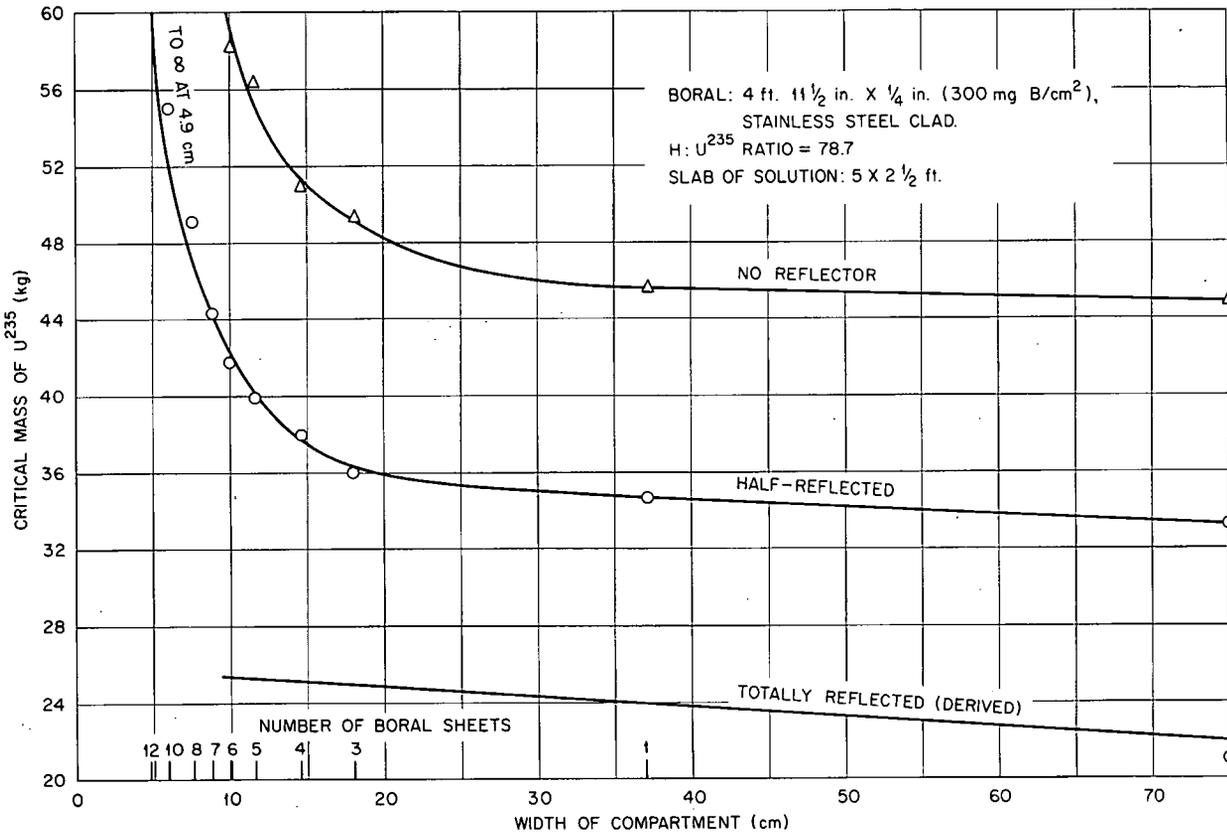


Fig. 2.4. Effect of Boron Poisoning on Critical Mass of a Slab of Uranium Solution.

on data for homogeneous systems,<sup>4</sup> had indicated that the minimum number of critical slugs was about 40% greater than the experimental figure given above.

**Test of Neutron Multiplication by Slightly Enriched Uranium**

D. F. Cronin                      L. W. Gilley  
E. L. Zimmerman                E. R. Rohrer

An experiment has shown that approximately 20,000 lb of liquid  $UF_6$  is subcritical when enriched to 2% in  $U^{235}$ , homogeneously mixed with 7 mole % HF, and contained in four closely packed cylinders 30 in. in diameter and 80 in. long. In this mass of  $UF_6$  there is 123 kg of  $U^{235}$ . In the absence of an aqueous neutron reflector the source neutrons were not significantly multiplied; with a

reflector, the apparent multiplication was as great as two, although no quantitative significance can be attached to that value. The experimental results are in agreement with multigroup Fermi age calculations which give a value of  $k_{\infty}$  of about 0.3. Conservative limits are set for certain process operations which are free from nuclear hazards. The conclusions are not applicable to small element lattices in a moderating medium.

**AIRCRAFT REACTOR STUDIES**

**GE-ANP AC-1 Reactor**

D. V. P. Williams                J. J. Lynn  
V. G. Harness                    J. W. Noaks<sup>5</sup>

Experiments have been completed on a preliminary assembly of the GE-ANP AC-1 Reactor, which is an air-cooled, water-moderated reactor with fuel

<sup>4</sup>C. K. Beck, A. D. Callihan, J. W. Morfitt, and R. L. Murray, *Critical Mass Studies, Part III*, K-343 (April 19, 1949).

<sup>5</sup>Pratt and Whitney Aircraft Division.

elements fabricated from uranium wire. One particular type of fuel element for this reactor is a woven-wire grid shaped somewhat like the lateral surface of two truncated right cones placed base to base, the element having an over-all length of 30 in. and a maximum diameter of 3 in. This construction gives a varying axial uranium density which is greatest at the center of the element.

The structure of this first preliminary test assembly differs significantly from that of the reactor itself. Uranium metal disks (enriched to 93.3% in  $U^{235}$ ) 0.01 in. thick and 2.86 or 1.43 in. in diameter were spaced in each unit cell to simulate the design fuel density distribution and the coolant air chambers. Each fuel disk was sandwiched between two disks of mild steel, 0.018 and 0.062 in. thick, respectively, which were used to represent structural material. These disk sandwiches were separated along the fuel element by aluminum tubing (see Fig. 2.5). A second type of fuel element which consisted of uranium disks between sheets of stainless steel 0.017 in. thick was necessary because of the geometrical limitations

imposed by the aluminum grid supporting the critical assembly (see Fig. 2.6). Each fuel element was surrounded by a 1-in. layer of Plexiglas to simulate the water moderator. The core, which was an approximate right cylinder 30 in. long and 30 in. in diameter, was surrounded radially by  $7\frac{1}{2}$  in. of beryllium reflector. Figure 2.7 is a photograph of a vertical section at the mid-plane of the assembly that shows the fuel cells, each surrounded by the hydrogenous moderator, and the beryllium reflector around the core. There was no end reflector.

The initial assembly as originally designed was an octagonal array of 37 fuel channels. This system, however, could not be made critical. Criticality was achieved with 43 fuel channels, 5 of which were loaded with twice the uranium prescribed. The critical mass was 29.8 kg of  $U^{235}$  with 98.9 kg of steel.

Neutron-flux distributions in the assembly were determined from the activities of bare and cadmium-covered indium foils. Power distributions in the direction of the axis were obtained from the fission-

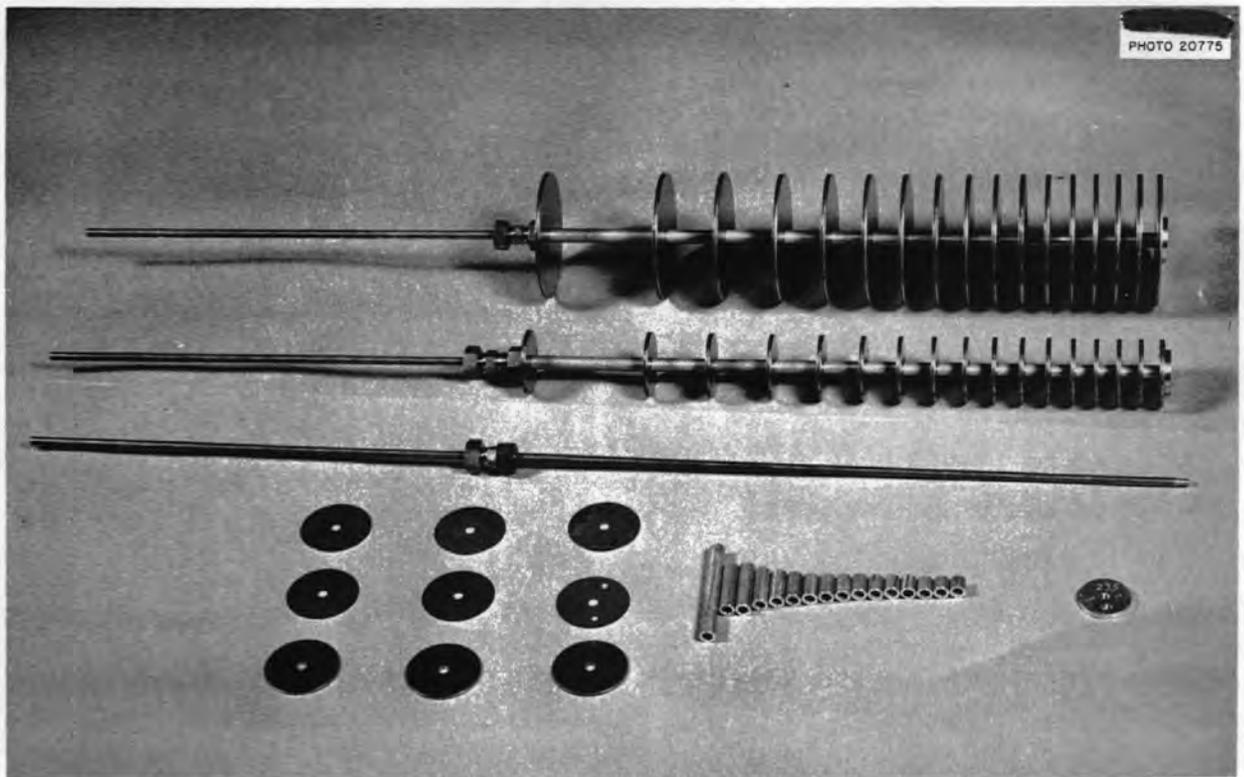


Fig. 2.5. Rod-Mounted Fuel Elements for GE-ANP AC-1 Reactor Critical Experiment.

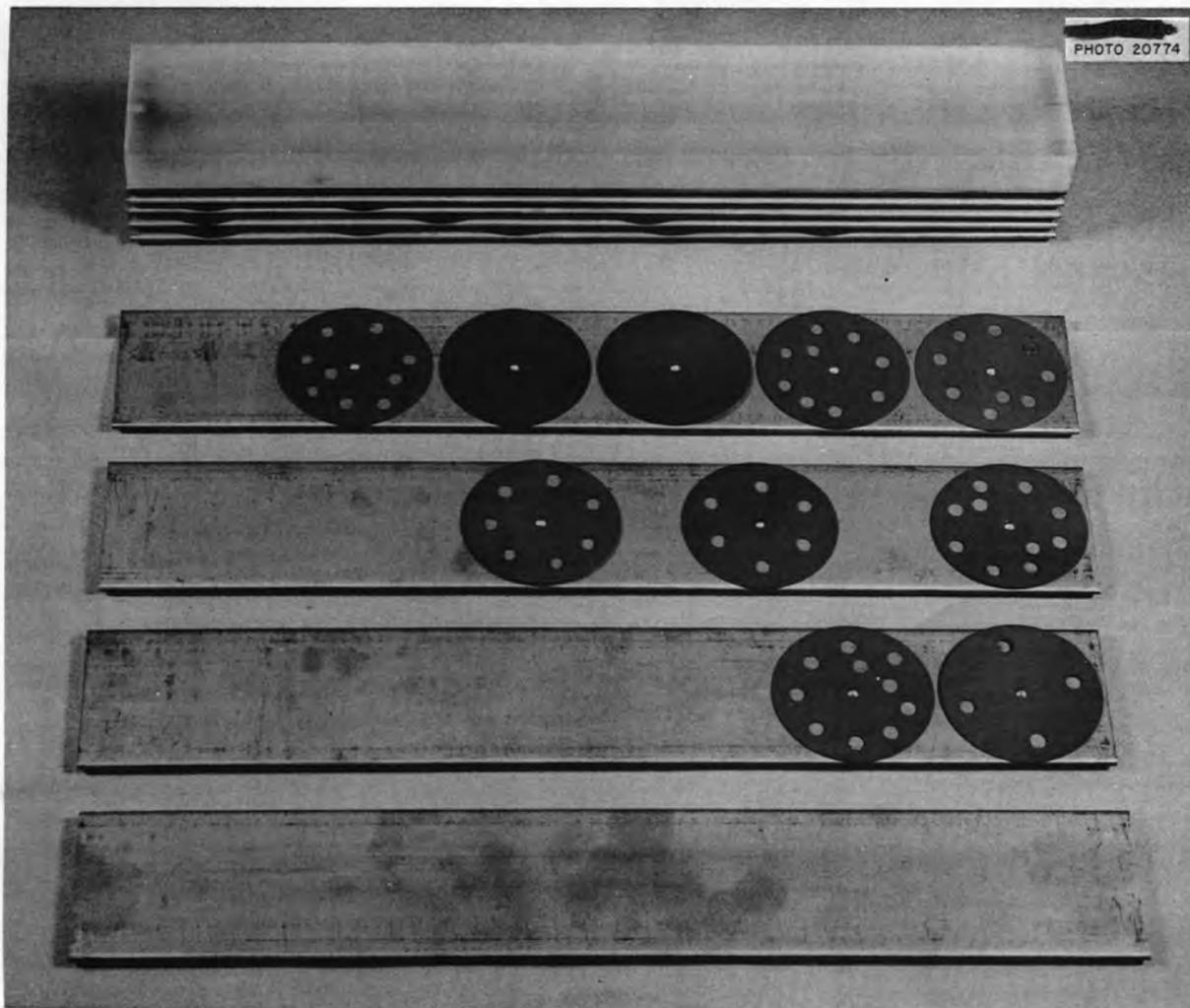


Fig. 2.6. Sheet-Mounted Fuel Elements for GE-ANP AC-1 Reactor Critical Experiment.

product activities collected on aluminum foils. The fraction of fissions produced by thermal neutrons varied from 83% at the center of the assembly to 93% at the mid-point of the unreflected end.

**GE-ANP AC-100-A Reactor**

D. V. P. Williams	J. J. Lynn
D. F. Cronin	E. R. Rohrer
J. D. Simpson <sup>6</sup>	W. Baker <sup>6</sup>
R. C. Evans <sup>6</sup>	H. E. Brown <sup>6</sup>

A preliminary assembly of the AC-100-A air-cooled, water-moderated reactor of the GE-ANP Project has been made. The fuel was in the form

of enriched uranium metal disks placed between steel disks and mounted inside aluminum tubes 4 in. in diameter. The height of this fuel section is 30 in. Thirty-seven of these aluminum tubes, in a pattern designed to give uniform radial power, constitute the core and can be immersed in water that serves as the neutron moderator and effectively infinite reflector. In order to make the system initially critical, it has been necessary to deviate significantly from the prescribed loading by increasing the uranium from 26.6 to 45.6 kg and by decreasing the steel content by about one-half. A series of measurements is in progress to ascertain the cause of the discrepancy between the reactivity of the experiment as designed and as changed so that it could be made critical.

<sup>6</sup>GE-ANP.

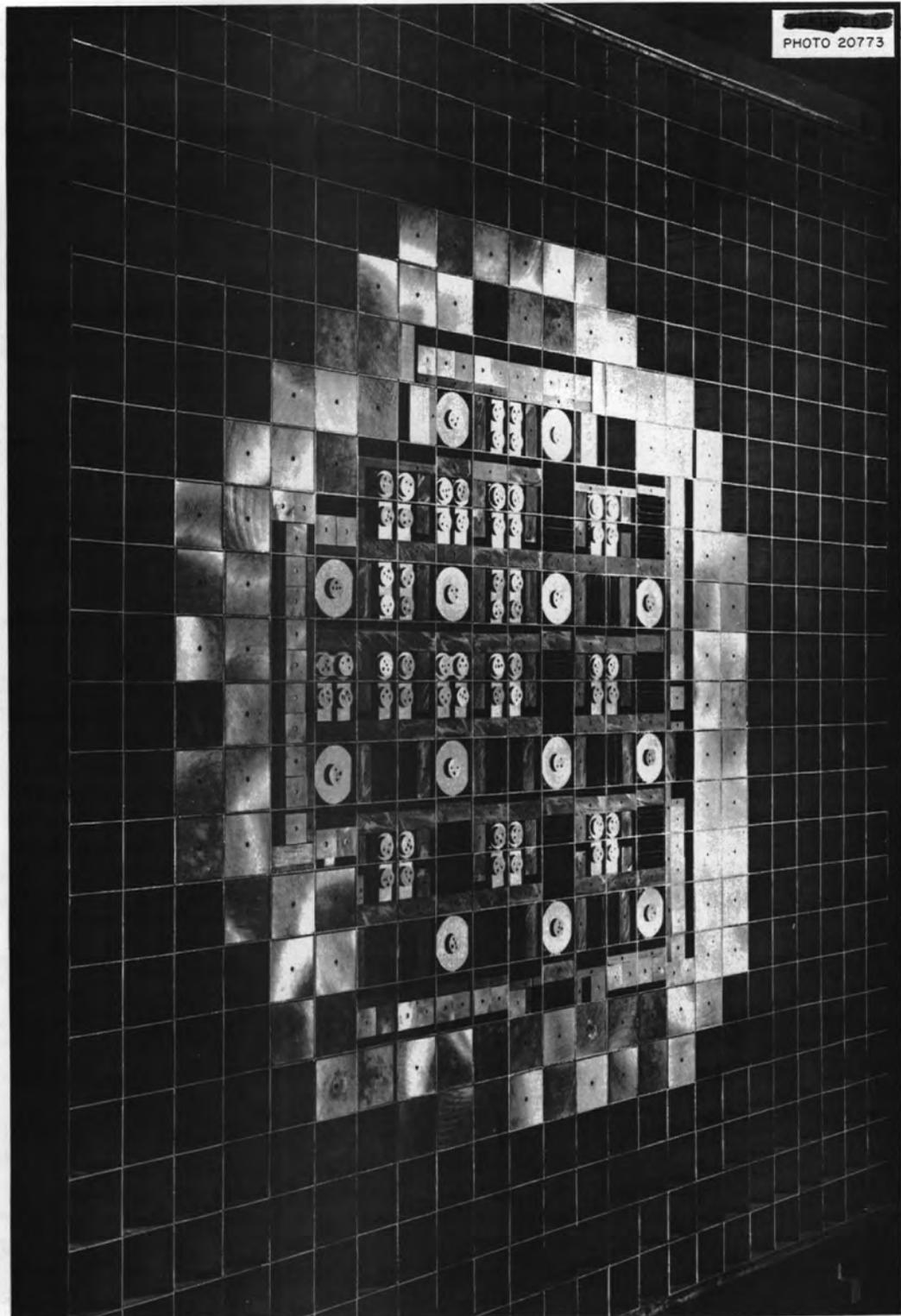


Fig. 2.7. Vertical Section of GE-ANP AC-1 Reactor Critical Assembly.

## CRITICAL EXPERIMENTS

### Reflector-Moderated Reactor

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The critical experiment program for the Reflector-Moderated Reactor has been altered to provide more fundamental information than did the previous experiments. The earlier work was designed for direct measurements on rough mockups of possible reactors with the purpose of establishing design parameters of those reactors. In general, these mockups were of complicated geometry and usually contained materials unique to the unit under study. The effort in the immediate future will be centered on reflector-moderated assemblies in order to check consistency with calculations and the fundamental constants on variations of simple geometry and materials which may be made. These results should also aid in the evaluation of previous reflector-moderated critical assemblies.

It is planned to build first a basic reflector-moderated reactor with distinct fuel and reflector regions. The first region is to contain uranium and a fluorocarbon plastic (Teflon) to simulate the fluoride fuels, and the second region is to be the beryllium reflector. The first region is to be rhombocuboctahedral in shape (essentially a cube with the edges and corners cut away), which approximates a sphere within the limitations imposed by the shape of the available beryllium. The purpose of this first experiment is to check machine calculations.

The program is set up so that either of two alternatives may be followed, depending on the results from the first assembly. If the experiments confirm the theoretical calculations to a sufficient degree, three-region octahedrons, having a beryllium "island" separated from the reflector by the fuel, will be built. The first of these assemblies will have no Inconel core shells, the second one will include Inconel, and the final one will mock up the reactor with end ducts.

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In the event that poor agreement between theory and the first experiment is found, a second two-region assembly of different core size will be constructed. In all cases the fuel region will be built of alternating sheets of uranium metal and Teflon, permitting some variation in the uranium density. The uranium sheets are to be 0.004 in. thick and will be coated with a protective film to reduce surface oxidation. The Teflon sheets will be  $\frac{1}{16}$  and  $\frac{1}{32}$  in. thick, making it possible to achieve a fairly homogeneous distribution of the uranium. The reflector and reflector-moderator will be made of beryllium metal.

### Supercritical Water Reactor

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A preliminary assembly of the Supercritical Water Reactor is being constructed in a 38-in. square cylindrical aluminum tank into which an organic liquid (furfural,  $C_5H_4O_2$ ), having a hydrogen density similar to that of water in the supercritical state, can be pumped to serve as the neutron reflector and as part of the moderator. The fissionable material, enriched uranium in an aqueous solution of  $UO_2F_2$ , is contained in 1-in.-dia stainless steel tubes located in the tank in a pattern designed to give a uniform radial thermal-neutron flux. The ratio of stainless steel to uranium in the core is varied by loading  $\frac{3}{16}$ -in.-OD steel tubes into the  $UO_2F_2$  solution.

The apparatus has been assembled, tested, and made initially critical with no stainless steel inserts in the fuel and with an effective loading of about 5 kg of  $U^{235}$ . In a series of experiments currently in progress, the height of the organic liquid reflector and moderator (assumed to be the effective core height) is being measured as a function of the number of fuel tubes loaded at increasing ratios of stainless steel to uranium in the core. In this manner, the designed core dimensions and core composition are being approached.

## 3. HIGH-VOLTAGE PHYSICS

VARIATION OF  $\nu(E)$  FOR  $U^{235}$  IN ENERGY  
RANGE OF 0 TO 14 Mev

J. L. Fowler

The number of prompt neutrons per  $U^{235}$  fission has been investigated as a function of the energy of the neutrons which produce the fission. The experiment, which detects fission neutrons in coincidence with fission fragments,<sup>1</sup> has been adapted for use with  $D(d,n)He^3$ ,  $Li^7(p,n)Be^7$ , and  $T(d,n)He^4$  neutron sources on the ORNL Van de Graaff machines and the Cockcroft-Walton machine.

Figure 3.1 is a drawing of the apparatus set up to utilize the D-D neutrons produced by deuterons from the 2.5-Mev Van de Graaff. The analyzed deuteron beam was incident upon the 0.05-mil nickel window of the deuterium gas cell. Runs were made with the energy of the beam incident upon the entrance foil equal to 2.00 and 1.00 Mev. After corrections for the deuteron energy loss in the foil and in the deuterium gas (at  $\frac{1}{2}$  atm pressure) were made, the neutron energies were calculated to be 5.0 and 3.7 Mev, respectively, for these runs. The neutron background found by substituting He for  $D_2$  in the gas cell or by allowing the beam to

strike a quartz viewer in front of this cell was about 10%.

For the  $Li^7(p,n)Be^7$  neutron source the protons were accelerated by the ORNL 5.5-Mev Van de Graaff. Here the target was lithium evaporated on a platinum disk. A measurement of the neutron yield at threshold indicated a target thickness of 76 kev. Runs made with protons of energies 2.265 and 4.410 Mev gave monoenergetic neutrons with average energies of 0.5 and 2.7 Mev, respectively. For such a neutron source, background due to the proton beam is negligible.

For a measurement at 14-Mev neutron energy, the  $T(d,n)He^4$  reaction was produced by deuterons from the Cockcroft-Walton machine. A thick zirconium tritide target was bombarded with 300-kev deuterons, and the neutrons emitted at 90 deg to the beam direction were used for the experiment. Since the cross section for the  $T(d,n)He^4$  reaction is much larger than other neutron-producing cross sections at this energy, background neutrons from the deuteron beam were negligible.

For all the neutron sources, including the Po-Be source described earlier,<sup>1</sup> the experimental procedure was similar. The fast neutrons produced fission in the 0.3 mg/sq cm layers of  $U^{235}$  plated on the surfaces of two concentric cylinders of an annular fission counter. A steel shadow cone

<sup>1</sup>J. L. Fowler, *Phys. Semiann. Prog. Rep. Sept. 10, 1953, ORNL-1630, p 38.*

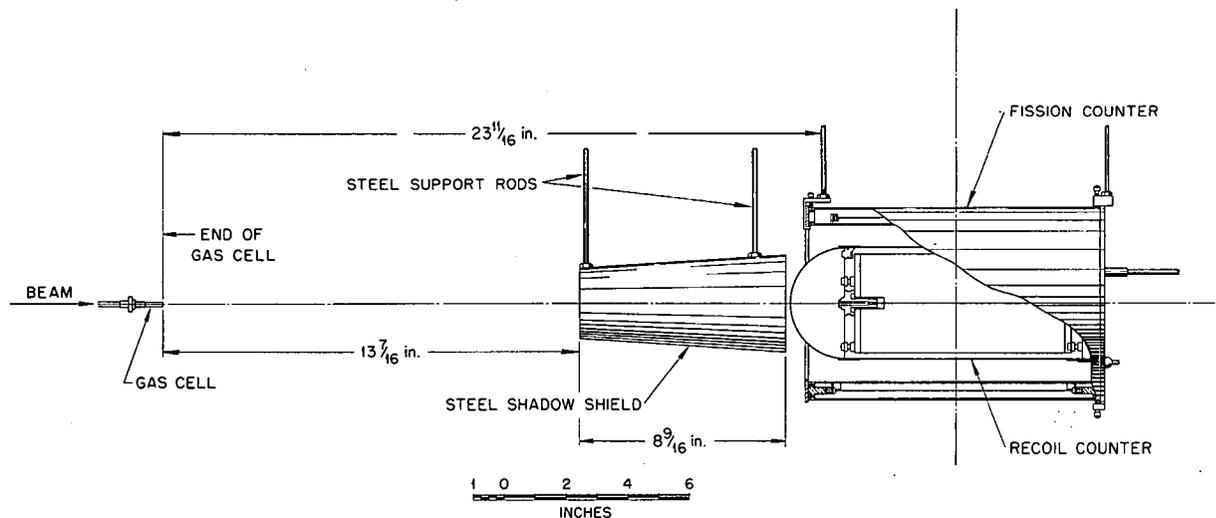
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Fig. 3.1 Experimental Arrangement for Neutrons per Fission Experiment.

shielded a proton-recoil counter mounted inside the fission counter. Fission neutrons detected by the recoil counter were counted in coincidence with fission-fragment pulses. Runs with neutrons slowed to thermal energies with paraffin allowed a calibration of the apparatus in terms of  $\nu$  for thermal fission.

The proton-recoil counter used in these experiments was filled with propane gas to a pressure between 0.3 and 0.4 atm. A 4-mil polyethylene foil mounted on the inside surface of the insulated cylindrical cathode of the counter increased its relative efficiency for detecting higher energy neutrons. The setting of the counter bias (100-300 kev) was accomplished by measuring the proton-recoil spectrum produced by monoenergetic neutrons from the  $\text{Li}(p,n)$  source on the 5.5-Mev Van de Graaff machine. A rough check of the relative efficiency of the recoil counter, made by use of the  $\text{T}(p,n)$  neutron source in this machine, indicated that the efficiency is constant to about  $\pm 15\%$  from 0.4 to 3.5 Mev. Theoretical considerations<sup>2</sup> show that it should also be constant to this accuracy to an energy of about 8 Mev. Thus the experiment does not depend critically upon the possible change of fission-neutron energy spectra with fission excitation energy. Measurements with a  $\text{Co}^{60}$  gamma-ray source also indicated that under the conditions in which the recoil counter was used during most of this experiment the counter was approximately 200 times as efficient for fission neutrons as it was for  $\text{Co}^{60}$  gamma rays.

The bias of the fission counter was set at the minimum between alpha background and fission-fragment pulses. At the position of the beams, about 5 ft above the grating floors of the scattering rooms, the fissions due to neutrons below the cadmium cutoff were approximately equal to fissions due to fast neutrons. The counter assembly, therefore, was surrounded with cadmium, and it was found that with a Po-Be source the fission counts varied correctly with distance from the source, indicating that a negligible number of fissions were caused by scattered neutrons.

In performing the experiment for studying  $\nu$ , the apparatus was set up, as shown in Fig. 3.1, in front of one or the other of the neutron sources. The fission counts, the recoil counts, and the

coincidence counts were recorded on tapes. Runs were made at approximately constant neutron intensity. This intensity was changed for different runs; so the coincidence rate varied from about 0.1 to 1 count/min. The coincidence resolving time, which for some of the experiment was calculated from the measured gate widths, was checked by the varying beam intensity in the case of the 0.50-Mev neutron-energy runs. The experimentally determined resolving time,  $2.2 \pm 0.4 \times 10^{-6}$  sec, agreed with the resolving time as found by measuring the gate widths ( $2.5 \times 10^{-6}$  sec) within the error of the two measurements. For the 0.5-, 3.7-, and 14-Mev data the accidental coincidence correction was made by use of the measured value of  $2.2 \pm 0.4 \times 10^{-6}$  sec for the resolving time. For the other data in which the accidental rate was of the order of 15%, the gate width resolving time was used.

The runs with fast neutrons were alternated with runs made with thermal neutrons (produced by surrounding a fast-neutron source with paraffin) and with the front cadmium cover removed from the counters. In some cases this fast-neutron source was the nuclear reaction produced by the particles accelerated by the machine. More often it was a Po-Be source. The real coincidences per fission were the same in the two cases within the statistics of the measurements. The quantity computed for each run was the real number of coincidence neutron counts per fission, and the ratio of this quantity for fast and for thermal runs is the ratio of fast  $\nu(E)$  to thermal  $\nu$  of  $\text{U}^{235}$ . The data were corrected for the effect (less than 6%) of background neutrons which produce a recoil in the neutron detector and then give a fission in the annular counter. This correction was calculated from the counting rate in the recoil-neutron detector and the quantity of uranium on the fission detector.

Figure 3.2 shows the experimental results. The errors shown are statistical errors of the measurements. The error in the experiment due to non-isotropic emission of the fission neutrons has been estimated to be less than  $2\frac{1}{2}\%$  and hence has been neglected. This departure from isotropy arises from the directional correlation of fission fragments with the fast source neutrons<sup>3</sup> and the directional

<sup>2</sup>G. S. Hurst, R. H. Ritchie, and H. N. Wilson, *Rev. Sci. Instr.* 22, 981 (1951).

<sup>3</sup>J. E. Brolley, Jr. and W. C. Dickinson, *Angular Distribution of Fragments from Neutron-Induced Fission*, LA-1552 (May 7, 1953).

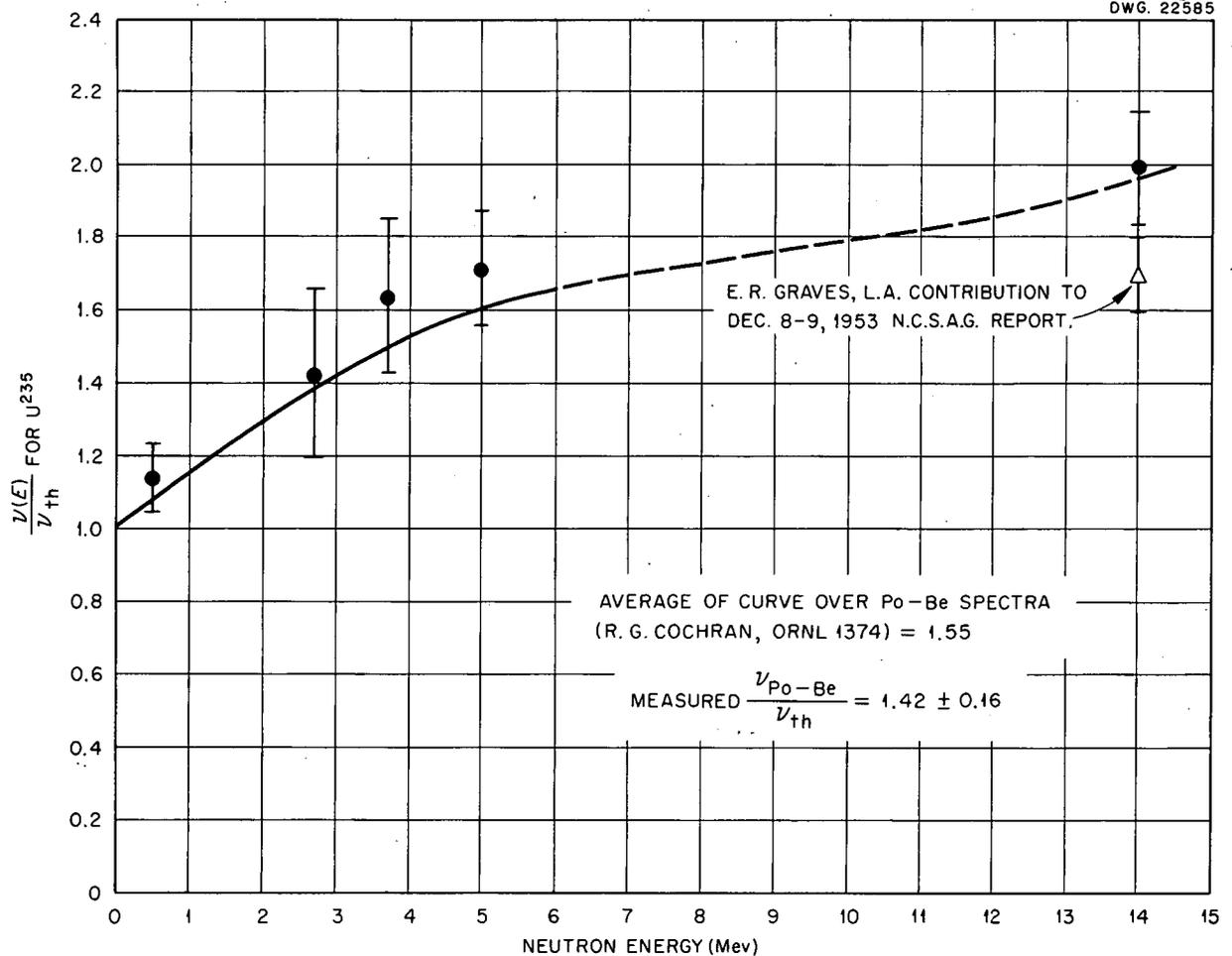


Fig. 3.2. Variation of  $\nu(E)/\nu_{th}$  for  $U^{235}$ .

correlation of fission neutrons with fission fragments.<sup>4</sup> While other sources of errors are being investigated, it is felt that the order of magnitude of the systematic error of the experiment is probably the same as, or less than, that of the statistical errors. The measurements for Po-Be neutrons give the value  $\nu_{Po-Be}/\nu_{thermal} = 1.42 \pm 0.16$  which is in agreement with the preliminary value given previously.<sup>1</sup> The average of the curve over the Po-Be spectra as measured by R. G. Cochran<sup>5</sup>

<sup>4</sup>R. R. Wilson, *Phys. Rev.* 72, 189 (1947); J. S. Fraser, *Phys. Rev.* 88, 536 (1952).

<sup>5</sup>R. G. Cochran, *Phys. Quar. Prog. Rep.* Dec. 20, 1951, ORNL-1374, p 3.

gives 1.55. The experimental curve is somewhat higher than the curve estimated from an evaporation theory of the variation of  $\nu$  with energy.<sup>6</sup>

The author would like to acknowledge the assistance of the Instrument Division of ORNL, in particular R. E. Zedler who saw to the details of the design of the counters. H. O. Finley of the Y-12 organization plated the  $U^{235}$  on the fission counter. Thanks are also due to the members of the ORNL High Voltage Group for their aid in operating the accelerators used in this experiment.

<sup>6</sup>J. L. Fowler, *Phys. Quar. Prog. Rep.* Dec. 20, 1951, ORNL-1374, p 14.

4. NEUTRON TIME-OF-FLIGHT SPECTROMETER

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CROSS SECTION OF Xe<sup>135</sup>

The total cross section of Xe<sup>135</sup> has been measured down to an energy of 0.01 ev. Figure 4.1 is a macroscopic cross-section plot computed from two representative runs, each of which took about 20 min. A replot of the same data, with  $E$  vs  $\Sigma\sqrt{E}$  shown in Fig. 4.2, permits an estimate of  $E_0 = 0.081$  ev and  $\Gamma = 0.11$  ev. The transmission of xenon was measured above 2 ev with a resolution of 0.6  $\mu$ sec/m. The only resonance observed was at 14.5 ev. The mass analysis of the sample shows that there is a relative enrichment of Xe<sup>134</sup> and Xe<sup>136</sup> compared with lighter stable isotopes;

<sup>1</sup>D. J. Hughes et al., *Neutron Cross Sections*, AECU-2040 (May 15, 1952).

so the resonance reported in AECU-2040<sup>1</sup> for stable xenon is attributed to one of these isotopes.

The calibration of the quantity of xenon used in the transmission measurements will be done mass spectrographically from the residual cesium in the sample holder. Since over 4000 transmissions were measured, the data will be reduced by the ORACLE. Final results will be the subject of a separate report.

The two 500-curie xenon samples used in these measurements were purified and loaded in sample holders by G. W. Parker and his associates in the Chemistry Division from the off gases of the HRE. T. Stephenson, F. J. Muckenthaler, and F. N. Watson of the Physics Division assisted in making transmission measurements.

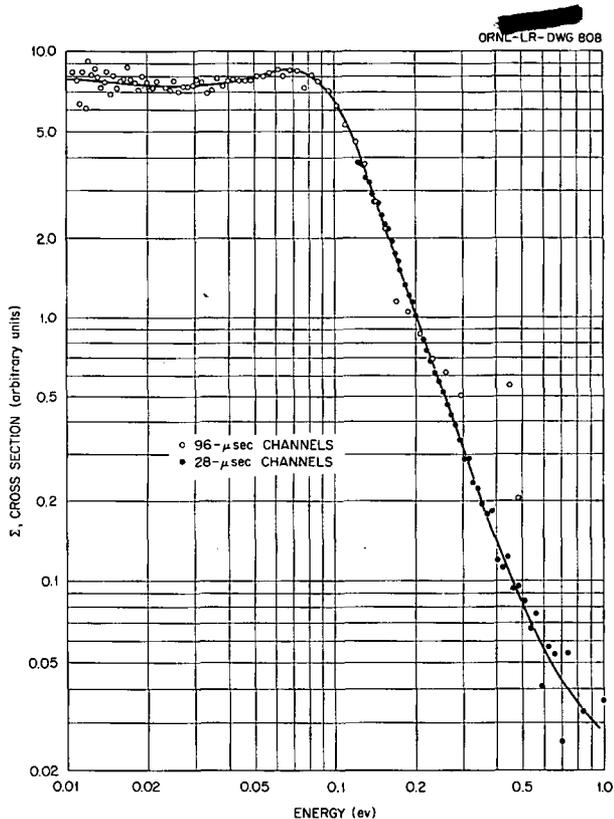


Fig. 4.1. The Total Relative Cross Section of Xe<sup>135</sup>.

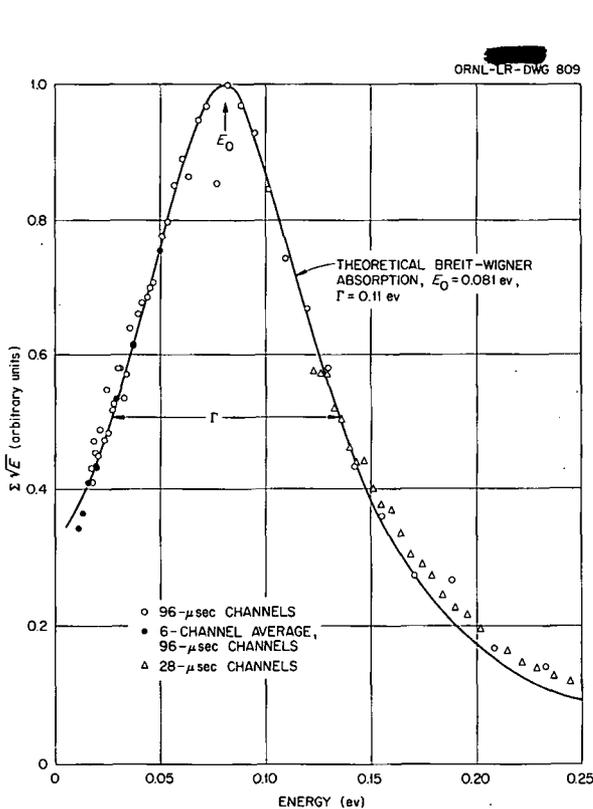


Fig. 4.2. The Parameter  $\Sigma\sqrt{E}$  for the Xe<sup>135</sup> Resonance.

**APPARATUS MODIFICATION**

The uniformity of the channel gating intervals has been improved. This modification permits much more rapid accumulation of background data and

allows narrowing the channel width for future increases of resolution. A new shutter which has been completed will have lower and more uniform background, as well as slightly improved resolution.