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FAST NEUTRON FLUX IN A MOCKUP
OF SM-1A CORE AND VESSEL

by

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ABSTRACT

Flux measurements were made in a cold (68 F) mockup of the SM-1A reactor core, thermal shields, and pressure vessel. Neutron flux distributions were mapped in radial and axial directions by means of detector foils to cover thermal, resonance, and fast neutron energy ranges. Multigroup calculations were also made for this cold mockup, using the P1MG-2 and PDQ (r z) - 2 codes. These results were correlated with hot (430-440 F) measurements and calculations of the fast flux above 1 Mev in the SM-1A reactor, in order to assist in prediction of radiation damage effects on the pressure vessel. The experimental results were up to 30 percent higher than calculated flux value. The best estimate of the fast flux above 1 Mev at the pressure-vessel surface was 4.1×10^3 n/cm²-sec-w, for the cold mockup (68 F), and 9.4×10^3 n/cm²-sec-w, for operating conditions (440 F).

CONTENTS

<u>LIST OF TABLES</u>	viii
1. <u>INTRODUCTION</u>	1
2. <u>DESCRIPTION OF FACILITY AND MOCKUP</u>	1
2.1 MOCKUP OF SM-1A CORE AND VESSEL	1
3. <u>EXPERIMENTAL TECHNIQUES</u>	2
3.1 DETECTORS	2
3.2 FLUX PERTURBATION BY FOILS	2
3.3 NEUTRON FLUX FROM FOIL ACTIVATION	3
3.3.1 Thermal Flux	5
3.3.2 Epithermal Flux	8
3.3.3 Fast Flux	10
3.4 COUNTING TECHNIQUES	11
3.5 REACTOR POWER LEVEL	12
4. <u>POWER-DISTRIBUTION MEASUREMENTS</u>	13
4.1 EXPERIMENTAL METHODS AND PROCEDURES	13
4.2 EXPERIMENTAL DATA AND RESULTS	14
5. <u>EXPERIMENTAL FLUX MEASUREMENTS</u>	17
5.1 CADMIUM RATIOS	17
5.2 THERMAL FLUX	17
5.3 EPITHERMAL FLUX	17
5.4 FAST FLUX	18
5.5 EXPERIMENTAL ACCURACY	18
6. <u>CALCULATIONAL MODELS</u>	30
6.1 P1MG-2	30
6.2 PDQ-2	31
6.3 MUFT-III	33
6.4 P ₃ CODE	33
6.5 CODE UTILIZATION	33
6.6 INPUT PARAMETERS FOR THE SM-1A at 68 F AND 440 F AT START OF LIFE	34

7.	<u>RESULTS OF CALCULATIONS</u>	40
7.1	SM-1A MOCKUP (68 F)	40
7.2	SM-1A REACTOR AT 440 F	40
8.	<u>EVALUATION AND COMPARISON OF RESULTS</u>	42
8.1	THERMAL FLUX	42
8.2	EPITHERMAL FLUX	42
8.3	FAST FLUX	43
8.4	FAST FLUX ABOUT 1 Mev	43
8.5	CONCLUSIONS	44
9.	<u>REFERENCES</u>	45
10.	<u>FIGURES</u>	47

TABLES

3-1	Detector Foil Properties	4
4-1	Normalized Average Power Generation of the SM-1A Mockup	16
5-1	Gold Foil Activity and Flux	20
5-2	Manganese Foil Activity and Flux	22
5-3	Vanadium Foil Activity and Flux	24
5-4	Cadmium Ratios and Epithermal Parameters	25
5-5	Sulphur-Foil Activity and Flux	26
5-6	Aluminum Foil Activity and Flux	28
5-7	Uranium-238 Foil Activity and Flux	29
6-1	Input Parameters for the SM-1A Mockup (68 F, 0 Mw-yr)	36
6-2	Input Parameters for the SM-1A Mockup (440 F, 0 Mw-yr)	37
6-3	Regionwise Number Densities of the SM-1A for P1MG-2 Calculations	38
6-4	Radial Description of SM-1A Reactor	39
7-1	SM-1A Vessel Wall Flux (440 F)	41

1. INTRODUCTION

Measurements and analysis of the fast neutron flux in Army nuclear reactors have been the subject of earlier studies.^{1, 2} These studies have been primarily related to radiation damage to pressure vessels. In order to extend these studies to cover the SM-1A reactor, an experimental program was continued at the Allis-Chalmers--operated Critical Facility at Schenectady, New York.

This program consisted of a series of flux measurements in a cold (68 F) mockup of the core, thermal shields, and vessel.

The measurements were carried out to allow correlation with irradiation of steel samples and flux monitors installed in the SM-1A reactor at startup.^{2, 3} At the same time, calculations were made of the fast-flux distributions for the cold (68 F) mockup, to compare with the experimental results taken from zero-power experiments performed on the core mockup. Taken together, these provide several checks on the predicted fast flux at hot (430 F to 440 F) operating conditions at the surface of the SM-1A reactor pressure vessel.

2. DESCRIPTION OF FACILITY AND MOCKUP

The Schenectady Critical Facility is owned by Alco Products, Inc., and operated by Allis-Chalmers Manufacturing Company. (A description of this facility is contained in an earlier report in this series.¹)

2.1 MOCKUP OF SM-1A CORE AND VESSEL

Figure 1 shows a plan view of the fuel-loading arrangement of the core mockup, the shielding mockup, and the pressure vessel mockup. Fuel elements on the east side of the core mockup were made of rejected plates from the SM-1A Core I or SM-1 Core II, having identical fuel and boron loadings. The remainder of the core mockup was made of plates and elements with fuel and boron loadings approximating the above cores in order to provide a balanced flux distribution.

The radial spacings of the core, shield, and pressure vessel mockup are also shown in Fig. 1. The mockup pressure vessel thickness was 2.75 in. with 0.25 in. of the thickness representing stainless-steel cladding on the inner surface. Foil strips (detectors) were inserted in the water gap, at an average radial distance of 22.5 in. Inside the core, foils were mounted on the fuel plates of elements 45 and 47. Outside the core, the foils were mounted on plastic strips inserted along the east-west plane of the core, so that radial and vertical flux distributions could be mapped in this plane.

Temporary stainless-steel end boxes were mounted on the mockup elements and additional steel blocks were added above the core to simulate the top plate and grid doors on top of the core support ring, as shown in Fig. 2. Figures 1 and 2 show the location of plastic strips 4 through 8. These strips correspond to the location of steel specimens and flux monitors being irradiated in the SM-1 and SM-1A reactors².

3. EXPERIMENTAL TECHNIQUES

3.1 DETECTORS

Neutron flux in the core and shield mockup was measured by the activation of detector foils. The thermal and epithermal fluxes were determined using both bare and cadmium-covered, gold manganese and vanadium foils. The fast flux was measured by foils using the following reactions and threshold energies:

U-238 (n, f), E_{eff} 1.57 Mev

S-32 (n, p) P-32, E_{eff} 3.0 Mev

Al-27 (n, p) Mg-27, E_{eff} 5.3 Mev

Al-27 (n, α), Na-24, E_{eff} 8.1 Mev

These foils have been discussed in a recent review article by J. Moteff,⁴ and the nuclear constants used in this report are taken from this paper. The physical properties of the foils used for this experiment are listed in Table 3-1.

Original plans called for fast-flux measurements to be made using neptunium-237 foils with a fission threshold energy of 0.75 Mev. Since the foils available had an unexpectedly high background of gamma radiation (extending up to 1.5 Mev in energy), and power levels at the Critical Facility were limited to less than 100 w, it was impossible to produce a measureable activation above this background and the use of neptunium-237 foils was discontinued.

3.2 FLUX PERTURBATION BY FOILS

The introduction of a foil into a flux field causes perturbations in the shape of the flux being measured. These perturbations are due to several conditions, including self-shielding in the foil, depression of the flux, and edge-effect corrections. These conditions have been recently reviewed in a number of articles, and correction factors have been calculated or measured. Hanna⁵ has made corrections for all these conditions in the following formula:

$$\frac{A}{A_0} = \frac{\alpha}{2\tau} \cdot \frac{(1 + \epsilon)}{(1 + g\alpha/2)}$$

where

- A = measured foil activity per unit mass
 A_0 = activity of an infinitely thin foil
 α = $1 - 2E_3(\tau) = 1 - 2 \int_1^\infty x^{-3} e^{-\tau x} dx$
 τ = foil thickness (mfp) = μt or Σt
 ϵ = edge-effect correction (see ref. 5)
 g = flux-depression parameter (see ref. 5).

The above formula and others in use are valid for monoenergetic neutrons, but the reference article⁵ shows how it may be simply modified to cover a $1/v$ absorber in a Maxwellian spectrum. The correction factors for thermal and fast detectors were calculated using the above formula, and are shown in Table 3-1.

For resonance absorbers, the self-shielding correction becomes very large, and requires special consideration. Corrected resonance integrals for gold and indium foils of various thicknesses have been calculated by Kelber.⁶ The resonance correction factors shown in Table 3-1 were obtained from his results. It should be noted that the gold foils used gave only 21 percent of the resonance activation that an infinitely thin foil would have shown.

3.3 NEUTRON FLUX FROM FOIL ACTIVATION

The activation of a detecting foil in a given neutron flux is given by the formula:

$$\frac{A(t)}{\text{eff}} = \sigma m \frac{N_0}{A} \phi (1 - e^{-\lambda T}) e^{-\lambda t}$$

where

- $A(t)$ = activity (count rate) of detector at time t , cps
 eff = counter efficiency for detector foil
 σ = activation cross section (10^{-24} cm^2), corrected for flux spectrum (a function of neutron energy)
 m = mass of foil, g
 N_0 = 0.6023×10^{24} , atoms/mole
 A = atomic weight, g/mole
 ϕ = flux being measured, $\text{n/cm}^2\text{-sec}$ (a function of neutron energy)
 λ = decay constant of product nuclide, sec^{-1}
 T = reactor operating time, sec
 t = shutdown or decay time, sec

TABLE 3-1
DETECTOR FOIL PROPERTIES

Foil	Au-197	Mn-55	V-51	Al-27	Al-27	S-32	U-238
Reaction	(n, γ)	(n, γ)	(n, γ)	(n, p)	(n, α)	(n, p)	(n, f)
Product	Au-198	Mn-56	V-52	Mg-27	Na-24	P-32	FP
T 1/2	2.697d	2.576h	3.76m	9.45m	14.97h	14.22d	
λ (sec ⁻¹)(10 ⁻⁷)	2.974(6)	7.474(5)	3.072(4)	1.224(4)	1.286(5)	5.641(7)	
radiation	γ	γ	γ	γ	γ	β	γ
E, Mev	0.412	0.845 1.81 2.13	1.435	0.843 1.015	1.368 2.754	1.71	
Counter eff (%)	4.15	4.58	2.75	3.25	5.10	3.11	
Cross sections:							
th σ_0 , barns	98.8	13.2	4.5				
res σ_{R1} , barns	1555	13.4	3.3				
fast σ_{eff} , barns				0.08	0.13	0.31	0.61
res E_0 , ev	4.9	337	4150				
fast E_{eff} , Mev				5.3	8.1	3.0	1.57
Mass, g	0.032	0.073	0.058	0.5845		0.85	0.304
Purity, %	100	80	100	100		95	99.6
Radius, in.	0.125	0.25	0.25	0.375		0.25	0.125
Thickness, in.:	0.002	0.003	0.003	0.030		0.187	0.020
τ , mfp	0.028	0.0072	0.0021	0.006		0.006	0.002
Flux perturbation A/Ao:							
th	0.930	0.978	0.990				
res	0.21	0.70	0.94				
fast				0.984	0.980	0.98	0.99

In a typical reactor medium of given scattering and absorbing properties, the neutrons have an energy spectrum ranging from the highest energies from fission (of the order of 10 Mev) down to those in thermal equilibrium with the medium. Flux measurements by detector foils may be separated into three energy ranges described below.

In the thermal range, the neutrons are assumed to be in equilibrium with the medium, and the flux is measured by a $1/v$ absorber, i.e., one whose cross section varies inversely with neutron velocity or its equivalents: square root of energy and square root of temperature. An upper limit of the thermal range is commonly taken at the cadmium cut-off energy, which is approximately 0.4 ev for a typical cadmium thickness of 0.020 in.

The epithermal range covers the slowing-down spectrum between the thermal equilibrium group and the fast or fission spectrum. The energy range extends from about 0.1 ev to 1 Mev, although other limits, such as the cadmium cutoff, are used for some purposes. The flux is measured by a resonance detector (gold, magnesium, or vanadium covered by cadmium), with a very high absorption cross section at one or more specific resonance energies.

The fast-flux range covers the fission spectrum from 1 to 10 Mev, but may be extended down to overlap the epithermal range. Threshold detectors sensitive to a reaction cross section above some specific threshold energy are employed. These high-energy reactions are usually of the (n, p) (n, α) or fission type. Reaction cross sections have usually been measured in a reactor fission spectrum, and converted to an effective cross section, σ_{eff} above the threshold energy, E_{eff} .⁴

3.3.1 Thermal Flux

The standard tabulated cross sections, (σ_0) are defined for a monoenergetic flux, (ϕ_0) , with a neutron velocity, $(v_0 = 2200 \text{ m/sec})$ which corresponds to an energy, $(E_0 = 0.0253 \text{ ev})$ or a neutron temperature, $(T_0 = 20.3 \text{ C } (293.6 \text{ K}))$. This is the modal velocity for a Maxwellian energy distribution in thermal equilibrium with the medium at this temperature.

In an absorbing medium, the flux is "hardened" by the selective absorption of lower energy neutrons, so that the Maxwellian temperature distribution is shifted to a higher effective neutron temperature T_n .⁷

where

$$\frac{T_n}{T_0} = 1 + \frac{A \Sigma_a}{\xi \Sigma_s}$$

For SM-1A core parameters at 20 C (68 F),

$$A = 0.854$$

$$\Sigma_a = 0.285 \text{ cm}^{-1}$$

$$\xi = 0.92$$

$$\Sigma_s = 2.10 \text{ cm}^{-1}$$

$$\text{giving } T_n/T_o = 1.26$$

It will be seen later that this is in very good agreement with results from cadmium ratio measurements.

The Maxwellian average thermal flux is:

$$\phi_m = \sqrt{\frac{4}{\pi} \frac{T_n}{T_o}} \phi_o = 1.265 \phi_o$$

This is in good agreement with the factor 1.24 used in APAE-95 for a Wigner-Wilkins spectrum.

The thermal cross sections for the foils in the flux spectrum being measured were obtained from the data of Westcott.⁸ Using his notation, the effective cross section is given by:

$$\hat{\sigma} = \sigma_o (g + r s)$$

where

g = factor (near unity) to correct for non $1/v$ cross section

r = epithermal neutron density fraction

s = tabulated form of resonance integral

In this form, the cross section includes both thermal and epithermal activations in the flux spectrum defined by the parameter r . If these two components (thermal and epithermal) are separated by using a cadmium filter, the effective cross section may be written:⁹

$$\hat{\sigma} = \sigma_o (g - r/K \sqrt{T_n/T_o} + r/K \sqrt{T_n/T_o} + r s)$$

where the term $r/K \sqrt{T_n/T_o}$ represents $1/v$ captures in the epithermal flux component. The first two terms thus give the thermal captures below the cadmium cut-off energy E_c , and the last two terms give the epithermal captures above E_c . The parameter K depends on the cadmium cut-off energy E_c , which depends on the cadmium thickness and geometry. For a cadmium thickness of 0.020 in. in an isotropic flux, K is given as 2.0728.⁹

If a bare foil is activated in the flux to be measured, its activation is given by the equation in Sec. 3.3, where the flux is ϕ_o and the cross section is $\hat{\sigma} = \sigma_o (g + r s)$. The value of the parameter r can be obtained from a cadmium ratio measurement, where

$$R_{cd} = \frac{\text{activity of bare foil}}{\text{activity of cadmium covered foil}}$$

From the expanded expression given above for $\hat{\sigma}$, we obtain:

$$R_{cd} = \frac{g + r s}{\frac{r}{K} \sqrt{\frac{T_n}{T_o}} + r s}$$

This expression may be solved to find $r \sqrt{T_n/T_o}$, where:

$$r \sqrt{\frac{T_o}{T_n}} = \sqrt{\frac{4}{\pi}} \frac{(\sigma_{R1} - \sigma_{1/v})}{\sigma_o}$$

and T_n is not necessarily known.⁹

It may sometimes be convenient to use the following expressions:

$$R_{cd} = \frac{A (\text{bare foil})}{A (\text{Cd covered foil})} = \frac{A (\text{sub-Cd}) + A (\text{epi-Cd})}{A (\text{epi-Cd})}$$

or

$$R_{cd} - 1 = \frac{A (\text{sub-Cd})}{A (\text{epi-Cd})}$$

or

$$\begin{aligned} \text{Cadmium difference} &= A (\text{bare foil}) - A (\text{Cd covered foil}) \\ &= A (\text{sub-Cd}) \\ &= A (\text{bare foil}) \times (R_{cd} - 1)/R_{cd} \end{aligned}$$

The latter expression may be used to determine the thermal flux, using a cross section of

$$\sigma(\text{sub-Cd}) = \sigma_0 (g - r/K \sqrt{T_n/T_0}).$$

The thermal flux so obtained is exactly the same as in the previous method. In APAE-95,¹ this latter method was used, but with the simple σ_0 cross section, instead of the corrected subcadmium cross section as given above.

The thermal flux for this report was obtained directly from the activation of bare foils, using the formula in Sec. 3.3 rewritten as:

$$\phi_m = \frac{1.265 A(t)/\text{eff}}{\sigma_0 (g + r s) m A/N_0 (1 - e^{-\lambda T}), e^{-\lambda t}}$$

3.3.2 Epithermal Flux

It is customary to assume that the flux in the slowing down region is inversely proportional to the neutron energy E . We may write:

$$\phi(E) = K_1/E$$

or

$$\int_{E_1}^{E_2} \phi(E) dE = \int_{E_1}^{E_2} K_1/E dE = K_1 \ln E \Big|_{E_1}^{E_2}$$

If the limits of integration are separated by a factor e ,

$$\int_{E_1}^{eE_1} \phi(E) dE = K_1 = \phi(E) \cdot E$$

Thus the flux per $\ln E$ energy interval, or per lethargy unit, is a constant (K_1), in accordance with the above assumption.

The expression for foil activation in Sec. 3.3, after writing K_2 for other constant parameters and saturation and decay factors, may be written as:

$$A_s = K_2 \int_0^{\infty} \sigma_a(E) \phi(E) dE = K_2 \phi_0 \sigma_0 (g + rs).$$

If the lower limit is taken as the cadmium cut-off E_c , we have only the epithermal activations:

$$A_{se} = K_2 \int_{E_c}^{\infty} \sigma_a(E) \phi(E) dE = K_1 K_2 \int_{E_c}^{\infty} \sigma_a(E) dE/E$$

This expression under the integral is known as the resonance integral σ_{RI} . Values for many materials, in infinitely thin samples, have been calculated and tabulated.^{7,8,10} The values used for gold, manganese and vanadium are listed in Table 3-1. The large self-shielding factors (A/A_0) for the resonance peaks are also given in Table 3-1. These were applied as correction factors to the tabulated resonance integrals.

The activation expression in Sec. 4.1 now gives the epithermal flux per ln E interval or $\phi(E) \cdot E = K_1$ when the resonance integral cross section for the activation A_e of a foil under a cadmium cover is used:

$$\phi(E) \cdot E = \frac{A_e(t)/\text{eff}}{\sigma_{RI} m A/N_0 (1 - e^{-\lambda T}) e^{-\lambda t}}$$

The epithermal flux is also related to the thermal flux, by the measured cadmium ratio:

$$R_{cd} = \frac{\phi_0 \sigma_0 (g + rs)}{\phi(E) \cdot E \sigma_{RI}}$$

or

$$\phi(E) \cdot E = \frac{\phi_0 \sigma_0 (g + rs)}{R_{cd} \sigma_{RI}} = \frac{\phi_0 \sigma_0 \left(g - \frac{r}{K} \sqrt{\frac{T_n}{T_0}} \right)}{(R_{cd} - 1) \sigma_{RI}}$$

This is equivalent to the expression used in APAE-95.¹ It is considered more satisfactory to relate the epithermal flux directly to the measured activations under a cadmium cover, rather than indirectly to the thermal flux and the cadmium ratio.

3.3.3 Fast Flux

The fast flux was measured by threshold detectors, as described in Sec. 3.1 and Table 3-1. A sharp cutoff in the reaction cross section is assumed to occur at the effective threshold energy, so the cross section is zero below this energy, and has a constant value, as tabulated, above the threshold.

The activation expression then gives the value of the integrated fast flux above the threshold energy as:

$$\phi_f = \frac{A(t)/\text{eff}}{\sigma_{\text{eff}} m N_0/A (1 - e^{-\lambda T}) e^{-\lambda t}}$$

For the sulphur detectors there was only one activation product, and the above formula was used directly. For aluminum, however, there were three products, Al-28, Mg-27, and Na-24, with half lives of 2.30 min, 9.45 min, and 14.97 hr, respectively. It was necessary to count one foil repeatedly to obtain a decay curve which fit the three individual decay curves. These curves were used to correct the total count rate and find the fraction which represented the product Mg-27. For the Na-24 count rate, it was simply a matter of waiting a few hours until the other two had decayed.

The U-238 fission foils were a special case, since the complex fission product activities were not related to the flux by a simple and precise formula. Several assumptions were made in relating the fast flux above the 1.57 Mev fission threshold to the thermal flux. The yield, energy, and decay scheme of the fission products were assumed to be similar for U-235 and U-238. The composition of the foils was given by the manufacturer as 0.4039 percent U-235 by weight. Then the ratio between thermal and fast activations is given by:

$$\frac{A_{\text{th}}}{A_f} = \frac{\phi_{\text{th}} m_{235} \sigma_{f235} A_{238}}{\phi_f m_{238} \sigma_{f238} A_{235}} = \frac{\phi_{\text{th}}}{\phi_f} \times 3.03$$

or

$$\frac{\phi_f}{A_f} = \frac{\phi_{\text{th}}}{A_{\text{th}}} \times 3.03 = F$$

Using the above assumptions, the factor F should be the same for thermal and fast activations, for a given activation time, T.

The foils were first activated without covers. After they had decayed, they were activated in the same positions with boron or cadmium covers. The boron covers were bulky spheres, about 1 in. in diameter, and could therefore be used in only a few locations such as the water reflector and the region above the upper core support plate. It was found that the activities of the cadmium-covered foil averaged about 1.55 times the activities of the boron-covered foils. Therefore activities of the cadmium-covered foils were corrected by this factor to obtain boron-covered activities, A_f , for all positions mapped. These values were then subtracted from the bare-foil activities to obtain the thermal activations, A_{th} , for each position. The thermal flux from the earlier measurements was then used to calculate the factor F . The fast flux was then obtained from the relation:

$$\phi_f = A_f F.$$

Considering all the assumptions and the practical difficulties in using the fission foils, they cannot be expected to give very accurate results in comparison with the other detectors.

3.4 COUNTING TECHNIQUES

The counting equipment was the same as that used in the earlier experiment, and is described in APAE-95.¹ The beta radiation from the sulphur detectors was counted by an end-window Geiger counter. The gamma radiation from all other foils was counted by a photo-multiplier and a single-channel differential analyzer.

The earlier calibrations¹ were checked using standard calibration sources. The beta-counter efficiency for the sulphur detectors was the same as that obtained earlier for a calibrated sulphur detector with the same geometry, so self shielding of the beta radiation was included. The gamma counter calibration curve is shown in Fig. 3, and agrees closely with the calculated interaction ratio for a similar crystal, as reported by Miller and Snow.¹¹ The discriminator bias setting was 0.25 Mev. for all gamma counting, and the efficiencies were determined for this setting. The counter efficiency is shown in Table 3-1 for each foil radiation product, taking into account the yield of each gamma energy in the decay schemes as reported by Strominger, Hollander, and Seaborg.¹²

The counting accuracy was limited by the statistical variations resulting from the low activities available. In only very few cases was it conveniently possible to obtain counts of the order of 10^4 , providing statistical variations of the order of 1 percent. For the low regions near the pressure vessel and above the core support plate, the count rates were very low, not much above background levels, and the statistical variations were of the order of 20 to 50 percent. This was particularly evident for the threshold reactions in aluminum, and also for the cadmium-covered foils. The data were generally discarded when the uncertainties exceeded 50 percent. Some runs in the low-flux regions were repeated at higher power (about 75 w) and for longer times in order to reduce statistical errors.

3.5 REACTOR POWER LEVEL

The relationship between reactor power level and the average thermal flux in the fuel is given by the following expression:

$$\text{Power, } w = \sigma_f M N_o / A \bar{\phi}_o \text{ (fissions/sec)} \times 3.12 \times 10^{-11} \text{ (w-sec/fission)}$$

or

$$\bar{\phi}_o / w = \frac{3.20 \times 10^{10}}{\sigma_f M N_o / A} \text{ (n/cm}^2\text{-sec.w)}$$

where

$$\sigma_f = 549.5 \text{ barns (fission cross section of U-235)}$$

$$M = 22382.92 \text{ g U-235 in SM-1A mockup core}$$

$$N_o = 0.6023 \times 10^{24} \text{ atoms/mole}$$

$$A = 235 \text{ g/mole}$$

The energy per fission is $3.12 \times 10^{-11} \text{ w-sec} = 195 \text{ Mev}$. The fission cross section $\sigma_f = \sigma_{fo} (g + r s)$ has been corrected for the epithermal flux spectrum as found from cadmium ratio measurements.⁸

The hardened Maxwellian average thermal flux is:

$$\phi_m = \sqrt{4 T_n / \pi T_o} \phi_o = 1.265 \phi_o$$

as defined in Sec. 3.3.1.

The resulting average thermal flux in the fuel is:

$$\bar{\phi}_m / w = 1.29 \times 10^6 \text{ (n/cm}^2\text{-sec-w)}$$

The above expression is equivalent to that used in APAE-95, Eq. 34.¹ The product $\sigma_f M N_o / A$ is equivalent to $\Sigma_f V$, but the mass of fuel and the microscopic cross section are known more precisely than the core volume and the macroscopic cross section. In the two-group expression, Σ_f is separated into thermal (Σ_f^{th}) and epithermal (Σ_f^{ep}/CF) components. Fissions above the cadmium cutoff energy contribute 14 to 15 percent of the total, and are in good agreement with analytical calculation. The epithermal flux is discussed in Sec. 3.3.2.

Because of the high absorption cross section of the fuel plates, the thermal flux at the plate surface, where the standard gold foils are exposed, is depressed by a factor $d=0.91$ in the fuel.¹³

The standard gold foil for each experimental run was exposed in element 34, plate j, at an elevation 4 in. above the core bottom. The mapping of the flux and power distribution in the core, described in Sec. 4, gave a factor of 1.543 for the flux at this position normalized to the core average of 1.

The power for each run was given by:

$$\text{Power, w} = \frac{\phi_m (\text{gold standard}) \times 0.91}{1.543 \times 1.29 \times 10^6}$$

All activation data for each run were divided by the power to convert the results to flux per watt.

4. POWER-DISTRIBUTION MEASUREMENTS

Power-distribution measurements were made on the core mockup to provide information for calculation of the absolute power level. The relative power distribution was obtained by uranium-foil activation measurements employing the experimental method and procedure described in Sec. 4.1. About 1600 data points were taken in one-half of the symmetrical core. These were processed to provide both the core average, and a detailed mapping of local flux. All power-distribution data were collected while operating the reactor with Rods A and B positioned at 19.00 in. and the five-rod bank withdrawn to 4.75 in.

4.1 EXPERIMENTAL METHODS AND PROCEDURES

Bare uranium-aluminum alloy fission foils were attached directly to the fuel plates and activated in the core to obtain relative power-distribution measurements. All foil data were corrected for radioactive decay and differences in foil masses. A reference foil was exposed at an identical core location during each activation experiment to provide a power normalization factor between runs. The reference foil for all power-distribution measurements in this report was located on the centerline of fuel plate i of element 34 and at an axial elevation 3 in. above the bottom of the active core. By taking an adequate number of data points and utilizing core symmetry, the relative-power distribution in the core was obtained. Each data point reported is normalized to a core average activation; thus, the core average becomes unity by definition.

Aluminum foils of 19 w/o fully enriched uranium, approximately 13 mg weight, 0.005 in. thickness, and 0.25 in. diameter, were used to map the fuel plates. The number, location, and symmetry of the fuel plates to be mapped were chosen to obtain complete radial and axial power traverses in the core. The activation foils were generally placed on the west side of plates a, e, j, and n, and the east side of plate r. The exceptions were the east boundary fuel elements, in which the activation foils were placed on the north side of fuel plates a, e, j, and n, and the south side of plate r. The foils were activated in the

core for 30 min at power levels of approximately 10 w, then removed and counted by two scintillation counters. Typical foil locations for a fuel plate are shown in Fig. 4, the fuel plate arrangement is shown in Fig. 5, and location of radial planes is given in Fig. 6.

The data were processed by the method as described in AP Note 246.¹⁴ This method yields activity of the different foils relative to the reference foil, allowing corrections for weight, counter background, and radioactive decay.

The core average was computed by two methods. In determining the normalization factor under Method A, the cell averages for the control elements have been weighted to include only that portion of each fuel plate which is actually inserted in the active core:

$$\begin{aligned} \text{Total Fuel} &= 38 \times 515.16 \text{ g (stationary elements)} \\ &\quad + 2 \times 401.12 \text{ g} \times \frac{18.125 \text{ in. (2 control elements)}}{20.875 \text{ in. (at 19 in. position)}} \\ &\quad + 5 \times 401.12 \text{ g} \times \frac{3.875 \text{ in. (5 control elements)}}{20.875 \text{ in. (at 4.75 in. position)}} \\ &= 20,644.92 \text{ g} \end{aligned}$$

$$\text{Normalization Factor} = \frac{20,644.92 \text{ g}}{515.16 \text{ g} \times 26.4280} = 1.5164$$

A typical calculation for obtaining the core average is presented in Appendix B of APAE 54, Supplement 1.¹⁵

A second core average was also determined by a similar method (Method B) with the exception that in determining the normalization factor, the cell averages for the control elements were based on their total fuel content, including the portion below the core:

$$\begin{aligned} \text{Total Fuel} &= 38 \times 515.16 \text{ g (stationary elements)} \\ &\quad + 7 \times 401.12 \text{ g (control elements)} \\ &= 22,383.92 \text{ g} \\ \text{Normalization Factor} &= \frac{22,383.92 \text{ g}}{515.16 \text{ g} \times 26.9846} = 1.6102 \end{aligned}$$

Each data point reported in this section is normalized to the average power generation determined by Method A.

4.2 EXPERIMENTAL DATA AND RESULTS

Figure 7 gives the complete normalized cell power averages in the core determined by both Method A and B.

The normalized average power generation and average relative activity for each individual fuel element is listed by both methods in Table 4-1.

The average relative power distributions along the axial traverses for the stationary and control rod fuel elements on the east side of the core are shown in Figs. 8 through 16.

Normalized power distributions along two central radial traverses are shown in Figs. 17 and 18. Figure 17 illustrates the north-south radial traverse parallel to the fuel plate surface across elements in positions 54 and 74 on plate j and control rods C and 1 in positions 44 and 64 on plate i. Figure 18 illustrates the east-west central radial traverse perpendicular to the fuel plate surface through the center of elements in positions 44, 45, 46, and 47. Figure 19 shows the power generation of the east, outer row of elements along the traverses perpendicular to the fuel plates at 10.08 in. and 8.82 in. from the north-south centerline of the core. All traverses were taken at the axial plane 3 in. above the core bottom, or 7.88 in. below the core center.

TABLE 4-1

NORMALIZED AVERAGE POWER GENERATION OF THE SM-1A MOCKUP

stationary element position	average relative activity		normalized power generation	
	A	B	A	B
12	5347	5347	0.811	0.861
13	5341	5341	0.810	0.860
14	5403	5403	0.819	0.870
15	5446	5446	0.826	0.877
16	5457	5457	0.827	0.879
21	5602	5602	0.849	0.902
22	7276	7276	1.103	1.172
23	7126	7126	1.081	1.147
25	7504	7504	1.138	1.208
26	7318	7318	1.110	1.178
27	5675	5675	0.861	0.914
31	5509	5509	0.835	0.887
32	7266	7266	1.102	1.170
34	7504	7504	1.138	1.208
35	8983	8983	1.362	1.446
36	7762	7762	1.177	1.250
37	5347	5347	0.811	0.861
41	5467	5467	0.829	0.880
43	7408	7408	1.123	1.193
45	7408	7408	1.123	1.193
47	5467	5467	0.829	0.880
51	5347	5347	0.811	0.861
52	7762	7762	1.177	1.250
53	8983	8983	1.362	1.446
54	7504	7504	1.138	1.208
56	7266	7266	1.102	1.170
57	5509	5509	0.835	0.887
61	5675	5675	0.861	0.914
62	7318	7318	1.110	1.178
63	7504	7504	1.138	1.208
65	7126	7126	1.081	1.147
66	7276	7276	1.103	1.172
67	5602	5602	0.849	0.902
72	5457	5457	0.827	0.879
73	5446	5446	0.826	0.877
74	5403	5403	0.819	0.870
75	5341	5341	0.810	0.860
76	5347	5347	0.811	0.861
CR-1-64	1075	1899	0.163	0.306
CR-2-42	1068	1865	0.162	0.300
CR-3-24	1075	1899	0.163	0.306
CR-4-46	1068	1865	0.162	0.300
CR-A-55	6481	7132	0.983	1.148
CR-B-33	6481	7132	0.983	1.148
CR-C-44	1550	2572	0.235	0.414
	26.4280	26.9846	40.075	43.450

5. EXPERIMENTAL FLUX MEASUREMENTS

5.1 CADMIUM RATIOS

The activity of the gold, manganese, and vanadium foils, both bare and cadmium covered, were corrected for decay, mass, power, and operating time for each run, to give equivalent saturation activities, A_s and A_{se} (cpm/g watt). At each foil location, the cadmium ratio ($R_{cd} = A_s/A_{se}$) was determined. The data are shown in Tables 5-1, 5-2, and 5-3 for the three foil materials.

The cadmium ratios were averaged for each region having similar properties, e.g., inside the core, at the thermal shield face, at the pressure vessel face, and above the upper support plate. For each average cadmium ratio, the epithermal parameters were calculated, as described in Sec. 3.3.1. The resulting cadmium ratios and epithermal parameters are listed in Table 5-4.

The three foil materials show good agreement for the parameter $r\sqrt{T_n/T_0}$ at each position. This represents the ratio between the epithermal flux per lethargy unit and the thermal flux. The consistent results from the three sets of foils are very satisfactory. The ratio is also a confirmation of the effective neutron temperature T_n , as calculated in Sec. 3.3.1.

The parameter $r s$ was used to obtain the effective cross section for total activation of bare foils. The factor, s , which is a normalized form of resonance integral,⁸ was corrected by the resonance self-shielding factor.

5.2 THERMAL FLUX

The bare-foil activities A_s were used to calculate the thermal flux ϕ_{th} for each point, using the activation formula as given in Sec. 3.3.1. The resulting thermal flux data from gold, manganese, and vanadium foils are given in Tables 5-1, 5-2, and 5-3, respectively. These results are also shown graphically and compared with analytical flux calculations in Sec. 8. The three sets of experimental measurements are quite consistent with each other and in satisfactory agreement with calculations.

5.3 EPITHERMAL FLUX

The cadmium-covered-foil activities A_{se} were used directly to obtain the epithermal flux $\phi(E) \times E$ per $\ln E$ energy interval, using the resonance integral cross section corrected for self-shielding, as explained in Sec. 3.3.2. The resulting epithermal flux data from the gold, manganese and vanadium foils are given in Tables 5-1, 5-2, and 5-3, respectively. These results are also shown graphically and compared with analytical flux calculations in Sec. 8. The agreement between the three sets of experimental results and the calculated flux is very good.

5.4 FAST FLUX

The activation of sulphur, aluminum, and depleted uranium foils gave the fast flux above each of four threshold energies, as described in Sec. 3.3.3. These results are given in Table 5-5 for sulphur foils, Table 5-6 for aluminum foils, and Table 5-7 for U-238 foils. They are also shown graphically and compared with calculations in Sec. 8.

The fast flux at the four energy thresholds are consistent with one another and with the calculations. The statistical accuracy of the aluminum- and uranium-foil data is poor, particularly at the low-flux locations. These results are useful mainly to confirm the sulphur data, and the shape of the attenuation curves. For determination of the integrated fast flux the sulphur-foil measurements were used, since these were considerably more accurate.

5.5 EXPERIMENTAL ACCURACY

The sources of error associated with this type of flux measurement have been discussed in some detail in APAE-95.¹ In general, conditions were quite similar in the present experiments, and the estimates of possible errors are about the same as before. All errors are reported as the estimated or calculated standard deviations.

The experimental techniques used in foil counting permitted two main sources of error, (1) counter-efficiency calibration errors and (2) statistical counting errors. Other errors due to the experimental techniques were estimated to be of the order of 1 percent and negligible by comparison. The counter calibrations were repeated at the new discriminator bias settings used in the present experiment, and were checked against calculated efficiencies, as discussed in Sec. 3.4. The standard deviation associated with these counter efficiencies was estimated to be 8 percent.

The counting statistics were the main limiting factor, as discussed in Sec. 3.4. The locations of most interest (at the pressure-vessel surface and the sample locations above the core) were unfortunately the positions of lowest flux, with foil activities close to background levels. In the case of sulphur foils, the standard deviations were generally less than 10 percent, and none were worse than 30 percent. For the aluminum foils, and uranium covered with boron or cadmium, the errors were larger, and the data were generally discarded when the standard deviations were greater than 50 percent. For this reason, the integrated fast flux was based primarily on the sulphur-foil data.

The statistical errors associated with the gold standard foils used for power-level measurements were much smaller, with a standard deviation of 3 percent or less. The uncertainty associated with the standard-foil activity relative to the core average power, as determined in Sec. 4, has an estimated standard deviation of 5 percent. The calculated relationship between power and flux has a standard deviation of about 3 percent.

The nuclear data used in all calculations were taken from the best and latest reference sources available. In most cases, the uncertainties are of the order of from 1 to 5 percent, but for corrected resonance integrals and threshold cross sections may range up to 10 or 15 percent.

The position of foils and reactor mockup materials was controlled within an accuracy of 0.1 in., which gives a flux uncertainty of 7 percent at some locations. The difference between flux measurements in the water gap at $R = 22.5$ in., and calculations under the vessel cladding at $R = 23.0$ in., was corrected as described in Secs. 8.4 and 8.5.

Since the formulae used for flux determination included all these parameters as factors, the over-all standard deviation is determined as:

$$S^2 (\%) = \sum_i S_i^2 (\%).$$

From the above estimates, the over-all standard deviation is about 19 percent in general, and may be as much as 36 percent in the worst cases. The 95-percent confidence limits are about twice the standard deviation.

Wherever possible, any sources of systematic errors were removed. It is not possible to estimate the magnitude of any such errors which may remain in either the measurements or calculations. For example, possible impurities in the foils which contribute additional activities, might account for higher flux measurements in the outer regions where the activity was close to background. The neglect of other correction factors could also be a cause of systematic errors.

TABLE 5-1

GOLD FOIL ACTIVITY AND FLUX

location*		activity, 10^5 cpm/g-w		R_{cd}	flux, 10^5 n/cm ² -sec	
<u>R, in.</u>	<u>Z, in.</u>	<u>A_s</u>	<u>A_{se}</u>		<u>ϕ_{th}</u>	<u>$\phi(E) \times E$</u>
1.54 (45a)	-10.88	29.8	11.2	2.66	39.2	4.61
	- 9.88	23.4	12.9	1.81	20.2	5.21
	- 7.38	24.9	15.4	1.62	21.5	6.23
	- 4.88	23.2	13.2	1.76	20.0	5.32
	- 0.88	14.5	9.27	1.56	12.5	3.74
	6.12	7.8	3.62	2.16	6.75	1.46
	10.12	2.26	1.56	1.45	1.96	0.63
	-10.88	32.0	9.27	3.45	42.2	3.74
3.00 (45j)	- 9.88	20.3	12.7	1.59	17.6	5.13
	- 7.38	26.8	14.6	1.83	23.2	5.89
	- 4.88	25.7	14.6	1.76	22.2	5.89
	- 0.88	19.6	10.9	1.80	17.0	4.37
	6.12	9.3	5.97	1.56	8.04	2.41
	10.12	4.27	1.89	2.26	3.69	0.76
	-10.88	23.5	9.75	2.41	31.0	3.93
	- 9.88	20.2	10.85	1.86	17.5	4.44
4.34 (45r)	- 7.38	23.6	14.9	1.59	20.5	6.03
	- 4.88	22.7	11.3	2.01	19.7	4.56
	- 0.88	13.7	9.1	1.51	11.9	3.67
	6.12	6.3	4.72	1.34	5.46	1.91
	10.12	2.9	1.73	1.68	2.50	0.70
	-10.88	19.6	6.76	2.90	25.9	2.73
	- 9.88	17.4	7.86	2.21	15.0	3.18
	- 7.38	19.5	10.05	1.94	16.9	4.06
7.56 (47W)	- 4.88	18.6	8.95	2.08	16.1	3.61
	- 0.88	12.8	7.86	1.63	11.1	3.18
	6.12	7.55	4.08	1.85	6.53	1.65
	10.12	3.14	1.89	1.66	2.72	0.76
	-10.88	16.6	4.39	3.78	23.1	1.73
	- 9.88	10.8	5.81	1.86	9.4	2.35
	- 7.38	14.2	8.32	1.71	12.3	3.36
	- 4.88	14.7	9.26	1.59	12.8	3.74
8.82 (47C)	- 0.88	12.8	7.07	1.81	11.1	2.86
	6.12	8.7	4.56	1.90	7.52	1.84
	10.12	3.65	2.04	1.79	3.16	0.83

*R = radial distance from core centerline

Z = axial distance from core centerline

Table 5-1 - Gold Foil Activity and Flux (cont'd)

location*		activity, 10^5 cpm/g-w		R_{cd}	flux, 10^5 n/cm ² -sec	
R , in.	Z , in.	A_s	A_{se}		ϕ_{th}	$\phi(E) \times E$
10.08	-10.88	14.7	3.62	4.07	20.5	1.46
(47E)	- 9.88	13.5	4.25	3.17	17.7	1.72
	- 7.38	17.9	5.96	2.99	23.6	2.41
	- 4.88	20.1	6.91	2.91	26.7	2.80
	- 0.88	19.0	6.60	2.88	25.0	1.73
	6.12	9.95	3.14	3.17	13.1	1.27
	10.12	4.92	1.58	3.11	6.5	0.63
11.0	- 7.4	37.66	4.66	8.1	59.0	1.88
12.0	- 7.4	35.48	2.89	12.3	59.3	1.16
14.0	- 7.4	14.70	0.85	17.3	24.9	0.35
16.0	- 7.4	5.43	0.29	18.5	9.22	0.12
17.7	-10.9	0.54	0.134	4.0	0.84	0.054
(TS)	- 7.4	1.05	0.170	6.2	1.63	0.069
	- 4.9	1.30	0.186	7.0	2.03	0.075
	- 0.9	1.24	0.185	6.7	1.94	0.075
	6.1	0.78	0.121	6.45	1.22	0.049
	10.1	0.57	0.067	8.5	0.89	0.027
20.0	- 7.4	0.137	0.088	1.56	0.18	0.035
22.5	-10.9	0.090	0.032	2.8	0.120	0.013
(PV)	- 7.4	0.125	0.035	3.6	0.167	0.014
	- 4.9	0.135	0.046	3.0	0.180	0.018
	- 0.9	0.128	0.052	2.5	0.171	0.021
	6.1	0.066	0.041	1.6	0.087	0.017
	10.1	0.047	0.013	3.6	0.063	0.005
25.5	-10.9	0.014	0.002	8.4	0.019	0.001
	- 7.4	0.022	0.005	4.5	0.029	0.002
	- 4.9	0.030	0.011	2.7	0.040	0.005
	- 0.9	0.033	0.019	1.7	0.044	0.008
	6.1	0.033	0.010	3.4	0.044	0.004
	10.1	0.023	0.008	3.0	0.030	0.003
11.0	20.2	0.045	0.011	3.9	0.060	0.004
(#5)	22.4	0.034	0.012	2.3	0.046	0.005
	24.5	0.026	0.006	2.6	0.035	0.002
(#6)	22.4	0.033	0.011	2.1	0.044	0.004
(#7)	22.4	0.033	0.011	3.0	0.044	0.004
(#8)	22.4	0.030	0.007	2.0	0.041	0.003
15.0	16.1	0.111	0.053	2.1	0.147	0.022
(#4)	17.4	0.098	0.031	3.1	0.132	0.013
	18.3	0.047	0.011	4.3	0.063	0.005
	19.6	0.033	0.007	4.6	0.044	0.003
	20.5	0.031	0.007	4.4	0.041	0.003
	21.7	0.026			0.034	

*R = radial distance from core centerline

Z = axial distance from core centerline

TABLE 5-2
MANGANESE FOIL ACTIVITY AND FLUX

location*		activity, 10^5 cpm/g-w		R_{cd}	flux, 10^5 n/cm ² -sec	
$R, \text{ in.}$	$Z, \text{ in.}$	A_s	A_{se}		ϕ_{th}	$\phi(E) \times E$
1.54 (45a)	-10.88	11.04	1.251	8.82	35.8	4.62
	- 9.88	6.79	1.595	4.26	23.5	5.88
	- 7.38	7.13	1.848	3.86	24.6	6.15
	- 0.88	3.61	1.238	2.91	12.5	4.56
	6.12	1.84	0.616	2.98	6.37	2.28
	10.12	0.79	0.227	3.47	2.73	0.83
3.00 (45j)	-10.88	12.80	1.246	10.26	41.4	4.74
	- 9.88	6.10	1.469	4.16	21.1	5.40
	- 7.38	6.90	2.008	3.44	23.9	7.41
	- 4.88	6.88	1.784	3.86	23.8	6.59
	- 0.88	5.18	1.530	3.39	18.0	5.64
	6.12	2.72	0.784	3.47	9.4	2.89
4.34 (45r)	10.12	1.21	0.291	4.17	4.2	1.07
	-10.88	9.42	1.077	8.75	30.6	3.97
	- 9.88	6.22	1.360	4.58	21.5	5.01
	- 7.38	6.61	1.622	4.08	22.9	5.97
	- 4.88	6.71	1.453	4.62	23.3	5.36
	- 0.88	3.25	1.127	2.89	11.3	4.16
7.56 (47W)	6.12	1.61	0.593	2.77	5.58	2.15
	10.12	0.79	0.242	3.28	2.74	0.89
	-10.88	7.61	0.728	10.46	24.6	2.69
	- 9.88	5.22	0.890	5.88	18.1	3.27
	- 7.38	5.40	1.133	4.76	18.7	4.18
	- 4.88	5.54	1.190	4.66	19.2	4.40
8.82 (47C)	- 0.88	3.45	1.047	3.30	11.9	3.87
	6.12	2.03	0.533	3.81	7.05	1.97
	10.12	0.94	0.240	3.91	3.25	0.88
	-10.88	6.72	0.535	12.58	21.8	1.97
	- 9.88	3.00	0.701	4.28	10.4	2.58
	- 7.38	3.98	0.968	4.10	13.7	3.58
10.08 (47E)	- 4.88	4.09	1.120	3.66	14.2	4.12
	- 0.88	3.96	1.134	3.49	13.7	4.17
	6.12	2.07	0.580	3.56	7.1	2.14
	10.12	1.07	0.235	4.52	3.7	0.87
	-10.88	6.00	0.370	16.21	20.4	1.42
	- 9.88	4.71	0.510	9.24	17.6	1.88
	- 7.38	6.20	0.762	8.04	22.9	2.86
	- 4.88	6.75	0.973	7.09	25.8	3.65
	- 0.88	6.08	0.902	6.74	22.7	3.33
	6.12	3.63	0.456	7.96	13.6	1.72
	10.12	1.27	0.198	6.37	4.7	0.75

* R = radial distance from core centerline
Z = axial distance from core centerline

Table 5-2 - Manganese Foil Activity and Flux (cont'd)

location*		activity, 10^5 cpm/g-w		R_{cd}	flux, 10^5 n/cm ² -sec	
$R, \text{in.}$	$Z, \text{in.}$	A_s	A_{se}		ϕ_{th}	$\phi(E) \times E$
11.0	- 7.4	17.36	0.580	29.9	68.2	2.09
12.0	- 7.4	17.26	0.327	52.8	68.8	1.16
14.0	- 7.4	8.55	0.096	89.	34.1	0.354
16.0	- 7.4	2.97	0.032	93.	11.8	0.122
17.7	-10.9	0.208	0.0127	16.4	0.81	0.0469
(TS)	- 7.4	0.424	0.0177	26.8	1.85	0.0651
	- 4.9	0.499	0.0223	22.5	1.95	0.0811
	- 0.9	0.527	0.0245	21.5	2.06	0.0904
	6.1	0.307	0.0144	21.3	1.20	0.0532
	10.1	0.243	0.0067	35.9	0.95	0.0249
			0.0093			0.0344
20.0	- 7.4		0.0034	12.3	0.157	0.0124
22.5	-10.9	0.041	0.0052	11.3	0.221	0.0191
(PV)	- 7.4	0.058	0.0055	11.4	0.242	0.0205
	- 4.9	0.064	0.0076	6.8	0.196	0.0279
	- 0.9	0.052	0.0037	7.5	0.104	0.0135
	6.1	0.027	0.0018	9.8	0.068	0.0068
	10.1	0.018	0.0012	3.7	0.016	0.0046
25.5	-10.9	0.0046	0.0013	3.7	0.017	0.0048
	- 7.4	0.0048	0.0013	3.8	0.017	0.0047
	- 4.9	0.0052	0.0011			0.0042
	- 0.9		0.0007			0.0028
	6.1		0.0018	7.7	0.053	0.0066
	10.1		0.0014	8.9	0.049	0.0052
11.0	20.2	0.0137	0.0009	11.9	0.038	0.0032
(#5)	22.4	0.0126	0.0014	19.0	0.042	0.0053
	24.5	0.0103	0.0013	8.7	0.045	0.0049
(#6)	22.4	0.0107	0.0013	8.4	0.041	0.0047
(#7)	22.4	0.0116	0.0055	8.6	0.181	0.0199
(#8)	22.4	0.0106	0.0023	19.3	0.170	0.0084
15.0	16.1	0.0470	0.0010	16.9	0.064	0.0036
(#4)	17.4	0.0440	0.0006	14.8	0.036	0.0023
	18.3	0.0166	0.0005	17.8	0.038	0.0020
	19.6	0.0092	0.0006	7.9	0.037	0.0016
	20.5	0.0098				
	21.7	0.0096				

* R = radial distance from centerline of core

Z = axial distance from core centerline

TABLE 5-3

VANADIUM FOIL ACTIVITY AND FLUX

location*		activity, 10^5 cpm/g-w		R_{cd}	flux, 10^5 n/cm ² -sec	
R , in.	Z , in.	A_s	A_{se}		ϕ_{th}	$\phi(E) \times E$
1.54	- 7.38	2.26	0.473	4.8	29.9	7.52
3.00	- 7.38	2.27	0.420	5.3	29.9	6.67
4.34	- 7.38	1.79	0.347	5.1	23.6	5.52
7.56	- 7.38	1.70	0.280	6.1	22.4	4.45
8.82	- 7.38	1.23	0.244	5.1	16.2	3.88
10.08	- 7.38	1.90	0.210	9.0	24.1	3.33
11.0	- 7.4	5.90	0.104	56.	72.6	1.65
12.0	- 7.4	5.94	0.0663	89.	73.1	1.05
14.0	- 7.4	3.10	0.0221	140.	37.1	0.35
16.0	- 7.4	1.02	0.0077	133.	12.5	0.122
17.7	-10.9	0.0645			0.81	
(TS)	- 7.4	0.1120			1.42	
	- 3.9	0.1727			2.19	
	3.1	0.1341			1.71	
	10.1	0.0745			0.95	
	- 7.4	0.0170			0.216	
20.0	- 7.4	0.0170			0.115	
22.5	-10.9	0.0088			0.185	
(PV)	- 7.4	0.0143			0.204	
	- 3.9	0.0153			0.128	
	- 0.9	0.0181			0.088	
	3.1	0.0072			0.123	
	6.1	0.0108			0.091	
	10.1	0.0077				

*R = radial distance from centerline of core

Z = axial distance from core centerline

TABLE 5-4

CADMIUM RATIOS AND EPITHERMAL PARAMETERS

foil material	location*		R_{cd}	$r\sqrt{T_n/T_o}$	$r/K\sqrt{T_n/T_o}$	rs
	$R, \text{ in.}$	$Z, \text{ in.}$				
gold	core	avg.	1.70	0.282	0.136	1.019
	11	-6.4	7.2	0.038	0.018	0.135
	12	-6.4	12.3	0.019	0.009	0.068
	14	-6.4	17.1	0.014	0.007	0.051
	17.7	-6.4	6.0	0.045	0.022	0.162
	22.7	-6.4	3.5	0.088	0.043	0.316
	11	22.4	3.5	0.088	0.043	0.316
	15	17.4	4.5	0.031	0.031	0.229
manganese	core	avg.	3.87	0.284	0.137	0.154
	11	-6.4	29.9	0.029	0.014	0.016
	12	-6.4	58.6	0.018	0.008	0.009
	14	-6.4	89.	0.011	0.005	0.006
	17.7	-6.4	22.5	0.049	0.023	0.026
	22.7	-6.4	11.3	0.100	0.048	0.054
	11	22.4	8.7	0.137	0.066	0.074
	15	17.4	16.9	0.060	0.029	0.033
vanadium	core	avg.	5.1	0.279	0.135	0.077
	11	-6.4	56.7	0.024	0.011	0.006
	12	-6.4	89.6	0.015	0.007	0.004
	16	-6.4	13.3	0.102	0.049	0.027

*R = radial distance from centerline of core

Z = axial distance from core centerline

TABLE 5-5
SULPHUR-FOIL ACTIVITY AND FLUX

location*		activity, 10^3 cpm/g-w		flux >3.0 Mev	
R, in.	Z, in.			10^5 n/cm ² -sec	
1.54	- 7.38	18.18		17.6	
3.00	-10.88	9.32		9.0	
(45j)	- 9.88	16.40		15.8	
	- 7.38	22.60		21.8	
	- 4.88	23.60		22.8	
	- 0.88	16.68		16.1	
	6.1	9.31		9.0	
	10.1	3.89		3.8	
4.34	- 7.38	16.73		16.2	
7.56	-10.88	5.58		5.4	
(47j)	- 9.88	12.27		11.8	
	- 7.38	14.67		14.0	
	- 4.88	13.80		13.4	
	- 0.88	12.77		12.3	
	6.1	6.67		6.5	
	10.1	3.47		3.4	
11.0	- 7.4	5.50	4.28	5.31	3.93
12.0	- 7.4	2.86	2.47	2.77	2.27
14.0	- 7.4	0.92	0.97	0.89	0.90
16.0	- 7.4		0.39		0.36
17.7	-10.9		0.11		0.10
(TS)	- 7.4	0.200		0.184	
	- 4.9	0.208		0.191	
	- 2.9	0.232		0.213	
	- 0.9	0.224		0.206	
	2.1	0.210		0.193	
	6.1	0.164		0.151	
	10.1	0.095		0.087	
20.0	- 7.4	0.0596		0.055	
	- 4.9	0.0697		0.064	
	- 0.9	0.0672		0.062	
22.5	-10.9	0.0138		0.0127	
(PV)	- 7.4	0.0188		0.0173	
	- 4.9	0.0194		0.0179	
	- 2.9	0.0181		0.0167	
	- 0.9	0.0203		0.0186	
	2.1	0.0159		0.0146	
	6.1	0.0126		0.0116	
	10.1	0.0069		0.0063	
11.0	20.2	0.0158		0.0145	

* R = radial distance from core centerline
Z = axial distance from core centerline

Table 5-5 - Sulphur-Foil Activity and Flux (cont'd)

location*		activity, 10^3 cpm/g-w	flux >3.0 Mev
R, in.	Z, in.		10^5 n/cm ² -sec
(#5)	22.4	0.0060	0.0056
	24.5	0.0026	0.0024
(#6)	22.4	0.0044	0.0040
(#7)	22.4	0.0059	0.0054
(#8)	22.4	0.0069	0.0063
15.0	16.1	0.0387	0.0356
(#4)	17.4	0.0267	0.0246
	18.3	0.0153	0.0141
	19.6	0.0102	0.0103
	20.5	0.0074	0.0068
	21.7	0.0042	0.0039

* R = radial distance from core centerline

Z = axial distance from core centerline

TABLE 5-6
ALUMINUM FOIL ACTIVITY AND FLUX

location*		activity, 10^3 cpm/g-w		flux, 10^3 n/cm ² -sec	
R, in.	Z, in.	Mg-27	Na-24	>5.3 Mev	>8.1 Mev
1.54	-7.4	1.38	0.514	397.	58.3
3.00	-7.4	1.52	0.589	440.	65.0
4.34	-7.4	1.36	0.422	391.	47.4
7.56	-7.4	1.06	0.393	307.	44.3
8.82	-7.4	0.86	0.369	249.	42.0
10.08	-7.4	0.66	0.280	191.	32.0
11.0	-7.4	0.256	0.130	73.9	14.65
12.0	-7.4	0.155	0.070	44.7	7.90
14.0	-7.4	0.081	0.0396	23.4	4.50
16.0	-7.4	0.024	0.0079	7.0	0.89
17.7	-7.4	0.021		6.1	
(TS)	-4.9	0.0167		4.81	
	-2.9	0.0171	0.0152	4.94	1.71
	-0.9	0.0086		2.49	
20.0	-7.4	0.0032	0.0025	0.91	0.28
	-4.9	0.0034		1.01	
22.5	-2.9	0.0011	0.0009	0.33	0.10

* R = radial distance from core centerline
Z = axial distance from core centerline

TABLE 5-7
URANIUM-238 FOIL ACTIVITY AND FLUX

location*		activity, 10^3 cpm/g-w			flux, 10^3 n/cm ² -sec >1.57 Mev
R, in.	Z, in.	bare	Cd-covered	B-covered	
1.54	- 7.4	26.1			
3.00	- 7.4	26.5			
4.34	- 7.4	24.2			
7.56	- 7.4	18.9			
8.82	- 7.4	14.9			
10.08	- 7.4	19.4			
11.0	- 7.4	43.4	2.86	1.84	810.
12.0	- 7.4	42.5	1.55	1.01	445.
14.0	- 7.4	20.3	0.514	0.33**	145.
16.0	- 7.4	7.7		0.144	63.
17.7	- 7.4	1.8	0.072	0.046**	21.
20.0	- 7.4		0.027	0.017**	7.5
22.5	- 7.4	0.20	0.015	0.009**	4.0
11.0	22.4	0.09		0.002	0.88
15.0	17.4			0.006	2.6

*R = radial distance from core centerline

Z = axial distance from core centerline

**These equivalent boron covered foil activities were determined from the actual cadmium covered foil activities in these locations.

6. CALCULATIONAL MODELS

Detailed knowledge of the spatial and energy variation of the fast neutron flux is necessary for the evaluation of radiation damage. In this section, the spectral composition and magnitude of the fast neutron flux is determined by using two codes, PDQ(r,z)-2¹⁶ and P1MG-2¹⁷. PDQ is a few-group two-dimensional code, which gives fluxes at any desired point within the core and non-core regions. P1MG-2 is a multigroup one-dimensional code, which gives the energy dependent flux within the core and non-core regions of the reactor.

Initial input data required for the above programs were obtained from P₃¹⁸ for the thermal group and from MUFT-III¹⁹ for the fast group. The results of these calculations are presented and compared to the experiment in Sections 7.5 and 8.6.

6.1 P1MG-2

The P1MG code solves for the spatially dependent multigroup flux, taking into account the slowing down of neutrons from elastic and inelastic scattering, the removal of neutrons from epithermal capture and fission resonances, and the regeneration of fast neutrons from fission in the epithermal groups or in the one thermal group. The code requires, as input, the reactor geometry, number of points and regions, composition, group-dependent boundary conditions, transverse buckling, thermal group parameters, and mesh sizes. It contains a built-in library of the epithermal nuclear properties of all the isotopes in each composition. The code will produce as output multigroup fluxes, currents, and isotopic slowing-down densities, in addition to pointwise and regionwise few-group microscopic cross sections.

The P1MG code was not designed to treat transport effects. Errors, resulting from the limitation of a P₁ approximation, occur in the P1MG equations as they do in most forms of the few-group diffusion equations.

The basic P1MG equations, using the Greuling-Goertzel approximation, may be written in the following form for each lethargy group:

$$D \cdot \bar{J} + (\Sigma_a + \Sigma_{in}) \phi + \sum_i \frac{\delta q_i}{\delta \mu} = S \quad (1)$$

$$\gamma_i \frac{\delta q_i}{\delta \mu} + q_i = (\xi \Sigma_s)_i \phi \quad (2)$$

$$1/3 \nabla \phi + \Sigma_{tr} \bar{J} + \sum_i \frac{\delta \bar{P}_i}{\delta \mu} = 0 \quad (3)$$

$$\sum_i \frac{\delta \bar{P}_i}{\delta \mu} + \bar{P}_i = (\eta \Sigma_s)_i \bar{J} \quad (4)$$

where,

$\bar{J}$ and ϕ are the current and scalar flux.

i is an index denoting various isotopes.

q_i is the isotropic slowing-down density resulting from the i^{th} isotope.

$\bar{P}_i$ is the anisotropic slowing-down density resulting from the i^{th} isotope.

S is the fission source S_F plus the source due to the inelastic scattering into lethargy group m .

γ_i , ξ_i , ζ_i , and η_i are parameters which are deduced from the slowing-down properties of each isotope.

Σ_a , Σ_s , Σ_{in} , and Σ_{tr} are the macroscopic absorption, scattering, inelastic scattering, and transport cross sections, respectively.

The basic limitations of the P1MG code are as follows:

(1) The code does not treat transport effects. Errors resulting from a P_1 approximation will occur in the output.

(2) The treatment of resonances implies that slowing-down predominates over leakage. If the flux in a region is determined primarily by leakage, rather than by slowing-down densities, the P1MG resonance model will be inadequate.

(3) The accuracy of the model will be determined, to a large extent, by the accuracy of the data on the prepared library tapes.

The code partially compensates for transport effects by using anisotropic slowing-down parameters of the various isotopes.

Insofar as the input to the prepared library tapes consists of parameters for the 54-group breakdown, it is felt that this is the largest source of error in assessing the P1MG output. In many cases, the cross-sectional data for various energy groups have not been measured and, consequently, it is necessary to rely on nuclear models to calculate these.

6.2 PDQ-2

PDQ-2 is a two-dimensional reactor design code for the IBM-704 computer. It finds a discrete numerical approximation to the few group, time-independent, neutron-diffusion equations for a heterogeneous reactor in a two-dimensional bounded region. The solution is obtained over the region in the (r,z) plane with a zero flux condition applied along each outer boundary.

The few-group, time-independent, neutron-diffusion equations solved by PDQ-2 in the (r,z) or (x,y) geometry are:

$$\left\{ -\nabla \left[D_i(\bar{x}) \nabla \phi_i(\bar{x}) \right] + \sigma_i(\bar{x}) \phi_i(\bar{x}) = \frac{x_i \Psi(\bar{x})}{\lambda} + \sum_{i=1}^R (\bar{x}) \phi_{i-1}(\bar{x}) \right\}_{i=1}^k$$

where:

$$\sigma_i(\bar{x}) = \Sigma_i^a(\bar{x}) + \Sigma_i^R(\bar{x}) + B_z^2 D_i(\bar{x})$$

and

$$\lambda(\bar{x}) = \sum_{i=1}^K \left[\nu_i \Sigma_i^f(\bar{x}) \right] \phi_i(\bar{x})$$

in addition

$$\Sigma_o^R(\bar{x}) = \Sigma_k^R(\bar{x}) = 0 \text{ and } B_z^2 = 0 \text{ in the (r, z) geometry.}$$

The physical interpretations of the symbols are:

- D = the diffusion constant
- Σ^a = the absorption cross section
- Σ^R = the removal cross section
- B_z^2 = the transverse buckling
- x_i = the integral of the fission spectrum over the lethargy range represented by group i
- Ψ = the fission source
- λ = the eigenvalue
- ν_i = the average number of neutrons produced by a fission in group i
- Σ^f = the fission cross section
- ϕ = the neutron flux

The output of the code includes a complete edit of the input, a "picture" of the mesh with all regions and interfaces indicated, condensed results during the iterations to indicate the progress of the problem, the region integrals and averages of the flux and source, and the pointwise flux and source values.

The PDQ (r,z) code is designed for application to a configuration which is concentrically symmetrical about the core centerline. The core and regions above and below it are heterogeneous regions. It is thus necessary to homogenize these regions, in order to comply with the symmetric restriction. The associated thermal shield, pressure vessel, and primary shielding are concentric annuli, and therefore, conform to the symmetry restriction.

The maximum-to-average ratio of the fast flux in the axial direction, at the pressure vessel, is used to correct the results of P1MG, which assume an average level of the flux in the axial direction.

6.3 MUFT-III

The IBM-650 MUFT-III code is used to compute fast constants based on solving the Fourier transform of the slowing-down distribution from a point source. The equations are solved using any of the four approximations: P-1, B-1, P-1 Selengut-Goertzel, and B-1 Selengut-Goertzel. The P-1 Selengut-Goertzel approximation has been used here. Atomic concentrations and buckling are the only input required.

6.4 P₃ CODE

The IBM-650 P₃ code is used to calculate the thermal cross sections of the core, taking into account the flux depression data perpendicular and parallel to the fuel plates. The calculation is based on the one velocity P₃ approximation to the Boltzmann equation.

6.5 CODE UTILIZATION

P1MG is used to calculate group fluxes at every point in the r direction, normalized to a source of 1 neutron/cm²-sec, i.e., - per unit core height.

The conversion to absolute flux is based upon the formula:

$$\text{absolute flux} = \text{C.F.} \times (\text{normalized flux}) \text{ with } \text{C.F.} = \frac{\nu \delta P}{kh}$$

where C.F. = conversion factor

ν = average number of neutrons/fission = 2.46

δ = fissions/w-sec = 3.2×10^{10}

P = power in w/cm²

k = eigenvalue

h = core height

Thus $\nu_c \delta P A$ is the total number of neutrons produced per second for a power of PA watts, where ν_c is a fictitious value of ν to make the eigenvalue equal to unity for calculational purposes, and A is the cross-section area of the core.

The average number of neutrons/cm³-sec is a $\nu_c \delta PA/V$, where V is the core volume.

Now

$$k_{\text{eff}} = \frac{\nu}{\nu_c} \quad \text{and} \quad V = Ah, \text{ so C. F.} = \frac{\nu \cdot \delta \cdot P}{kh}$$

The PDQ(r,z) results have been used to derive an axial peak-to-average ratio of the fast flux as a function of position in the radial direction. This axial peak-to-average ratio as a function of radial position is used to modify the flux calculated by P1MG, which in turn is based on the average axial power level. Therefore utilization of the peak/average factor yields the maximum absolute fast flux at the pressure vessel.

The flux value derived in this manner is conservative, however, since the control rods will be withdrawn with increasing core history for reactivity compensation, thereby causing a relocation of the axial power peak. Therefore, the belt-line of maximum exposure at the start of life will not receive the maximum incident flux throughout the core life.

6.6 INPUT PARAMETERS FOR THE SM-1A AT 68 F AND 440 F AT START OF LIFE

The basic input to the few-group codes consist of:

- (1) geometrical data describing the configuration, and
- (2) material and energy group-dependent nuclear parameters for each region.

Using two groups requires two sets of nuclear parameters for each composition. The parameters corresponding to the epithermal flux are: D_i , Σ_i^a , Σ_i^R and $\nu\Sigma_i^f$ for composition i, the same set of parameters is needed for the thermal group. For the multigroup P1MG-2 code, it is not necessary to specify any epithermal parameters, since these are calculated by the code itself.

The fast group parameters for PDQ-2 were obtained from MUFT-III. The thermal group parameters were calculated using P_3 theory. Table 6-1 lists the SM-1A mockup (68 F) thermal and epithermal group parameters for the two group PDQ-2. The thermal group parameters used as input for the multigroup P1MG-2 code are also shown and are identical to those of the two group code.

Table 6-2 lists the same type of input for the SM-1A reactor at operating temperature of 440 F at startup. The regionwise number densities used for P1MG for the SM-1A mockup and SM-1A reactor are shown in Table 6-3.

The (r, z) geometry of the SM-1A core and vessel is illustrated in Figs. 1 and 2. Radial dimensions from the core centerline through the pressure vessel, as used in the calculational models are given in Table 6-4. It should be noted that the inner radius of the pressure vessel is 58.44 cm or 23.0 in., which is at the surface of the carbon steel, beneath the 0.25 in. stainless-steel cladding. In addition, the core equivalent radius is 28.20 cm or 11.1 in., whereas the actual flat core surface of element 47 is 10.28 in. from the center. The effect of these differences between calculational and experimental models is discussed in Section 8.

TABLE 6-1

INPUT PARAMETERS FOR THE SM-1A MOCKUP
68 F, 0 Mw-yr

composition	epithermal group parameters				thermal group parameters		
	$D, \text{ cm}$	$\Sigma_a, \text{ cm}^{-1}$	$\Sigma_R, \text{ cm}^{-1}$	$\nu\Sigma_F, \text{ cm}^{-1}$	$D, \text{ cm}$	$\Sigma_a, \text{ cm}^{-1}$	$\nu\Sigma_F, \text{ cm}^{-1}$
core (control absorbers present) . .	1.317022	0.010241	0.030595	0.013394	0.170449	0.355223	0.416732
core (no control absorbers present) .	1.317022	0.010241	0.030595	0.013394	0.170449	0.274830	0.416732
s.s. + H ₂ O (below core)	1.292673	0.004667	0.028371	0	0.223082	0.081416	0
s.s. + H ₂ O (above core)	1.465499	0.001909	0.041076	0	0.188207	0.037148	0
water reflector	1.591717	0.000612	0.046895	0	0.172303	0.016960	0
first thermal shield	1.314316	0.008086	0	0	0.311170	0.193228	0
water gap	1.591717	0.000612	0.046895	0	0.172303	0.016960	0
second thermal shield	1.314316	0.008086	0	0	0.311170	0.193228	0
inlet passage	1.591717	0.000612	0.046895	0	0.172303	0.016960	0
pressure vessel	1.314316	0.008086	0	0	0.311170	0.193228	0

TABLE 6-2

INPUT PARAMETERS FOR THE SM-1A MOCKUP
440 F, 0 Mw-yr

composition	epithermal group parameters				thermal group parameters		
	D, cm	Σ_a, cm^{-1}	Σ_R, cm^{-1}	$\nu\Sigma_F, \text{cm}^{-1}$	D, cm	Σ_a, cm^{-1}	$\nu\Sigma_F, \text{cm}^{-1}$
core (control absorbers present) . .	1.49034	0.009363	0.026102	0.012185	0.247063	0.294619	0.324239
core (no control absorbers present) .	1.49034	0.009363	0.026102	0.012185	0.247063	0.214298	0.324239
s.s. + H ₂ O (below core)	1.25498	0.004385	0.024005	0	0.292292	0.061850	0
s.s. + H ₂ O (above core)	1.33333	0.003518	0.027432	0	0.274103	0.026936	0
water reflector	1.89484	0.000459	0.039381	0	0.265809	0.011014	0
first thermal shield	1.31446	0.008132	0	0	0.338232	0.150037	0
water gap	1.89484	0.000459	0.039381	0	0.265809	0.011014	0
second thermal shield	1.31446	0.008132	0	0	0.338232	0.150037	0
inlet passage	1.89484	0.000459	0.039381	0	0.265809	0.011014	0
pressure vessel	1.31446	0.008132	0	0	0.338232	0.150037	0

TABLE 6-3

REGIONWISE NUMBER DENSITIES OF THE SM-1A FOR P1MG-2 CALCULATIONS

 $\times 10^{-24}$ atoms/cm³SM-1A Mockup
68 F

<u>element</u>	<u>region 1, core</u>	<u>region 2, reflector</u>	<u>region 3, thermal shield</u>	<u>region 4, water</u>	<u>region 5, thermal shield</u>	<u>region 6, water</u>	<u>region 7, pressure vessel</u>
H	0.053141	0.06673	-	0.06673	-	0.06673	-
O-16	0.027492	0.033365	-	0.033365	-	0.033365	-
Fe	0.010352	-	0.05942	-	0.05942	-	0.05942
Ni	0.001277	-	0.00730	-	0.00730	-	0.00730
Cr	0.002969	-	0.01704	-	0.01704	-	0.01704
U-235	0.00042886	-	-	-	-	-	-
B-10	0.00000706	-	-	-	-	-	-

SM-1A Reference
440 F

<u>element</u>	<u>region 1, core</u>	<u>region 2, reflector</u>	<u>region 3, thermal shield</u>	<u>region 4, water</u>	<u>region 5, thermal shield</u>	<u>region 6, water</u>	<u>region 7, pressure vessel</u>
H	0.044442	0.055807	-	0.055807	-	0.055807	-
O-16	0.022992	0.027904	-	0.027904	-	0.027904	-
Fe	0.010352	-	0.059420	-	0.059420	-	0.059420
Ni	0.001277	-	0.007300	-	0.007300	-	0.007300
Cr	0.002969	-	0.017040	-	0.017040	-	0.017040
U-235	0.00042886	-	-	-	-	-	-
B-10	0.00000706	-	-	-	-	-	-

TABLE 6-4

RADIAL DESCRIPTION OF SM-1A REACTOR

<u>region</u>	<u>material</u>	<u>outer radius, cm</u>	<u>thickness, cm</u>
core	-	28.20	28.20
reflector	water	45.70	16.90
thermal shield	steel	50.80	5.10
coolant	water	52.07	1.27
thermal shield	steel	57.17	5.10
coolant	water	58.44	1.27
pressure vessel	steel	66.06	7.62

7. RESULTS OF CALCULATIONS

Using the calculational models described in Sec. 6, energy and spatial fast neutron flux distributions have been calculated for the SM-1A at 68 F and 440 F, at initial startup. Fast flux distributions in an axial plane are shown at various radial distances from the core centerline to the inner surface of the pressure vessel for the SM-1A at 68 F.

7.1 SM-1A MOCKUP (68 F)

Figure 20 shows the axial distributions of fast neutrons obtained from PDQ (r, z) - 2 at the core centerline, the core reflector interface, the inner surfaces of the thermal shields, and the inner surface of the pressure vessel.

Figure 21 shows the axial peak-to-average fast flux as a function of position in the radial direction, as calculated by PDQ (r, z) - 2. This illustrates the axial peak flattening as a function of radial distance from the core.

The energy dependent fast neutron flux (in arbitrary units) at various positions in the reactor is shown in Fig. 22, as calculated by PIMG-2.

The fast flux of neutrons with energies greater than 1 Mev was calculated to be 3.12×10^3 nv/w at the peak axial location, and at the inner surface of the pressure vessel (R = 23 in.).

7.2 SM-1A REACTOR AT 440 F

A comparison of the fast neutron spectrum at the core centerline and inner surface of the pressure vessel of the SM-1A at operating temperature of 440 F is shown in Fig. 23.

Unlike the SM-1 at 68 F and 440 F, the spectrum "softens" towards the pressure vessel, since in the SM-1A there are two thermal shields with a total thickness of 10.2 cm. The removal of fast neutrons with energies greater than 1 Mev is higher in steel than for those below 1 Mev. Consequently, the fast neutron spectrum becomes "softer."

Calculations of integrated fast neutron fluxes at points of interest in the SM-1A (440 F, 0 Mw-yr) are given in Table 7-1. The values reported in this table were obtained from the codes described in Sec. 6. No correction factors, such as measured to calculated ratios or probable errors, are included as was done in previous reports.^{1,2}

TABLE 7-1

SM-1A VESSEL WALL FLUX (440 F)

<u>location</u>	<u>nv/w</u>	<u>nvt/Mw-yr</u>	<u>nvt (20-yr 60% L.F.)</u>
A. <u>($E_n > 1$ Mev)</u>			
pressure vessel*	7.16×10^3	2.26×10^{17}	5.42×10^{19}
support ring	1.15×10^3	3.62×10^{16}	8.70×10^{18}
nozzle weld	7.16×10^2	2.26×10^{16}	5.42×10^{18}
B. <u>($E_n = 0.1$ to 1.0 Mev)</u>			
pressure vessel*	1.77×10^4	5.58×10^{17}	1.34×10^{20}
support ring	2.84×10^3	8.97×10^{16}	2.14×10^{19}
nozzle weld	1.77×10^3	5.58×10^{16}	1.34×10^{19}

*Value in axial plane of maximum exposure.

8. EVALUATION AND COMPARISON OF RESULTS

8.1 THERMAL FLUX

The thermal flux distribution is shown radially in Fig. 24, and axially in Fig. 25. The points plotted resulted from measurements taken using gold, manganese, and vanadium foils, and the curves show the analytical predictions. The radial data points were all taken 3.5 in. above the core bottom, i.e., 7.4 in. below the core center. This was the elevation of peak flux in element 34, as determined from preliminary measurements. As seen from Fig. 25, the peak flux was at higher elevations as the radial distance increased, and was about 3 or 4 in. below core center at the pressure vessel surface. The axial flux peak was only slightly greater (about 10 percent) than the value obtained at the 3.5 in. elevation. The calculated curve is for the axial peak flux.

The axial distribution above the core support ring, at $R = 15$ in. and $R = 11$ in., shows the thermal flux is less than at the pressure-vessel peak. Plastic sample strips, 6, 7, and 8, at $R = 11$ in. and $Z = 22.4$ in., gave results similar to strip 5 (within experimental error).

The agreement between all sets of data is considered quite satisfactory, except at the very low flux positions where experimental accuracy was poor. The measured and calculated flux peak in the water reflector differ in position by about 1 in. This is due to the difference between the experimental core radius of 10.28 in. and the calculated equivalent cylindrical core radius of 11.1 in., as described in Sec. 6.6.

8.2 EPITHERMAL FLUX

The epithermal flux distribution is shown radially in Fig. 26, and axially in Fig. 27. The points plotted were taken for the three cadmium-covered foil materials: gold, manganese, and vanadium. The radial data were taken 7.4 in. below the core centerline, as before. The axial peak flux appears to be slightly higher (an inch or so in elevation) than for the thermal flux. The magnitude of the peak flux is again about 10 percent greater than the value at the -7.4 in. elevation.

The axial distribution above the core support ring, at $R = 15$ in. and 11 in., shows a lower epithermal flux value than the peak at the pressure vessel. Sample strips 6, 7, and 8, at $R = 11$ in. and $Z = 22.4$ in., gave results similar to strip 5 (within experimental error).

The curves in Figs. 26 and 27 show the calculated flux distribution, radially at the axial peak by PIMG-2 and axially by PDQ (r, z) - 2, normalized to experiments at the axial peak values.

The agreement between all sets of data is very good, and is much better than for the thermal or fast results.

8.3 FAST FLUX

The fast flux distribution is shown radially in Fig. 28, and axially in Fig. 29. The radial data points show results from uranium-238, sulphur, and aluminum foils, for each of the four threshold reactions. These were all taken at the -7.4 in. elevation, and the axial peak was again about 10 percent greater.

The axial distributions show only the sulphur-foil data, since the accuracy of these was much better than for the other foil materials. The flux at the sample strip-5 locations above the core support ring, at $R = 11$ in. and $Z = 20.2$ in., 22-4. in., 24-5 in., is less than at the pressure vessel, but for sample strip 4, at $R = 15$ in. and $Z = 16.1$ in. to 21.7 in., the two lower points show a flux somewhat greater than the pressure vessel peak.

The curves in Figs. 28 and 29 show the calculated flux distributions for each of the energy thresholds. The axial distributions are relative values only, and are normalized to agree with experiments at the peak of the curve.

The agreement between experiments and calculations is fairly satisfactory. The experimental accuracy is poor for the aluminum and uranium foils, as discussed in Sec. 5.4. There is a tendency for experimental results to be higher than calculations at the outer positions, particularly at the pressure vessel surface. The sulphur-foil measured flux at this location is higher than the calculated flux by a factor of about 1.68.

8.4 FAST FLUX ABOVE 1 Mev

Radiation monitors were installed in the SM-1A reactor at startup, and removed after 384 Mwd of irradiation at 435 F. They were located in the water gap at $R = 22.5$ in., between the second thermal shield and the pressure vessel cladding. The activation results were reported in NRL Memorandum Report 1402³ as the integrated fast flux above 1 Mev, assuming a fission neutron spectrum. These have been converted to the equivalent flux per watt, and are shown in Fig. 30.

Also shown in Fig. 30 are the sulphur-foil activations from the present experiments at 68 F, converted to the equivalent flux above 1 Mev, assuming a fission neutron spectrum. The data points shown are for foils located in the water gap at $R = 22.5$ in., near the pressure-vessel surface.

The two curves in Fig. 30 show the calculated axial flux above 1 Mev at the pressure-vessel surface, under the stainless-steel cladding ($R = 23.0$ in.), and for the cold (68 F) and hot (440 F) conditions. These calculations are not directly comparable to the experimental results shown in the same figure, because of the 0.5 in. difference in radial position. The slope of the radial flux distribution curve in Fig. 28 indicates that this 0.5-in. difference requires the experimental results to be corrected by a factor of 0.74.

As stated in Sec. 7.2, the fast neutron spectrum is flatter than a fission spectrum. This would have two effects on the experimental results. The effective threshold energies of

the detectors would be different from those determined and tabulated for a fission spectrum. Also, the ratio of the flux above 1 Mev to the flux above the threshold energy would be smaller than for a fission spectrum. The magnitude of the errors due to these spectrum effects could not be determined. This problem is acknowledged by most experimenters working in the field of fast-flux measurements and radiation damage studies.⁴

8.5. CONCLUSIONS

The flux measurements in the water gap near the pressure vessel ($R = 22.5$ in.) should be corrected by a factor of 0.74 to give the flux at the surface of the vessel under the cladding ($R = 23.0$ in.). When this is done, the ratio of experimental/calculated flux is as follows:

S-32, >3.0 Mev, 68 F, ratio 1.24
 S-32, >1.0 Mev, 68 F, ratio 1.14
 Fe-54, >1.0 Mev, 435 F, ratio 1.29

The conservative, best estimate of the integrated flux above 1 Mev at a given location may be taken as the calculated value, corrected by a factor of 1.30. This gives axial peak values at the pressure vessel surface as follows:

	<u>cold (68 F)</u>	<u>hot (440 F)</u>
nv per watt ($\text{n}/\text{cm}^2\text{-sec}$)	4.1×10^3	9.4×10^3
nvt per Mw-yr (n/cm^2)	1.3×10^{17}	2.9×10^{17}
nv (20 yr 60% L.F.) (n/cm^2)	3.1×10^{19}	7.1×10^{19}

The 95 percent confidence limits of ± 38 percent for the experimental results may be applied to these estimates.

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10. FIGURES

1. Mockup SM-1A Core and Shield - Plan View and Elevation Along Core Centerline
2. Shield Mockup above Core
3. Counter Calibration Curve
4. Typical Uranium Foil Locations on Fuel Plates (Element 15)
5. Fuel Plate Arrangement
6. Location of North Radial Planes
7. Normalized Average Power Per SM-1A Element
8. Average Axial Power Distribution - Elements 14, 15, 16
9. Average Axial Power Distribution - Elements 25, 26, 27
10. Average Axial Power Distribution - Elements 34, 35, 36, 37
11. Average Axial Power Distribution - Elements 45, 47
12. Average Axial Power Distribution - Elements 54, 56, 57
13. Average Axial Power Distribution - Elements 65, 66, 67
14. Average Axial Power Distribution - Elements 74, 75, 76
15. Average Axial Power Distribution - Control Rod Elements 24(3), 44(C), 46(4)
16. Average Axial Power Distribution - Control Rod Elements 55(A)
17. North-South Radial Power Distribution - Elements 44(C), 54, 64(1), 74
18. East-West Radial Power Distribution - Elements 44(C), 45, 46(4), 47
19. North-South Power Distribution, Outer Row Traverse - Elements 67, 57, 47, 37, 27
20. Calculated Axial Fast Neutron Flux Distributions at Various Locations in the SM-1A Mockup (Cold, Clean)
21. Calculated Axial Peak to Average Ratio of the Fast Neutron Flux Distribution in the SM-1A as a Function of Distance from the Core Centerline (SM-1A Mockup, Cold, Clean)
22. Energy Dependent Neutron Flux at Various Radial Positions in the SM-1A (Cold, Clean)
23. Energy Dependent Fast Neutron Flux at the Core Centerline and Inner Surface of the Pressure Vessel as Calculated by P1MG-2 (SM-1A 440 F)
24. Thermal Flux Distribution - Radial
25. Thermal Flux Distribution - Axial
26. Epithermal Flux Distribution - Radial
27. Epithermal Flux Distribution - Axial
28. Fast Flux Distribution - Radial
29. Fast Flux Distribution - Axial
30. Comparison of Fast Flux above 1 Mev at Pressure Vessel Surface

MOCKUP SM-1A CORE AND SHIELD - PLAN VIEW
AND ELEVATION ALONG CORE CENTERLINE

FIG. 1

SHIELD MOCKUP ABOVE CORE

FIG. 2

COUNTER CALIBRATION CURVE

FIG. 3

TYPICAL URANIUM FOIL LOCATIONS
ON FUEL PLATES (element 15)

FIG. 4

Stationary Element Fuel Plate Arrangement

Control Rod Element Fuel Plate Arrangement

FUEL-PLATE ARRANGEMENT

FIG. 5

LOCATION OF NORTH RADIAL PLANES (section of core through center of elements 14, 24, 34, and 44)

FIG. 6

element position number

.....
*.....
.....
.....

relative activity, avg (method A)
relative activity, avg (method B)
normalized power (method A)
normalized power (method B)

control rod number

	12 5347 0.811 0.861	13 5341 0.810 0.860	14 5403 0.819 0.870	15 5446 0.826 0.877	16 5457 0.827 0.879	
21 5602 0.849 0.902	22 7276 1.103 1.172	23 7126 1.081 1.147	24 1075 *1899 0.163 0.306	25 7504 1.138 1.208	26 7318 1.110 1.178	27 5675 0.861 0.914
31 5509 0.835 0.887	32 7266 1.102 1.170	33 6481 *7132 0.983 1.148	34 7504 1.138 1.208	35 8983 1.362 1.446	36 7762 1.177 1.250	37 5347 0.811 0.861
41 5467 0.829 0.880	42 1068 *1865 0.162 0.300	43 7408 1.123 1.193	44 1550 *2572 0.235 0.414	45 7408 1.123 1.193	46 1068 *1865 0.162 0.300	47 5467 0.829 0.880
51 5347 0.811 0.861	52 7762 1.177 1.250	53 8983 1.362 1.446	54 7504 1.138 1.208	55 6481 *7132 0.983 1.148	56 7266 1.102 1.170	57 5509 0.835 0.887
61 5675 0.861 0.914	62 7318 1.110 1.178	63 7504 1.138 1.208	64 1075 *1899 0.163 0.306	65 7126 1.081 1.147	66 7276 1.103 1.172	67 5602 0.849 0.902
	72 5457 0.827 0.879	73 5446 0.826 0.877	74 5403 0.819 0.870	75 5341 0.810 0.860	76 5347 0.811 0.861	

NORMALIZED AVERAGE POWER PER SM-1A ELEMENT

FIG. 7

AVERAGE AXIAL POWER DISTRIBUTION -
ELEMENTS 14, 15, 16

FIG. 8

AVERAGE AXIAL POWER DISTRIBUTION -
ELEMENTS 25, 26, 27

FIG. 9

AVERAGE AXIAL POWER DISTRIBUTION -
ELEMENTS 34, 35, 36, 37

AVERAGE AXIAL POWER DISTRIBUTION -
ELEMENTS 45, 47

FIG. 11

AVERAGE AXIAL POWER DISTRIBUTION -
ELEMENTS 54, 56, 57

FIG. 12

AVERAGE AXIAL POWER DISTRIBUTION -
ELEMENTS 65, 66, 67

FIG. 13

AVERAGE AXIAL POWER DISTRIBUTION -
ELEMENTS 74, 75, 76

FIG. 14

AVERAGE AXIAL POWER DISTRIBUTION - CONTROL
ROD ELEMENTS 24(3), 44(C), 46(4)

FIG. 15

AVERAGE AXIAL POWER DISTRIBUTION - CONTROL
ROD ELEMENTS 55(A)

FIG. 16

NORTH-SOUTH RADIAL POWER DISTRIBUTION -
ELEMENTS 44(C), 54, 64(1), 74

FIG. 17

EAST-WEST RADIAL POWER DISTRIBUTION -
ELEMENTS 44(C), 45, 46(4), 47

FIG. 18

NORTH-SOUTH POWER DISTRIBUTION, OUTER
ROW TRAVERSE - ELEMENTS 67, 57, 47, 37, 27

FIG. 19

CALCULATED AXIAL FAST NEUTRON FLUX DISTRIBUTIONS
AT VARIOUS LOCATIONS IN THE SM-1A MOCKUP (cold, clean)

FIG. 20

CALCULATED AXIAL PEAK TO AVERAGE RATIO OF THE FAST NEUTRON FLUX DISTRIBUTION IN THE SM-1A AS A FUNCTION OF DISTANCE FROM THE CORE CENTERLINE (SM-1A mockup, cold, clean)

FIG. 21

ENERGY DEPENDENT NEUTRON FLUX AT VARIOUS
RADIAL POSITIONS IN THE SM-1A (cold, clean)

ENERGY DEPENDENT FAST NEUTRON FLUX AT THE CORE CENTERLINE AND INNER SURFACE OF THE PRESSURE VESSEL AS CALCULATED BY PIMG-2 (SM-1A 440 F)

FIG. 23

THERMAL FLUX DISTRIBUTION - RADIAL

FIG. 24

THERMAL FLUX DISTRIBUTION - AXIAL

FIG. 25

EPITHERMAL FLUX DISTRIBUTION - RADIAL

FIG. 26

EPITHERMAL FLUX DISTRIBUTION - AXIAL

FAST FLUX DISTRIBUTION - RADIAL

FIG. 28

FAST FLUX DISTRIBUTION - AXIAL

FIG. 29

COMPARISON OF FAST FLUX ABOVE 1 MEV AT
PRESSURE VESSEL SURFACE

FIG. 30