MASTER

OAK RIDGE NATIONAL LABORATORY

Operated by
UNION CARBIDE NUCLEAR COMPANY
Division of Union Carbide Corporation Post Office Box X
Oak Ridge, TennesseeExternal Transmittal
AuthorizedORNL
CENTRAL FILES NUMBER

58-8-1

DATE: August 14, 1958

COPY NO. 65

SUBJECT: FAST-NEUTRON FLUX MEASUREMENTS IN THE ORNL
GRAPHITE REACTOR SHIELD

TO: Distribution

AUTHORS: T. V. Blosser J. F. Nichols* T. O. P. Speidel*
 G. W. Bond* R. C. Reid* D. W. Vroom*
 L. A. Lee* A. B. Reynolds* M. A. Welt*
 D. T. Morgan*

Abstract

In a study of the nuclear and physical properties of the concrete shield of the ORNL Graphite Reactor measurements were made of the fast-neutron fluxes in a hole through the shield. These fluxes were not included in the formal report of the study (ORNL-2195) since several uncertainties associated with the measurements made the absolute values unreliable. The attenuation curves at one energy may be reasonably accurate, however. The results are reported here as a matter of record.

*MIT Engineering Practice School, Gaseous Diffusion Plant, Oak Ridge, Tennessee.

NOTICE

This document contains information of a preliminary nature and was prepared primarily for internal use at the Oak Ridge National Laboratory. It is subject to revision or correction and therefore does not represent a final report.

The information is not to be abstracted, reprinted or otherwise given public dissemination without the approval of the ORNL patent branch, Legal and Information Control Department.

611 001

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

FAST-NEUTRON FLUX MEASUREMENTS IN THE ORNL GRAPHITE REACTOR SHIELD

During the investigation of the nuclear and physical properties of the ORNL Graphite Reactor shield reported in ORNL-2195 (Ref. 1), threshold detectors were employed to obtain an insight into the fast-neutron energy distribution at several shield thicknesses. The results of these measurements were not included in ORNL-2195 because the absolute values of the flux levels measured cannot be regarded as reliable for several reasons noted below, although the attenuation curves at one energy may be reasonably accurate. The results of this investigation are presented here as a matter of record. No attempt will be made to redescribe the experiment since this was given in detail in ORNL-2195.

Experimental Results

The measured fast-neutron fluxes (neutrons/cm²·sec) above the "effective threshold energy" are presented in Table 1 and Fig. 1 as a function of the shield thickness. From each of the curves in Fig. 1 a value for the relaxation length was obtained. The average of the relaxation lengths in the berytes-haydite concrete was approximately 9.9 cm. This relaxation length is close to the fast-neutron dose-rate relaxation length of 10.2 cm, which was expected since these neutrons are the source of the measured fast-neutron dose rate in the shield.

Further study of Fig. 1 indicates that the integrated flux as measured with the $S^{32}(n,p)$ reaction ($E_{\text{eff}} = 2.9$ Mev) is a factor of about 75 less than the flux as measured with the $U^{238}(n,f)$ reaction ($E_{\text{eff}} = 1.5$ Mev). Such a drastic variation in flux in such a short energy interval is difficult to believe; however, checks made of the calibration of the foils used for both of these reactions demonstrated no apparent discrepancies. Additional investigations will be carried out when possible in order to ascertain with greater certainty whether a large change in flux actually does occur between 1.5 and 2.9 Mev.

1. T. V. Blosser, et al., "A Study of the Nuclear and Physical Properties of the ORNL Graphite Reactor Shield," ORNL-2195 (1958).

Table 1. Fast-Neutron Flux above "Effective Threshold Energy" as a Function of Shield Thickness

Shield Thickness (cm)	$\int_{E_{eff}}^{\infty} \phi(E) dE$, Fast-Neutron Flux above E_{eff} (neutrons/cm ² .sec)						
	Pu ²³⁹ Reaction, $E_{eff} = 360$ ev	Np ²³⁷ Reaction, $E_{eff} = 0.7$ Mev	U ²³⁸ Reaction, $E_{eff} = 1.5$ Mev	S ³² (n,p)P ³² Reaction, $E_{eff} = 2.9$ Mev	Al ²⁷ (n,p)Mg ²⁷ Reaction, $E_{eff} = 4.6$ Mev	Mg ²⁴ (n,p)Na ²⁴ Reaction, $E_{eff} = 6.3$ Mev	Al ²⁷ (n, α)Na ²⁴ Reaction, $E_{eff} = 8.1$ Mev
0					4.69×10^6	6.31×10^5	4.26×10^5
3.8				1.06×10^7			
5.0	1.35×10^9	8.07×10^8	7.01×10^8				
18.8					5.24×10^5	1.38×10^5	6.58×10^4
22.6				2.36×10^6			
24.0	2.33×10^8	1.84×10^8	1.72×10^8				
32.1					2.19×10^5	3.53×10^4	1.45×10^4
37.1	4.06×10^7	3.59×10^7	3.90×10^7				
48.3				1.91×10^5			
49.2	9.90×10^6	9.20×10^6	1.14×10^7				

UNCLASSIFIED
2-01-056-0-424

Fig 1. Fast Neutron Flux Above "Effective Threshold Energy" vs Shield Thickness

In order to calculate fast-neutron flux values from the measured threshold foil activities, it was necessary to assume that the cross section of each threshold detector could be represented as a function of energy by a step function, i.e., the cross section was zero below some "effective threshold energy" (discussed below) and was constant above this energy. Although in nuclear reactions such ideal cross sections do not exist, some reactions, such as the fast fission of Np^{237} , U^{238} , and Pu^{239} , the $\text{S}^{32}(\text{n,p})\text{P}^{32}$ reactions, and to a lesser degree the $\text{Al}^{27}(\text{n,p})\text{Mg}^{27}$, $\text{Al}^{27}(\text{n},\alpha)\text{Na}^{24}$, and $\text{Mg}^{24}(\text{n,p})\text{Na}^{24}$ reactions, do approach this ideal condition. Cross-section data for the fast fission reactions and for the $\text{S}^{32}(\text{n,p})\text{P}^{32}$ reaction are fairly well known.² Unfortunately, adequate cross-section data for the aluminum and magnesium reactions are not available. Furthermore, the assumption that the cross section of the $\text{Al}^{27}(\text{n,p})\text{Mg}^{27}$ reaction is a step function of the energy is particularly uncertain since the $\text{Al}^{27}(\text{n},\alpha)\text{Na}^{24}$ reaction soon begins to compete with the n,p reaction.

The curves in Fig. 1 for the Np^{237} , U^{238} , and Pu^{239} fission reactions become interlaced at shield thicknesses greater than 2 ft. The reason for this interlacing was not definitely determined; it may simply be the result of statistical error due to low counting rates. Another possible cause may be that while the fission cross sections are represented by step functions, the actual cross sections start to increase at higher energies (> 5 Mev) and the neutron spectrum is hardening as the shield thickness increases.

A further limitation on the validity of the absolute values of the reported fast-neutron fluxes results from the assumption that the neutron spectrum is of the same shape as the fission-neutron spectrum, which is done in order to obtain effective threshold energies and cross sections as discussed below.

-
2. G. S. Hurst, J. A. Harter, P. N. Hensley, W. A. Mills, M. Slater, and P. W. Reinhardt, "Techniques of Measuring Neutron Spectra with Thermal Detectors - Tissue Dose Determination," Rev. Sci. Instr. **27**, 153 (1956).

Calculation of Fluxes from the Measurements

The theory of determining fast-neutron fluxes with threshold detectors is discussed in detail by Hughes,³ Trice,⁴ Shor and Cassidy,⁵ Cohen,⁶ and others; the use of fast-neutron fission threshold detectors and the $S^{32}(n,p)P^{32}$ reaction is discussed by Hurst.² The methods used in the present investigation are briefly outlined below.

When threshold detectors are irradiated in a fast-neutron flux, the activity induced by neutrons above the true threshold energy for the nuclear reaction is represented by the following proportionality

$$A \propto \int_{E_t}^{\infty} \phi(E) \sigma(E) dE$$

where

$\phi(E)$ = neutron flux per unit energy,

$\sigma(E)$ = activation cross section as a function of energy,

E_t = true threshold energy.

By approximating the cross section, $\sigma(E)$, by a step function such that the cross section is zero below an "effective threshold energy," E_{eff} , and equal to the constant $\bar{\sigma}$ above this effective threshold, the above relation for the activity becomes:

$$A \propto \bar{\sigma} \int_{E_{eff}}^{\infty} \phi(E) dE$$

3. D. J. Hughes, "Pile Neutron Research," Addison-Wesley Publishing Co., Cambridge, Mass. (1953).
4. J. B. Trice, "Fast-Neutron Flux Measurements in E-25 of the Brookhaven Graphite Reactor," ORNL-CF-55-7-130 (1955).
5. S. W. W. Shor and J. M. Cassidy, "Effect of Water Soaking on Shielding Properties of the Oak Ridge Pile Shield," ORNL-203 (1949).
6. B. L. Cohen, "High Energy Neutron Threshold Detectors," Nucleonics 8, No. 2, 29 (1951).

The integral, $\int_{E_{\text{eff}}}^{\infty} \phi(E) dE$, is the value reported in Table 1 and Fig. 1,

where it is referred to as the "fast-neutron flux above the effective threshold energy."

Methods for defining E_{eff} and $\bar{\sigma}$ are given in Refs. 2-6. In each of these references the values for E_{eff} and $\bar{\sigma}$ are valid only when the detectors are irradiated by fission-spectrum neutrons. Since the spectrum in the shield may be significantly different from the fission spectrum, a serious limitation is imposed on the validity of the absolute flux values reported here. The values of E_{eff} and $\bar{\sigma}$ for the threshold reactions used in this investigation are given in Table 2.

Each aluminum and magnesium foil was surrounded by a 30-mil-thick cadmium cover when irradiated to reduce thermal-neutron activation. These foils were counted on a beta proportional counter, which was calibrated by comparison with a 4 π ion chamber. The fast-neutron flux above the effective threshold energy was determined from the following relation:

$$\phi_f = \frac{AE_0}{N_0 \bar{\sigma} (1 - e^{-\lambda t})}$$

where

ϕ_f = fast-neutron flux above the effective threshold energy,

A = activity at end of irradiation, counts/minute,

E_0 = counter efficiency, disintegrations $\cdot$ sec $^{-1}$ /counts $\cdot$ min $^{-1}$,

N_0 = number of target atoms in foil,

$\bar{\sigma}$ = effective cross section,

λ = decay constant,

t = time of irradiation.

The sulfur samples and the counting equipment used for determining this activity were supplied by G. S. Hurst of the ORNL Health Physics Division of Oak Ridge National Laboratory. The samples were surrounded by 2 cm of B 10

Table 2. Threshold Detectors Used for Neutron Flux Measurements

Detector Radiation	E_{eff} , Effective Threshold Energy	$\bar{\sigma}$ (barns)	Half-life of Product	Reference
$\text{Pu}^{237}(\text{n},\text{f})$	360 ev	1.94	-	a
$\text{Np}^{237}(\text{n},\text{f})$	0.75 Mev	1.4	-	a
$\text{U}^{238}(\text{n},\text{f})$	1.5 Mev	0.525	-	a
$\text{S}^{32}(\text{n},\text{p})\text{P}^{32}$	2.9 Mev	0.300	14.3 day	a
$\text{Al}^{27}(\text{n},\text{p})\text{Mg}^{27}$	4.6 Mev	0.039	9.8 min	b
$\text{Mg}^{24}(\text{n},\text{p})\text{Na}^{24}$	6.3 Mev	0.048	14.8 hr	b
$\text{Al}^{27}(\text{n},\alpha)\text{Na}^{24}$	8.1 Mev	0.111	14.8 hr	b

a. G. S. Hurst, J. A. Harter, P. N. Hensley, W. A. Mills, M. Slater, and P. W. Reinhardt, Rev. Sci. Instr. 27, 153 (1956).

b. J. B. Trice, "Fast-Neutron Flux Measurements in E-25 of the Brookhaven Graphite Reactor," ORNL-CP-55-7-130 (July 27, 1955).

when irradiated. The fast-neutron flux above the sulfur threshold energy was determined from the measured activity by a calibration similar to that described in Ref. 2.

The Np^{237} , U^{238} , and Pu^{239} foils, as well as the equipment used for counting these foils, were also supplied by Hurst, and these fission foils were also covered by 2 cm of B^{10} when irradiated. In order to determine the fast-neutron flux, the gamma-ray activity produced by neutron interaction with the foil was counted with a sodium iodide scintillation counter. The counter was biased at 1.13 Mev so that a small Co^{60} source could be conveniently used as a monitor. For calibration of the system, a bare Pu^{239} foil was exposed in a known thermal-neutron flux for the same period of time as each unknown exposure. The foils were counted and the counting rate was plotted as a function of the decay time for each foil. The decay curves for the calibration exposure and an unknown exposure were essentially parallel since the relative amounts of fission products present at any decay time were approximately constant for the two exposures. By comparing the two curves at a given decay time, the fast-neutron flux could be calculated from the relation:

$$\phi_f = \frac{\phi_T \sigma_T N_T C_f}{\bar{\sigma}_f N_f C_T}$$

where

ϕ_f = fast-neutron flux above the effective threshold energy, neutrons·sec⁻¹/cm²,

ϕ_T = known thermal-neutron flux, neutrons·sec⁻¹/cm²,

σ_T = thermal-neutron fission cross section for Pu^{239} ,

N_T = number of atoms in Pu^{239} calibration foil,

$\bar{\sigma}_f$ = effective fast-fission cross section,

N_f = number of atoms in foil irradiated in unknown flux,

C_T = counts per minute from the Pu^{239} calibration foil at a given decay time,

C_f = counts per minute from the foil exposed in the unknown flux at a given decay time.

It should be pointed out that, in the case of the Np^{237} , the reaction $\text{Np}^{237} + {}_0^1\text{n} \rightarrow \text{Np}^{238}$ is competitive with the fission reaction. The Np^{238} emits a 1.0-Mev gamma ray with a half-life of 2.1 days. A correction for this gamma ray must be made either by setting the bias of the detection equipment high enough so that the gamma-ray will not be counted, or, if a lower bias is used, by "peeling off" the 2.1-day half-life from the Np^{237} fission-product decay curve. The former method was used in this investigation.

INTERNAL DISTRIBUTION

1. E. P. Blizard	18. G. W. Keiholtz (K-25)
2. A. D. Callihan	19. G. T. Chapman
3. C. E. Clifford	20. E. R. Taylor (K-25)
4. F. L. Keller	21. G. deSaussure
5. F. C. Maienschein	22. V. R. Cain
6. A. Simon	23. F. J. Muckenthaler
7. W. Zobel	24. R. B. Murray
8. C. D. Zerby	25. J. M. Miller
9. D. K. Trubey	26. H. Bertini
10. S. K. Penny	27-31. T. V. Blosser
11. R. W. Peelle	32. K. M. Henry
12. L. B. Holland	33. H. Moran
13. L. Jung	34. M. J. Skinner
14. E. G. Silver	35-51. R. C. Reid, Chemical Engineering
15-16. A. B. Reynolds	Department, MIT, Cambridge,
17. C. D. Cagle	Massachusetts
	51. Laboratory Records Department
	Document Reference Section
	63-64. General Research Library
	65-79. TISE, Also

**DO NOT
PHOTOSTAT**