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NEUTRON FLUX AND SPECTRA
MEASUREMENTS IN THE SANDIA
PULSED REACTOR FACILITY (SPRF)

by

W H Buckalew

January 1962

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ABSTRACT

Neutron flux measurements were made in the reactor building of the Sandia Pulsed Reactor Facility using threshold detectors for the neutrons with energies greater than 10 kev. Bare and cadmium-covered gold were used for measurement of the thermal neutron fluxes. Midplane measurements were made as a function of distance outward from the reactor vertical centerline in several different directions within the building. In addition, detailed measurements were made on a midplane traverse around the circumference and a vertical traverse from the base of the safety screen over the domed top and to the base on the opposite side. All measurements were "free field" in the sense that no spectrum-perturbing experiments were near the reactor during these measurements. The fluxes above 10 kev follow a $(1/r)^2$ attenuation, whereas the thermal fluxes show very little attenuation with distance. The reactor is, to a good approximation, a point isotropic neutron emitter.

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NEUTRON FLUX AND SPECTRA MEASUREMENTS IN THE SANDIA PULSED REACTOR FACILITY (SPRF)

Introduction

Neutron measurements were made in the pulsed reactor building and on the safety screen of the pulsed reactor in order to determine the neutron yield of the reactor as a function of (1) distance from the reactor centerline, (2) direction in the reactor building, and (3) position on the reactor safety screen.

The reactor and reactor facilities have been described in detail by O'Brien.¹ Any description, here, is limited to that pertinent to the nuclear measurements. The reactor (Fig. 1) is located in the center of a thick-wall concrete building (Fig. 2) with the horizontal midplane of the reactor being 60 inches above the floor. The dome-shaped reactor building is 30 feet inside diameter and is 20 feet high at the inside apex.

An aluminum screen (Fig. 1) 9-1/2 inches in diameter and 16 inches high completely encloses the reactor nuclear material assembly. The radius of the domed cap of the screen is 5-1/2 inches. Since the screen is perforated with 3/8-inch-diameter circular holes on 9/16-inch centers the average density of the aluminum, which is 0.064 inch thick, is reduced to 0.23 gm/cm². The distance from the surface of the screen to the geometric center of the reactor is 4.75 inches on the horizontal midplane and is 4.1 inches along the vertical axis.

Midplane Flux Profile Measurements

The nuclear reactions, Pu²³⁹ (n, f), Np²³⁷ (n, f), U²³⁸ (n, f), and S³² (n, p) P³², were used to measure the integrated neutron fluxes above the effective energy thresholds² of 0.010, 0.7, 1.5, and 3.0 mev, respectively. Bare and cadmium-covered gold Au¹⁹⁷ (n, γ) Au¹⁹⁸ was used for the thermal neutron ($E \lesssim 0.4$ ev) flux determinations.

Threshold Foil Measurements

The neutron flux measurements at the reactor midplane (60 inches above the floor) were made radially outward from the reactor vertical centerline in the four directions shown in Fig. 3.

The neutron yield of the reactor has been verified experimentally to be a linear function of reactor temperature rise per burst, at least within the temperature limits of 30° to 110°C. The average reactor temperature rise per burst during these measurements was of the order of 80°C.

Since the leakage neutron yield is a linear function of reactor temperature rise per burst, the results of these measurements are reduced to integrated neutron flux per degree centigrade of reactor peak temperature rise, ΔT . The threshold detector measurements are grouped according to threshold energy (e.g., all data obtained with plutonium regardless of direction from the reactor are treated in one group) and plotted as integrated neutron flux, $E_n \geq E$, per degree $\Delta T(^{\circ}\text{C})$ versus distance.

Figures 4, 5, 6, 7a, and 7b are plots of the integrated neutron fluxes per $\Delta T(^{\circ}\text{C})$ $[\text{neutrons}/\text{cm}^2/\Delta T(^{\circ}\text{C})]$ for neutrons above the 0.01-, 0.7-, 1.5-, and 3.0-mev thresholds, respectively. The direction in the building of any measurement can be determined from Fig. 3 since the identity of each measurement is retained on the graph. The solid-line curve appearing in each graph is a $(1/r)^2$ attenuation which has been normalized to the measured values at 13.75 inches. In addition to the 13.75-inch normalizations, a normalization at 4.75 inches (on the safety screen) appears in the sulfur measurements given in Fig. 7b. The measured points are within 10 percent of either normalization; however, at distances less than 8.5 inches, the measured points more closely follow the 4.5-inch normalization. If a departure from the 13.75-inch normalization exists, it is a smooth transition in the region of 8 to 9 inches from the reactor center. Table I lists the integrated fluxes measured at 13.75 inches. It should be noted (Table I) from the normalized integrated flux tabulations that no appreciable difference exists between leakage neutron spectra measured inside and outside the reactor building.

TABLE I

Effective threshold-mev	0.01	0.70	1.50	3.0
Integrated flux per $\Delta T(^{\circ}\text{C})$ above threshold	2.26×10^{10}	1.45×10^{10}	0.739×10^{10}	0.298×10^{10}
Fraction of integrated flux above threshold, inside burst	1	0.640	0.327	0.132
Fraction of integrated flux above threshold, outside burst	1	0.642	0.356	0.135

Examination of Figs. 4 through 7 shows, for a given distance from the reactor, the fluxes above 10 kev are independent of direction in the reactor building (i. e., the midplane fluxes are isotropic). In addition, their close agreement with the $(1/r)^2$ attenuation allows for accurate (within 10%) prediction of the "free field" environment to be associated with any given ΔT over a wide latitude in distance from the reactor. Although the fission foil measurements only go to within 6 inches of the safety screen, their values can be deduced from the "close-in" sulfur measurements since the ratios of the fluxes above the various thresholds do not vary with distance.

Thermal Neutron Measurements

The reactor midplane measurements of the integrated thermal neutron fluxes, per ΔT , are plotted as a function of distance from the reactor centerline in Fig. 8. The direction of these traverses, in the reactor building, can be determined from Fig. 3. The dashed lines about the measured points are least squares fits to the measurements. The almost constant thermal integrated neutron fluxes (Fig. 6) strongly indicates that their origin is due to neutron scattering from the walls and floor of the reactor building.

Sulfur Neutron Measurements on the Safety Screen

The circumferential traverse consisted of positioning sulfur detectors on 2-inch centers, in the horizontal midplane of the reactor, around the safety screen. Results of the circumferential traverse and the positioning of the sulfur detectors are shown in Fig. 9. The spread in the measurements (Fig. 9) is most probably the result of positioning, i. e., deviation of the detectors from the horizontal midplane.

Figure 10 shows, schematically, the vertical traverse. In this traverse the detectors were on 1-inch centers. The results of the vertical traverse--integrated neutron flux, $E_n \geq 3$ mev, per $\Delta T(^{\circ}\text{C})$ versus distance--are plotted in Fig. 11. The distance, in this instance, is measured from the geometric center of the reactor (Fig. 10) and the $(1/r)^2$ attenuation is normalized to the midplane screen measurement. The geometric center is taken to be equidistant from the reactor base and the apex of the reactor dome on the vertical center-line. The reasonable agreement between the measured points and the $(1/r)^2$ attenuation of the vertical traverse coupled with the agreement of the horizontal traverse measurements with a $(1/r)^2$ attenuation clearly demonstrates that the reactor may be reasonably described as a point isotropic emitter.

The ratio of uncorrected to corrected sulfur measurements as a function of distance from the reactor center is plotted in Fig. 12. Corrected measurements, plotted in Fig. 10, account for the oblique incidence of neutrons on the detectors. The correction, the cosine of the angle shown in Fig. 11, is an increasing function of distance below the reactor midplane and should be taken into account when placing experiments below the horizontal midplane of the reactor.

Reference to Fig. 10 indicates any on-screen experiment will be strongly influenced by vertical position; and indeed, any sizeable object (e.g., a disc with 1-square-inch surface area and 1/2-inch thick) will most certainly receive a nonuniform exposure. Calculation of the vertical plane nonuniformity could be difficult, particularly if the samples were below the horizontal midplane of the reactor. In addition, the total flux incident on the front and rear surfaces of the disc, assuming it to be nonabsorbing and nonscattering, would differ by approximately 20 percent.

Estimation of Uncertainties

Uncertainties in the neutron flux mapping of SPR using threshold detectors are the result of the uncertainties in the following factors: (a) detector calibration, (b) foil counting, (c) detector size and weight, and (d) positioning of the detector.

The uncertainties in detector calibration³ arise from uncertainties in cross section, counting, neutron sources used for calibration, etc. The calibration uncertainties, $\frac{\Delta K}{K}$, for the various detectors are listed in Table II.

TABLE II

Detectors	S^{32}	Pu^{239}	Np^{237}	U^{238}	Au^{197}
Calibration uncertainty $\frac{\Delta K}{K}$	$\pm 8.8\%$	$\pm 13\%$	$\pm 13\%$	$\pm 13\%$	$\pm 10\%$

The integrated neutron flux above some threshold per burst $[\phi(E_n \geq E)]$ at some distance, r , from the reactor is related to the counting rate of a detector positioned at r by

$$\text{int. } \phi(E_n \geq E) = \frac{\text{CTK}}{\Delta T}$$

where

CT = total count observed in time interval ΔT

K = detector calibration factor.

Varying the terms in the above equation (ΔT is omitted since one must assume the counter is functioning properly) yields for the uncertainties in the calculated flux

$$\frac{\Delta\phi}{\phi} = \frac{\Delta CT}{CT} + \frac{\Delta K}{K}.$$

Since $\frac{\Delta K}{K}$ is systematic and does not effect the uncertainties between measurements, we need to consider only

$$\frac{\Delta CT}{CT}.$$

The extremes in counts, totaled over 10 minutes, were 30,000 counts for detectors 4.75 inches from the reactor to 400 counts for detectors at 72 inches from the reactor. The minimum and maximum uncertainties due to counting are then

$$\frac{\Delta CT}{CT} = \frac{100}{(CT)^{1/2}}; \text{ min} = \pm 0.58\%; \text{ max} = \pm 5\%.$$

Consider the uncertainties due to foil dimensions which are described by

$$\left[\left(\frac{2\Delta r}{r} \right)^2 + \left(\frac{\Delta w}{w} \right)^2 \right]^{1/2}.$$

This uncertainty is numerically, e.g., for sulfur,

$$\left[\left(\frac{0.003}{0.375} \right)^2 + \left(\frac{0.03}{3.20} \right)^2 \right]^{1/2} = \pm 1.2\%.$$

The mapping data clearly demonstrate the $(1/r)^2$ dependence of the fluxes above 10 keV; hence $\text{int.}\phi(E_n \geq E)$ is described by

$$\text{int.}\phi(E_n \geq E) = f(1/r^2).$$

Deviation of the measured points from the $(1/r)^2$ curve is given by

$$\frac{\Delta\phi}{\phi} = \frac{2\Delta r}{r}$$

and the ratio of the maximum and minimum values of ϕ that could be plotted at some point r due to an error in measurement Δr is

$$R = \frac{(r + 2\Delta r)^2}{(r - 2\Delta r)^2} \approx \frac{r^2 + 4r\Delta r}{r^2 - 4r\Delta r}$$

$$R \approx \frac{1 + \frac{4\Delta r}{r}}{1 - \frac{4\Delta r}{r}}.$$

Numerically for an uncertainty of 1/8 inch, a realistic assumption, at a distance of 5 inches from the reactor centerline,

$$R = \frac{1 + 0.1}{1 - 0.1} = 1.22 .$$

Hence it is not unreasonable to expect values of the flux measured close to the reactor to deviate by $\pm 10\%$ from the $(1/r)^2$ curves appearing in the graphs.

From the above analysis it is concluded that the uncertainties in "close-in" measurements are the result of positioning while at extreme distances counting errors are the main cause of the uncertainties. Variations in foil dimensions are negligible.

The uncertainties in the thermal flux measurement are the result of counting errors.

REFERENCES

1. O'Brien, P. D., Hazards Evaluation of the Sandia Pulsed Reactor Facility (SPRF), SC-4357A (RR) (1961).
2. Buckalew, W. H., Determination of Effective Energy Thresholds and Average Cross Sections for Neutron Threshold Detectors (to be published).
3. Buckalew, W. H., Calibration of Neutron Detecting Foils (to be published).

Fig. 1 -- Sandia Pulsed Reactor (SPR)

REACTOR BUILDING FLOOR PLAN

Fig. 2 -- Sandia Pulsed Reactor Building

SCHEMATIC OF MIDPLANE FLUX MEASUREMENTS

Fig. 3 -- Position of Measured Flux Traverse with Respect to Reactor and Building

Fig. 4 -- Integrated Pu^{239} Flux Profile

Fig. 5 -- Integrated Np^{237} Flux Profile

Fig. 6 -- Integrated U^{238} Flux Profile

Fig. 7a -- Integrated S^{32} Flux Profile

Fig. 7b -- Integrated S^{32} Flux Profile

Fig. 8 -- Integrated Thermal Neutron Flux Profiles

Fig. 9 -- Circumferential Symmetry of Neutron Fluxes of SPR

AXIAL SULFUR TRAVERSE

Fig. 10 -- Position of Detectors for Vertical Traverse

Fig. 11 -- Neutron Flux Vertical Traverse Measurements

Fig. 12 -- Ratio of Uncorrected to Corrected Vertical Flux Measurements

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