IN-1143
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Issued: November 1967

USE OF THE ARMF FOR A FAST REACTOR SUPPORT PROGRAM

by

E. Fast, C. L. Beck, D. A. Millsap, R. G. Nisle,
J. W. Rogers and J. J. Scoville

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ACKNOWLEDGEMENTS

This report includes contributions directly and indirectly from many individuals. Measurements were made in the flux converter tube by the ARMF group under D. W. Knight. B. W. Howes was responsible for the foil activation measurements to determine flux spectral information. S. R. Sastry and V. A. Walker developed the conceptual design of a fast flux converter irradiation facility. Many others not specifically mentioned entered into the discussion of the overall program.

SUMMARY

It is proposed that available thermal reactors be used to support fast reactor research. Calculations show that flux spectra approximating those of fast reactors can be generated in a relatively small region in the center of the ARMF. Adequate sensitivity is available for making reactivity measurements of the usual small samples. The proposed concept also permits component parts of the fast flux core to be varied, thus producing a range of spectra. Irradiated specimens may be handled with a hot sample changer presently being used with a cadmium shielded facility. Specimens may be irradiated in an intense fast flux environment in a flux converter-thermal filter capsule in the MTR or ETR.

A simple converter tube facility with a UO_2 shielded sample volume was used to obtain preliminary measurements for comparison with calculations. Reactivity values were obtained with 17 specimens, mostly metals, including stainless steels and natural uranium. A 26-group one-dimensional diffusion code gave results that were in reasonable agreement with experiment in most cases. A proton-recoil spectrometer has been built for measurement of the keV-MeV neutron energy spectrum providing independent information to check calculations.

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I. INTRODUCTION

The trend in recent years has been toward the development of the fast reactor concept. Much fundamental data is still needed because of the lack of fast flux facilities in which to make measurements. Since thermal reactors--both high flux irradiation and low power reactivity measurement facilities--are available at this site, a study is under way to adapt these to measurements of interest in the fast reactor program.

The sensitivity of the Advanced Reactivity Measurement Facilities (ARMF-I and ARMF-II) to small thermal neutron perturbations is well known. Published results include, among others, measurement of resonance integrals of various elements, isotopes, and fission products⁽¹⁻⁴⁾; study of the transient behavior and I^{135} yields of fissile isotopes^(4,5); neutron reproduction parameters (η) for fissile isotopes⁽⁶⁾; reactivity changes in fertile isotopes with long-term burnup⁽⁷⁻⁹⁾; burnup of neutron absorber materials used in control rods and as burnable poisons⁽¹⁾, and reactivity effects of neutron scattering⁽¹⁰⁻¹²⁾. Details of construction and performance characteristics of the ARMF reactors are given elsewhere and will not be repeated in this report⁽¹³⁾.

The ARMF can be used to measure preferentially the response to fast neutrons by eliminating thermal and low energy neutrons from the sample volume and by enhancing the fast flux. A simple concept that approaches this case is a sample volume shielded by a fissile isotope, such as U^{235} , commonly called a "flux converter facility. Further tailoring of the real and adjoint flux spectra can be done by using various absorbers and

fissile isotopes in various combinations although in essentially the same geometric configuration. However, better results in the way of flux simulation are obtained with a larger mockup of a fast reactor core section driven by a thermal reactor to make a critical assembly. Such a concept is being used by the French in their low-power critical facility, MINERVE, in the ERMINE experiment⁽¹⁴⁾, which attains a neutron spectrum and importance very similar to the center of MASURCA.

The present report describes preliminary work done at the MTR site related to the use of thermal reactor facilities in support of fast reactor technology, and discusses studies relating to further potential along this line. Phases of this program to be presented here include: (1) studies on converter tube concepts for use in the ARMF, including some experiment results; (2) studies of core modification to give a coupled fast flux experimental region; (3) conceptual studies of an MTR or ETR converter irradiation capsule; (4) flux measurement techniques for measurement of fast flux spectra; (5) program of measurements. These topics will be discussed in the order listed. A general discussion and evaluation concludes the report.

II. ARMF FLUX CONVERTER TUBE FACILITY

A preliminary model of flux converter tubes was built for the ARMF and some measurements were made. Designs of possible future models have been studied calculationally. A simplified analysis of the reactivity effects is also presented.

A. Experimental

The converter tube facility, built as a preliminary test model, is shown in the assembly drawing in Figure 1. A hollow cylindrical stainless

Figure 1 - ARMF Flux Converter Facility

steel shell of 0.25 inch annulus was packed with enriched UO_2 to about 0.31 of theoretical density for a total of 432 gms of the oxide, which is equivalent to approximately 0.038 inch of U^{235} metal. The inside dimensions representing the experimental volume are 1.12 inch diameter and about 6 inches long. A hollow cap, containing enriched U^{235} metal foils, is attached to a lucite plastic rod on the one side. The other side has a quick disconnect coupling for sample capsules.

A typical sample capsule is also shown in the figure. The capsule is made of aluminum and is large enough to contain the usual cylindrical sample of about 0.86 inch diameter and 4.6 inches long. The quick-disconnect cap is threaded on one end to screw into the capsule.

A measurement in the ARMF was made in the following manner. A reading was taken of the regulating rod position consecutively with no sample in place, then with a sample in place, followed by another one of the unperturbed reactor. A reading consisted of 8 averages each of 256 or 512 samplings of the regulating rod position, taken at the rate of 16 per second, and punched out on an IBM punch card. These data cards along with pertinent calibration data were processed through a computer code⁽¹⁵⁾ to obtain the net reactivity of the sample. The calibration consisted of an analytical expression obtained from repeated measurement of a small sample at various parts of the regulating rod span and scaled by use of period measurements and the inhour equation to give reactivity in $\Delta k/k$. The unit adopted for ARMF measurements is $10^{-6} \Delta k/k$ called micro k (μk).

The samples which have been measured are listed in Table I. These were all solid cylindrical slugs and all except the fuel bearing samples were machined to the above dimensions. Results of reactivity measurements

TABLE I

Summary of ARMF Converter Tube Measurements

Sample		Calculated Reactivities ^(a) _(μk)				Measured Reactivity μ k	
Material	Weight gm	Absorption	Scatter	n,2n	Total	Total	Per Gram
Be.	85.912	- 84.3	211.7	356.4	483.9	642.00	7.473
C	76.482	- 0.26	123.2	----	122.9	163.90	2.143
Mg	78.147	- 125.4	52.8	----	-72.5	75.22	0.9625
Al(1)	121.383	- 17.2	88.5	----	71.3	82.60	0.6805
Al(2)	122.809					83.82	0.6825
Y	200.09	- 51.7	(b)	----	(b)	41.31	0.2065
Zr	300.755	- 50.6	101.0	----	50.4	128.04	0.4257
Sn	328.561					- 49.22	-0.1498
Pb	505.070	- 5.6	83.9	----	78.3	97.54	0.1931
Bi	444.700	----	82.6	1.4	83.9	99.69	0.2242
304 SS	348.84	- 367.1	156.5	----	210.6	-225.11	-0.6453
347 SS	349.87					-268.95	-0.7687
17-4 SS	341.58					-212.95	-0.6234
U (nat.)	673.6 ^(c)					74.98	0.1113
UO ₂ (nat.)	122.433 ^(c)					55.85	0.4562
UO ₂ (PBF depl. ternary)	72.483 ^(c,d)					0.72	0.0099
UO ₂ (PBF fuel, ternary)	41.872 ^(c,d)					31.90	0.7619

(a) Calculated reactivities of full length sample scaled by 0.31

(b) No scatter cross sections available

(c) Less than full sized sample

(d) 57.6 wt. % UO₂, rest ZrO₂ and small amount CaO

inside the converter tube are also tabulated, and will be discussed later. The choice of samples was made of materials or machined specimens that were readily available.

Flux measurements were made by activation of foils. Neutron flux of greater than 1 MeV was obtained from the threshold reactions $\text{Ni}^{58}(\text{n,p})\text{Co}^{58}$, $\text{Zn}^{64}(\text{n,p})\text{Cu}^{64}$ and $\text{Al}^{27}(\text{n},\alpha)\text{Na}^{24}$, and the assumed fission flux spectrum. Results were about $1.2 \times 10^7 \text{ n/cm}^2\text{sec}$ per watt of operating power. Thermal flux from copper activation, i.e., $\text{Cu}^{63}(\text{n},\gamma)\text{Cu}^{64}$ was about 2×10^6 per watt. The procedure was one usually used in thermal reactors where the assumption of $1/v$ cross section is fairly good, but was probably less accurate here. The value of thermal flux obtained by calculation appears to be considerably lower (compare Figure 3).

B. Interpretation of Reactivity Measurements

The resultant "reactivity" from an ARMF measurement is an integral value, the composite of all interactions contributing to this effect. The primary mechanisms are neutron absorption, fission, and scatter with occasionally some other that contributes significantly. Absorption always produces a negative component of reactivity and fission neutrons always give a positive contribution, whereas scatter may result in either positive or negative values depending on the neutron importance of the final relative to the initial energy state. Fortunately, not all of these phenomena contribute significantly in all sample measurements. The relative sensitivity varies considerably with measurement position and core loading. This fact has been used extensively in thermal reactor measurements to distinguish between poisons and fuels in samples^(16,17) on a two-energy group basis.

A perturbation theory approximation, applicable to dilute samples[†], assumes that the partial reactivity contributions will add algebraically:

$$\rho = \rho_a + \rho_f + \rho_s + \dots \quad (1)$$

For neutron absorption the reactivity

$$\rho_a \propto \int_E N \phi^*(E) \sigma_a(E) \phi(E) dE, \quad (2)$$

where N is the number of absorber nuclei in the sample, $\phi^*(E)$, $\phi(E)$ the adjoint and real fluxes respectively; and $\sigma_a(E)$ the neutron absorption cross section. The expression is integrated over all neutron energies. To obtain the averaged quantities for the energy spectrum, consider that the expression (2) may be written:

$$\begin{aligned} \rho_a &= \frac{CN \int_E \phi^*(E) \sigma_a(E) \phi(E) dE}{\int_E \sigma_a(E) \phi(E) dE} \frac{\int_E \sigma_a(E) \phi(E) dE}{\int_E \phi(E) dE} \int_E \phi(E) dE \\ &\equiv CN \bar{\phi}^* \bar{\sigma}_a \phi = N \bar{\sigma}_a \bar{W}_a = \bar{\Sigma}_a \bar{W}_a. \end{aligned} \quad (3)$$

The average statistical weight for absorption

$$W_a = \frac{\rho_a}{\bar{\Sigma}_a}. \quad (4)$$

The dependence on position in the reactor is not shown because it is assumed all quantities are approximately constant over the small sample volume. Also, since the sample is in the center of the reactor any gradient is a minimum.

[†]The case of non-dilute samples is treated approximately by Millsap in IDO-16977, pp. 14-18.

The usual fissionable isotope presents only a slight further complication. Normally, both absorption and fission are involved, giving a total reactivity which is the sum of these two reactions.

$$\rho = \rho_a + \rho_f$$

$$\propto N \int_E \phi^*(E) \sigma_a(E) \phi(E) dE + \int_{E_f} \phi^*(E_f) S(E_f) dE_f, \quad (5)$$

where $S(E_f)$ is the fission neutron production term. The absorption (first) integral term of (5) may be handled as was done above. The second integral term,

$$\begin{aligned} \int_{E_f} \phi^*(E_f) S(E_f) dE_f &= \frac{\int_{E_f} \phi^*(E_f) S(E_f) dE_f}{\int_{E_f} S(E_f) dE_f} \int_{E_f} S(E_f) dE_f \\ &\equiv \overline{\phi_f^*} \int_{E_f} S(E_f) dE_f. \end{aligned} \quad (6)$$

This final integral term is the total fission flux which is given by

$$\int_{E_f} S(E_f) dE_f = N \int_E \eta(E) \sigma_a(E) \phi(E) dE. \quad (7)$$

By defining averages analogous to the absorption case, one obtains for a fission sample

$$\begin{aligned} &= CN \overline{\sigma_a} \overline{\phi^*} \overline{\phi} + CN \overline{\sigma_a} \overline{\eta} \overline{\phi_f^*} \overline{\phi} \\ &\equiv \overline{\Sigma_a} \overline{W_a} + \overline{\eta} \overline{\Sigma_a} \overline{W_f} \end{aligned} \quad (8)$$

or

$$\frac{\rho}{\Sigma_a} = \bar{W}_a + \bar{\eta} \bar{W}_f \quad (9)$$

where $\bar{\eta} \bar{W}_f$ is the average statistical weight for fission energy neutrons that are produced.

The reactivity effect due to scattering of neutrons to a first approximation may be represented by

$$\rho_s \propto N \int_E^f \int_{E'}^f [\phi^*(E') - \phi^*(E)] \sigma_s(E \rightarrow E') \phi(E) dE' dE. \quad (10)$$

For elastic scattering, E' ranges from E to αE with about equal probability, where $\alpha \equiv \left(\frac{A-1}{A+1}\right)^2$ and A is the atomic weight of the nucleus. It is assumed that a single scattering event occurs[†]. If some inelastic interactions occur, which is the usual case with the heavier elements, the distribution of E' may be modified considerably. It is possible to simplify Eq. (10) to some extent when the right hand is broken up into two integrals, since there is no functional dependence on E' in one of them. However, it appears obvious that a statistical weight for scattering is not as simply defined as was done for absorption and fission. No common parameter has been found to date that might apply generally to all scattering nuclei.

Any other significant contribution to the total reactivity would be handled on an individual basis similar to the procedures outlined above.

The proposed procedure to establish standard statistical weights for analysis of burnup or generally "unknown" samples may be outlined as follows:

1. Determine absorption statistical weight with a sample having relatively high absorption cross section and negligible cross sections of other types. The absorption differential cross section must be well

[†]See Ref. (12) for the case of multiple scattering interactions.

known. Correction for a possible small contribution due to scattering will be made by calculation if necessary.

2. Consistency of the absorption statistical weight will be evaluated by determination of the same quantity from several sample measurements in any one core, and by repeating measurements of these absorber samples in one or more additional cores of different flux spectra.

3. With the absorption statistical weights established the scattering effects can be studied on samples in which this effect is appreciable. This study should aid in segregating the reactivity effect due to scattering for unknown samples and for comparison with calculations.

4. The fission statistical weight will be determined independently with a spontaneous Cf^{252} sample. This quantity will also be obtained from other fissile isotopes by use of absorption statistical weights as determined above and calculated scatter reactivities.

5. If any other mechanism contributes significantly to the reactivity, it can be deduced by use of the statistical weights determined above and the calculation thereof compared with other measurements.

C. Calculations of Flux Spectra

A multigroup one-dimensional diffusion code, MONA⁽¹⁸⁾, was used in flux and reactivity calculations for the ARMF with converter tube. A cylindrical model was assumed with 20 regions, 125 space points and 26 lethargy or energy groups. The ARMF core diagram is shown in Figure 2, from which a cylindricized model was assumed for calculational purposes. Table II lists the energy and lethargy intervals for the 26 groups used here. Details of the calculational procedure and the assumption are

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Figure 2 - ARMF-II Converter Facility Core Loading

TABLE II
MONA Group Data

<u>Group No.</u>	<u>Energy (eV)</u>		<u>Lethargy, u</u>		<u>Δu</u>
	<u>High</u>	<u>Low</u>	<u>Low</u>	<u>High</u>	
1	1.0×10^7	6.07×10^6	0	0.5	0.50
2	6.07×10^6	3.68	0.5	1.0	0.50
3	3.68	1.74	1.0	1.75	0.75
4	1.74	8.21×10^5	1.75	2.50	0.75
5	8.21×10^5	3.88	2.50	3.25	0.75
6	3.88	1.83	3.25	4.00	0.75
7	1.83	8.65×10^4	4.00	4.75	0.75
8	8.65×10^4	4.09	4.75	5.50	0.75
9	4.09	1.93	5.50	6.25	0.75
10	1.93	9.12×10^3	6.25	7.00	0.75
11	9.12	4.31	7.00	7.75	0.75
12	4.31	2.04	7.75	8.50	0.75
13	2.04	9.61×10^2	8.50	9.25	0.75
14	9.61×10^2	4.54	9.25	10.00	0.75
15	4.54	2.15	10.00	10.75	0.75
16	2.15	1.01	10.75	11.50	0.75
17	1.01	4.79×10^1	11.50	12.25	0.75
18	4.79×10^1	2.26	12.25	13.00	0.75
19	2.26	1.07	13.00	13.75	0.75
20	1.07	5.04×10^0	13.75	14.50	0.75
21	5.04×10^0	2.38	14.50	15.25	0.75
22	2.38	1.13	15.25	16.00	0.75
23	1.13	5.32×10^{-1}	16.00	16.75	0.75
24	5.32×10^{-1}	2.50	16.75	17.50	0.75
25	2.50	2.53×10^{-2}	17.50	19.80	2.30
26	2.53×10^{-2}	0	19.80	∞	∞

Figure 3 - Flux Spectra for UO_2 Converter Tube, ARMF-II
0.038 inch U^{235} Equivalent.

described elsewhere⁽¹⁹⁾. The calculations give group real and adjoint fluxes, and reactivities for the various processes for which group cross sections are available. Scatter reactivities are listed for a neutron scattering from any group to any lower energy group. Tables V and VI are illustrative of the detailed information which can be obtained.

Calculations were made for a series of possible converter type tubes for the ARMF. The group real and adjoint fluxes are given in Figures 3 to 11. The first of these (Figure 3) is for the converter tube actually built and used, for which experimental results have been given in Table I. It has an equivalent thickness of 0.038 inch of U metal in the 0.25 inch annulus thickness which is filled with UO_2 . Other calculated models include a double thickness of UO_2 , and various combinations of UO_2 with Cd, Hf and B_4C . The order in which the elements are listed is the order they are encountered in going from the outside to the axis of the facility. Included for comparison are the cases of the unperturbed reactor with fuel in the center position except for a 1.25 inch square hole (Figure 10) and this loading with a cadmium shielded tube used in resonance integral measurements (Figure 11). These two are the usual loadings used in the past for the ARMF-II.

From the results of these calculations, it is apparent that considerable tailoring is possible in both the real and adjoint fluxes. The greatest change from the unperturbed reactor, of course, lies in the thermal region. The peak flux is at the fission energy. In this range the adjoint generally shows its maximum rate of rise which results in a strong reactivity contribution from scattered neutrons. Erratic variations in adjoint at the low energy end may confuse the issue of scattered neutron contributions to some extent since both negative and positive reactivities

Figure 4 - Flux Spectra for UO₂ Converter Tube, ARMF-II
0.076 inch U²³⁵ Equivalent.

Figure 5 - Flux Spectra for UO_2 -Cd Converter Tube, ARMF-II

Figure 6 - Flux Spectra for UO_2 -Cd-Hf Converter Tube, ARMF-II

Figure 7 - Flux Spectra for Cd-Hf-UO₂ Converter Tube, ARMF-II

Figure 8 - Flux Spectra for $\text{UO}_2\text{-Cd-UO}_2$ Converter Tube, ARMF-II

Figure 9 - Flux Spectra for $\text{UO}_2\text{-B}_4\text{C}$ Converter Tube, ARMF-II

Figure 10 - Flux Spectra for Unperturbed ARMF-II

may result. The normally greater absorption cross section of the low energy end may predominate if there is any measurable flux, especially if the sample has high cross section resonances.

D. Discussion

The general trend in the fluxes appears obvious from an analysis of the calculated spectra. A fissile isotope near the inside of the shield keeps the adjoint in the low energy region high, whereas an absorber will depress it. The fast real flux is raised by fuel on the outside of the shield while thermal flux is depressed in any case of fuel or poison. The case of UO_2 outside, B_4C inside shows the most uniform gradation toward lower values in the low energy region of both real and adjoint fluxes.

The real flux spectrum of the converter facilities emphasizes the fission neutron peak relative to the keV range compared to what is usually found in fast reactors. At the same time the low energy or thermal component is relatively higher. This latter characteristic implies that a more effective absorber is needed for low energy neutrons, i.e., a thicker shield. To bring the peak energy down, a more extensive core mockup is needed, although some trend in this direction may be achieved with homogeneous mixtures of fuel absorbers and moderator materials. Further study along various lines will be done before the next model of converter tube is built.

The experience with the present converter tube has suggested several changes be made in the next model. (1) The tube should be full length through the core and of uniform construction. This not only would simplify calculations but would decrease the activation of the cap to which the sample is attached. (2) There is an appreciable thermal component of flux that should be eliminated by a neutronicallly thicker shield. (3) The tube should

Figure 11 - Flux Spectra for Cd-Shielded Tube in ARMF-II

be made so it can also be used with the hot sample changer. (4) Proton recoil spectrum measurements should be made before operating at any appreciable power, i.e., before any sample measurements are made, or a duplicate facility should be utilized to maintain a minimum gamma background.

Comparison of the calculated and experimental results of sample measurement in the converter tube is given in Table I. The calculated results are generally lower than the experimental. This would indicate that the assumed scale factor of 0.31 (ratio of normal to full core length sample reactivities) is low. Since the scale factor was determined in an aluminum air tube it should not necessarily be applicable here. In addition the cross section data for magnesium and zirconium are inadequate or the computational procedures need improvement which demonstrates valuable results from the program already carried out.

III. COUPLED FAST REACTIVITY MEASUREMENT FACILITY

A proposal was submitted several years ago for design and construction of a reactivity measurement facility sensitive to neutrons in the intermediate and fast energy range. A complete core can generally be made more representative of a power reactor core spectrum than the simple converter tube discussed previously. However, as shown by the French experiment ERMINE⁽¹⁴⁾, mockup of a relatively small core section with a thermal driver can be made to yield both real and adjoint flux spectra very similar to that found in a fast reactor.

The use of a fast core simulation with proper coupling to a thermal reactor as driver has a number of advantages. The most obvious is that of cost. Whereas a rough cost estimate of a separate fast critical reactor would be several million dollars, the smaller core coupled with present

ARMF systems should be at least an order of magnitude lower. At the same time, variations in such a loading can be made readily with a minimum of cost. With proper design the facility could be made to represent a large series of spectral types by varying component composition. Much of the ARMF sensitivity for reactivity measurement would be retained.

In the CFRMF design study, it was specified that the facility have (1) a maximum reactivity in the keV-MeV range; (2) a spectrum representative of fast reactors; (3) sensitivity sufficient for useful measurements and (4) minimum cost. Preliminary results of this study are given below. One concept uses the 4 x 4 element center section in the 6 x 6 grid of the ARMF. The other is designed for the 2 x 2 element center hole as is presently available in ARMF-I. Calculations are made using one-dimensional cylindrical models. Constants were obtained from PHROG⁽²⁰⁾ and INCITE⁽²¹⁾, and the one dimensional multigroup diffusion code MONA⁽¹⁸⁾ was used to calculate fluxes. The PERT-II⁽²²⁾ code served to calculate reactivity effects of the samples. The 26 energy groups were the same as those previously used and described in Table II.

A. Calculation Models of the CFRMF

Model 16-A

Initial calculations were made on a coupled system composed of an air cooled central fast core occupying 16 lattice positions, and an outer thermal driver zone of standard ARMF fuel elements completely surrounding this central core. In the mathematical description, five cylindrical regions were assumed with dimensions and atom densities as shown in Table III. The center fast core was enclosed in an aluminum tank with a small water gap between the tank and thermal section of the core. The fast fuel section was of plates containing 20% enriched uranium. A region highly poisoned with

TABLE III

Composition of CFRMF Model 16-A

Region →	1		2		3		4		5	
O.D. (cm) →	8		35.26		37.26		55.888		95.888	
Material	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density
H ₂ O	100				50	1.675×10^{22}		2.346×10^{22}	100	3.35×10^{22}
Al			28.57	1.720×10^{22}	50	3.010×10^{22}		1.777×10^{22}		
Fe			10	8.48×10^{21}						
Hf			2	8.98×10^{20}						
U-235			6.17	2.952×10^{21}				1.258×10^{20}		
U-238			24.69	1.181×10^{22}				7.7476×10^{18}		
VOID			28.57							
Mark-I ARMF-II Fuel							100

Figure 12 - Flux Spectra for CFMME 16-A

hafnium was included in an effort to decrease the temperature instability problems caused by power peaking near the interface and to reduce low energy flux. The overall power balance was 28.8% in the fast core and 71.2% in the thermal driver. Temperature was controlled in the fast core with forced air to keep variations to 1°C and the rise in temperature to 10°C at 100 watt operation.

Calculations of real and adjoint flux spectra at the center position are given in Figure 12. The neutron flux peaks at about 400 keV and the spectrum compares favorably with representative fast reactor spectra of EBR-II shown in Figure 13[†]. In the region of the regulating rod, the real and adjoint fluxes are of a magnitude the same as for the usual loading. Thus, the effectiveness of the reactivity observed should be comparable.

Model 16-B

This model "B" was designed to reduce the volume of air required by model A to maintain the temperature control as previously specified. A decoupler region of 0.063 cm cadmium was placed around the aluminum tank encasing the fast core. In addition, a 2 cm water gap was maintained between the fast and thermal core sections to flatten the power distribution through the thermal core. The thermal core remained unchanged while the dimensions of the fast core were reduced to allow for these changes. The fuel volume was decreased but the enrichment increased so that no significant changes resulted in the amount of U²³⁵ used. The assumed core composition represented as 8 annular regions is given in Table IV. Note the addition of boron to the aluminum in the tank. The calculated flux spectra are shown in Figure 14.

[†]Figure 13 is also inserted on a fold-out in back. It can be superposed on any of the figures for comparison.

Figure 13 - Real Flux Spectra from EBR-II

TABLE IV

Composition of CFRMF Model 16-B

Region→	1		2		3		4	
OD (cm)→	8.0		30.513		31.783		31.91	
Material	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density
H ₂ O	100		6	2.694x10 ⁺²¹	99	5.9598x10 ²²		
Al								
B								
Hf								
U-235								
U-238								
VOID								
Cd							100	4.64x10 ²²
Mark-I } ARMF-II }								
Fuel								
SS			38.57					

Region→	5		6		7		8			
OD (cm)→	35.91		37.18		55.808		95.808			
Material	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density		
H ₂ O	100	3.35x10 ²²	50	1.675x10 ²²		2.346x10 ²²	100	3.35x10 ²²		
Al										
B										
Hf										
U-235										
U-238										
VOID										
Cd										
Mark-I } ARMF-II } Fuel }					100	1.258x10 ²⁰ 7.7476x10 ¹⁸

Figure 14 - Flux Spectra for CFRMF 16-B

TABLE V

Calculated Reactivities for Carbon Slug in CFRMF 16-B

Group (g)	Absorption Reactivity (μk)	Scatter Reactivity (μk)				
		$g \rightarrow g + 1$	$g \rightarrow g + 2$	$g \rightarrow g + 3$	$g \rightarrow g + 4$	$g \rightarrow g + 5$
1		- 1.571	- 0.667	- 0.497	- 0.237	- 0.093
2		+ 0.288	- 0.080	- 0.050	- 0.010	+ 0.004
3		-11.735				
4		- 4.932				
5		+16.525				
6		+25.460				
7		+14.246				
8		+18.742				
9		+ 3.437				
10		+ 1.722				
11		- 0.581				
12		- 0.310				
13		+ 0.196				
14		+ 0.016				
15		- 0.047				
16		- 0.001				
17						
.						
.						
.						
26						
		-0.0				

Total net reactivity calculated 59.861 μk

Actual reactivity calculations were made for carbon and U^{235} . The detailed reactivity contributions of various interactions are given in Tables V and VI. Carbon contributes reactivities due to scatter only, which are both positive and negative, for a net total of about 60 μk . U^{235} , however, has components of absorption, fission, and scatter. Much of the high energy scatter is inelastic. In both cases, full core length samples of 2 cm OD were used. The correction factor of about 0.31 must be applied for the usual sample length of about 12.7 cm. The sensitivity in reactivity per macroscopic unit cross section is given in Table VII for each energy group and for absorption, fission and scatter. From this tabulation, the maximum sensitivity appears in about groups 4 to 6 or the $\times 10^5$ eV range. This is also the range in which the maximum reactivity contribution came in the case of both U^{235} and carbon.

Model 16-BW

To determine the shutdown margin required with the fast core flooded, a series of three calculations was carried out for the model 16-B loading. The reactivity was found to increase with flooding (assuming water added homogeneously throughout the fast core) until about 20 volume per cent was added. The excess reactivity reached a maximum of 2.4%. Additional water caused a slow decrease in reactivity as the increased moderation enhanced the effectiveness of the poison. More poison could be added to compensate for the excess reactivity under flooded conditions without seriously affecting the normal operation of this facility.

The above analysis suggests the possibility of water cooling in part of the fast core. This would simplify the cooling problem by eliminating the need for special air ducts, and reduce considerably the size of the water tight tank required.

TABLE VI

Calculated Reactivities for U²³⁵ Slug in CFRMF 16-B

Group (g)	Absorption Reactivity (μk)	Fission Reactivity (μk)	Scatter Reactivity (μk)				
			$g \rightarrow g + 1$	$g \rightarrow g + 2$	$g \rightarrow g + 3$	$g \rightarrow g + 4$	$g \rightarrow g + 5$
1	- 23.605	82.348	- 0.233	- 0.855	- 2.242	- 1.593	- 0.502
2	- 83.677	241.865	+ 0.083	- 2.983	- 3.250	- 0.578	+ 0.189
3	- 372.734	948.230	- 9.716	- 14.843	- 3.573	+ 1.006	+ 0.738
4	- 500.069	1257.847	- 5.462	+ 8.134	+ 10.521	4.776	2.061
5	- 497.255	1177.337	14.169	16.928	8.337	3.682	1.031
6	- 500.480	1060.489	12.348	7.290	4.084	1.228	0.353
7	- 451.152	830.716	4.061	2.713	0.926	0.292	0.055
8	- 355.175	606.393	2.860	0.329	0.116	0.021	0.003
9	- 235.801	358.083	0.406				
10	- 104.606	156.176	0.288				
11	- 58.782	80.798	- 0.070				
12	- 34.298	45.794	- 0.037				
13	32.893	44.491	0.023				
14	- 17.269	22.610	0.002				
15	- 3.772	4.901	- 0.006				
16	- 0.480	0.739					
17	- 0.038	0.061					
18	- 0.001	0.002					
19			~0.0				
20							
. . . 26 }	~0.0	~0.0					

Total net reactivity calculated 3711.205 μk

TABLE VII

Calculated Reactivities per Unit Cross Section in CFRMF 16-B

Group (g)	Absorption Reactivity (μk)	Fission Reactivity (μk)	Scatter Reactivity (μk)				
			$g \rightarrow g + 1$	$g \rightarrow g + 2$	$g \rightarrow g + 3$	$g \rightarrow g + 4$	$g \rightarrow g + 5$
1	- 9107.68	7527.56	-1396.94	- 1364.86	- 1864.70	- 1980.28	- 1633.33
2	- 35793.19	34942.88	+ 148.89	- 2171.34	- 2707.87	- 1097.33	+ 1158.94
3	-142531.33	138568.89	-9201.07	-11328.73	- 4941.99	+ 4005.44	+10651.09
4	-190893.16	198393.49	-3046.23	+ 6097.87	+18908.19	+28422.98	+47587.95
5	-194316.72	205226.52	+9459.04	+22710.56	+32553.06	+52378.10	+59537.42
6	-169696.17	170904.30	11035.33	19231.76	35741.25	41703.24	50255.13
7	-121046.74	114464.88	5489.61	16547.04	20540.14	26267.86	21263.38
8	- 75130.72	67963.29	6565.07	8936.20	12337.02	9365.63	6248.77
9	- 39209.90	32618.89	1137.91	2770.13	1344.01	- 151.92	+ 1139.23
10	- 13952.48	11279.78	564.43	11.21	- 446.03	+ 0.46	+ 109.15
11	- 5495.86	4270.34	- 186.70	- 382.54	- 213.51	- 172.36	- 1005.54
12	- 2670.78	2148.20	- 98.52	- 13.49	+ 7.21	- 411.92	- 489.14
13	- 1876.64	1567.25	+ 62.04	+ 77.14	- 228.65	- 284.99	- 521.87
14	- 648.11	523.95	+ 5.05	- 97.18	- 116.01	- 195.20	- 287.04
15	- 94.51	75.81	- 14.79	- 17.52	- 28.98	- 42.26	- 82.19
16	- 8.15	7.75	- 0.28	- 1.45	- 2.81	- 6.89	- 6.53
17			- 0.07	- 0.14	- 0.38	- 0.35	- 0.35
18							
.							
.							
.							
26							
	~0.0	~0.0		~0.0			

The real flux spectrum is given in Figure 15. It is apparent from the magnitude of the flux in the eV range that low energy reactivity effects might overshadow the effects from the range of interest. However, a reduction in the amount of water or changing the type of moderator might improve the spectrum in the direction desired. In ERMINE graphite was used⁽¹⁴⁾ for the purpose of softening the spectrum.

Model 4-A

This model is an extended filter-converter assembly concept designed to occupy a 4-element center position. The ARMF-I normally is operated with this size center water hole. The reactor region description with this facility is given in Table VIII. The calculated real and adjoint fluxes are plotted in Figure 16. The real flux compares well with that of EBR-II although the peak is at slightly lower energy. The adjoint shows a steady drop in the region of interest in contrast to the other adjoint spectra of this series. The large amount of boron and hafnium make the low-energy flux negligible before the adjoint reverses its direction.

Reactivity calculations were also made for samples of carbon and U^{235} in this facility. Results are summarized in Tables IX and X. Only scattering effects register in the case of carbon in the energy region that has a significant neutron flux. The group contributions are negative with a maximum from groups 7 and 8 (40-200 keV). In U^{235} the fission contribution with positive reactivity contributions predominates, although both absorption and scatter reactivities are negative. The peak reactivities are in the same general energy region as before. Table XI lists the sensitivities as reactivity per macroscopic unit cross sections for absorption, fission, and scatter. Again, adequate response is indicated in the spectral region of interest.

Figure 15 - Real Flux Spectrum for CFRMF 16-BW.

TABLE VIII

Composition of CFRMF Model 4-A

Region →	1		2		3		4	
OD (cm) →	3.21		17.35		18.62		35.91	
Material	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density
H ₂ O								2.346×10^{22}
Al					99	5.9598×10^{22}		1.777×10^{22}
B			10		1			
Hf			10	4.49×10^{21}				
U-235			1	3.826×10^{20}				1.258×10^{20}
U-238			79	3.788×10^{22}				7.7476×10^{18}
VOID	100							
Cd								
Mark-I } ARMF-II } Fuel }							100	

Region →	5		6		7		
OD (cm) →	37.18		55.808		95.808		
Material	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	
H ₂ O	50	1.675×10^{22}		2.346×10^{22}	100	3.35×10^{22}	
Al	50	3.01×10^{22}		1.777×10^{22}			
B							
Hf							
U-235				1.258×10^{20}			
U-238				7.7476×10^{18}			
VOID							
Cd							
Mark-I } ARMF-II } Fuel }			100

Figure 16 - Flux Spectra for CFRMF 4-A

TABLE IX

Calculated Reactivities for Carbon Slug in CFRMF 4-A

Group (g)	Absorption Reactivity (μk)	Scatter Reactivity (μk)				
		$g \rightarrow g + 1$	$g \rightarrow g + 2$	$g \rightarrow g + 3$	$g \rightarrow g + 4$	$g \rightarrow g + 5$
1		- 0.754	- 0.361	- 0.292	- 0.152	- 0.079
2		- 0.524	- 0.061	- 0.043	- 0.026	- 0.011
3		- 6.17				
4		- 5.97				
5		- 5.79				
6		-12.351				
7		-22.723				
8		-24.274				
9		- 8.518				
10		- 3.001				
11		- 1.414				
12		- 0.835				
13		- 0.248				
14		- 0.016				
15		- 0.015				
16						
.						
.						
.						
26						
		-0.0				

Total net reactivity calculated -93.64 μk

TABLE X

Calculated Reactivities for U^{235} Slug in CFRMF 4-A

Group (g)	Absorption Reactivity (μk)	Fission Reactivity (μk)	Scatter Reactivity (μk)				
			$g \rightarrow g + 1$	$g \rightarrow g + 2$	$g \rightarrow g + 3$	$g \rightarrow g + 4$	$g \rightarrow g + 5$
1	- 7.225	19.880	- 0.112	- 0.463	- 1.318	- 1.023	- 0.429
2	- 19.308	49.060	- 0.152	- 2.281	- 2.816	- 1.489	- 0.557
3	- 72.775	168.972	- 5.110	- 9.470	- 6.442	- 2.756	- 0.965
4	- 92.987	241.724	- 6.614	- 8.383	- 5.268	- 2.357	- 0.859
5	- 103.886	279.151	- 4.963	- 5.498	- 3.380	- 1.448	- 0.438
6	- 144.871	397.815	- 5.991	- 4.937	- 2.602	- 0.853	- 0.236
7	- 186.216	530.722	- 6.477	- 3.260	- 1.215	- 0.358	- 0.097
8	- 148.921	499.199	- 3.704	- 0.512	- 0.169	- 0.049	- 0.013
9	- 72.135	320.334	- 1.007				
10	- 31.977	187.822	- 0.502				
11	- 17.117	125.166	- 0.171				
12	- 8.381	80.296	- 0.100				
13	- 3.165	49.909	- 0.030				
14	- 0.843	23.533	- 0.002				
15	- 0.356	11.845	- 0.002				
16	- 0.006	0.436					
17		0.002					
. . . 26	~0.0	~0.0	~0.0				

Total net reactivity calculated 1967.399 μk

TABLE XI

Calculated Reactivities per Unit Cross Sections in CFRMF 4-A

Group (g)	Absorption Reactivity (μk)	Fission Reactivity (μk)	Scattered Ractivity				
			$g \rightarrow g + 1$	$g \rightarrow g + 2$	$g \rightarrow g + 3$	$g \rightarrow g + 4$	$g \rightarrow g + 5$
1	- 197.01	128.43	- 47.36	- 52.27	- 77.44	- 89.87	- 98.61
2	- 583.68	500.91	- 19.15	- 117.32	- 164.78	- 199.88	- 241.71
3	- 1966.73	1745.09	- 341.99	- 510.82	- 629.64	- 775.37	- 984.17
4	- 2508.63	2694.46	- 260.67	- 444.14	- 669.14	- 991.54	-1401.04
5	- 2869.06	3438.93	- 234.15	- 521.33	- 932.80	- 1455.44	-1787.81
6	- 3471.53	4530.85	- 378.36	- 920.48	-1609.07	- 2046.97	-2375.44
7	- 3531.02	5172.20	- 618.86	-1404.92	-1904.81	- 2279.77	-2634.86
8	- 2226.31	3954.08	- 600.93	- 983.09	-1269.75	- 1541.21	-1820.24
9	- 847.71	2062.24	- 199.31	- 348.82	- 490.40	- 635.92	- 728.01
10	- 301.43	958.71	- 69.50	- 135.32	- 202.97	- 245.79	- 254.73
11	- 113.10	467.52	- 32.10	- 65.09	- 85.97	- 90.33	- 103.06
12	- 46.12	266.20	- 18.79	- 30.67	- 33.15	- 40.41	- 38.67
13	- 12.76	124.25	- 5.55	- 6.71	- 10.09	- 9.28	- 10.85
14	- 2.24	38.54	- 0.36	- 1.41	- 1.16	- 1.64	- 1.80
15	- 0.63	12.95	- 0.35	- 0.27	- 0.43	- 0.48	- 0.59
16	- 0.007	0.32					
17		0.01					
.			-0.0				
.							
.							
26	~0.0	~0.0					

B. Discussion

Calculations have shown that both the 16-element and 4-element fast core in the ARMF can produce flux spectra representative of fast reactors. The larger core offers a slight advantage in that the spectral shape does not vary spatially near the region of the sample position. For the smaller core, the spatial flux distributions for energies above 9 keV change by less than 1% in going 1 cm radially outside the sample volume into the core. In the energy range 400 to 900 eV the change is less than 20%. Flux level below this energy is insignificant.

The adjoint flux spectrum is another important basis of comparison. It has a negative gradient in the smaller core giving rise to consistent negative reactivity due to scatter. The large core oscillates between positive and negative gradients which give both positive and negative scatter reactivity. Thus, an inventory of cores could be made having various spectral shapes for both real and adjoint fluxes. In this manner scattering effects could be cancelled or emphasized and various fast reactor spectra matched.

The variable spectrum capability would be most nearly practical for the smaller core. The necessary fuel inventory for the larger core would probably be prohibitive. For the 4-element core the concept pictured in Figure 17 is advanced as a method of realizing this capability with a minimum expense. The central square element contains the cylindrical hole in which samples are positioned for measurement. This element has the cross section of a normal ARMF element. Surrounding it are 12 elements of dimensions half that of the ARMF elements of $1/4$ the cross sectional area. These "cans" would be filled with loose leaf plates that could be exchanged according to the loading desired. They could be used for liquid

* Sample Tube
 + + Fast Flux Converter Elements (12)

Figure 17 - Generalized CFRMF Model 4-Type Cores

or powder material likewise. It is not proposed to detail this concept further at this time. The above suggestions should be sufficient to indicate the large number of permutations possible.

The sensitivity of either core appears ample for measurement of small samples in the energy range of interest. Contributions of reactivity reach a maximum for U^{235} at about 1 MeV for the larger core and at about 60 keV for the smaller one, with significant contributions in either case from 200 eV up to 10 MeV. The reactivity of 1 to 2 grams should be measurable to about $\pm 10\%$ and 50 grams to better than 1%. For carbon about 3 to 5 grams would give about $\pm 10\%$ and a full sized sample (2 cm OD and 12.7 cm long) would show a reproducibility in measurement of 1 to 2%.

IV. FAST FLUX IRRADIATION FACILITIES

An irradiation facility of high level fast neutron flux is needed for accelerated burnup of materials important to the fast reactor program. Several solutions to this problem are proposed and evaluated based on information available to date. As a further alternative for the immediate future, irradiation space may possibly be obtained in EBR-II. Flux spectra at a representative irradiation position is given in Figure 13. Cost for such irradiations is for sample handling only.

Preliminary work was done on a conceptual design of a flux converter and filter type capsule⁽²³⁾ intended for an MTR reflector position. The capsule proposed had an outer finned case of aluminum with tubular U^{235} converter element and cadmium or boron filter all immersed in sodium as the heat transfer medium. Inside the filter cylinder and along the capsule axis was a volume of about one inch diameter in which samples could be placed for irradiation. Calculations showed that such a capsule could take the stress and discharge the heat generated under reasonable

assumptions. The fast flux was estimated to be 0.5 to 1.0×10^{15} n/cm²sec (> 1 MeV).

A more precise calculation was made recently on the flux in such a capsule. For calculational study, the capsule was considered in an environment of the ARMF core which should be very nearly the neutron spectral composition of the MTR or ETR. The converter irradiation capsule was placed in the center and calculations made by the procedures used for the CFRMF flux determinations. Table XII gives the assumed capsule and core compositions in ten annular regions of the cylindrical geometry. It should be noted that sodium was replaced with equal density of aluminum because cross sections were not available for sodium. The spectral shape and fast flux gain will not be affected appreciably by this modification.

The problem was run without and with the UO₂ converter annulus. Results show about 1.5 times the original flux above 1 MeV. Although the gain in the fast flux would be greater in a reflector position, the magnitude of fast and thermal flux is greater in a fuel lattice position to give an overall advantage.

The flux spectrum of the center of the irradiation capsule is plotted in Figure 18. The thermal flux is adequately suppressed. However, the flux in the eV range is only a factor of 10 smaller than the peak value at about 1 MeV. Since absorption cross sections in the eV range are generally larger by more than a factor of 10 over those at 1 MeV, the reactions resulting from the lower energy would predominate in this simple model of irradiation capsule.

An alternate design to consider would be similar to that of the Model 4-A CFRMF. This insert would require a 6 x 6 inch test hole in the ETR. Although there is no particular gain in the fast flux the

Figure 18 - Real Flux Spectrum Fast Flux Irradiation Capsule

TABLE XII

Composition of Calculational Core for Fast Flux Irradiation Capsule

Region→	1		2		3		4		5	
O.D. (cm)→	2.53		3.162		4.338		4.772		5.228	
Material	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density
H ₂ O										
Al					42	2.54×10^{22}			50	3.01×10^{22}
B										
Hf			50	2.25×10^{22}						
UO ₂							100	2.23×10^{22}		
VOID	100									
Cd			50	2.32×10^{22}						
Mark-I										
ARMF-II										
Fuel										
SS										

Region→	6		7		8		9		10	
O.D. (cm)→	5.86		6.336		8.566		4.1766		81.766	
Material	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density	Vol. %	# Density
H ₂ O			67	2.23×10^{22}	100	3.35×10^{22}			100	3.35×10^{22}
Al	42	2.54×10^{22}	33	2.01×10^{22}						
B										
Hf										
UO ₂										
VOID										
Cd										
Mark-I										
ARMF-II							100			
Fuel										
SS										

spectral shape shows that no significant reactions would occur in the eV range. Thus, the burnup of samples would be due primarily to neutrons from the keV and MeV range. The ease with which variations in spectrum can be accomplished would offer the advantage of matching the spectra of various reactor of interest if such were desired. Sodium coolant could be used if necessary instead of air as was proposed for the CFRMF.

The hot sample changer (HSC) is a necessary adjunct to measurement of irradiated samples in the CFRMF. This facility has been operated successfully and can readily be adapted to the facilities proposed in this report. A flange matching the one on the bottom of the HSC would be welded at the proper height on the sample tube of the CFRMF. Samples would be made of uniform size acceptable to the HSC, i.e., about 1" diameter and 5 inches long.

Additional calculations need to be made to determine the most desirable type of irradiation facility. Burnup of the converter and filter elements and the spectral changes as a result must be determined.

V. FLUX SPECTRUM MEASUREMENTS

The general usefulness of facilities of the type described here in making measurements of reactor constants needed in fast reactor technology depends directly on how well the neutron energy spectrum is known from both experiment and calculation. If the spectrum is not well known, the reactivity measurement as an integral reaction rate cannot be universally applied. On the other hand, if the spectrum is well known and if calculations yield this spectrum to a fairly high degree of accuracy, the integral data can be generally useful. Then one can compare basic differential data through a calculated integral value and an experimentally determined

integral quantity. The proper interpretation of measured reactivity then yields values of use in any other reactor system.

To measure the possible flux range anticipated, a combination of various foil activation techniques and proton-recoil spectrometer measurements will be used.

A. Proton Recoil Spectrometer

A proton-recoil spectrometer patterned to a considerable extent after that of Bennett⁽²⁴⁾ but with some modifications has been built here. Improvements have been made, both in the tube design and in the electronics to extend the useful range of the measurement. A smaller diameter tube and lead shielding has improved the measurements in the low keV range which is of considerable interest. The spectrum of a Pu-Be neutron source was measured in a standard pile in the energy range from about 0.5 to 100 keV. The recoil spectrum was satisfactorily $1/E$ as expected over the greater portion of this range. Deviations occurred at the extremes. However, the detector tube thus calibrated is usable in other spectra also. Computer techniques for data reduction have been developed.

Preliminary experience in the flux converter tube shows that gamma background quickly exceeds that tolerable in proton-recoil spectrum measurements. Thus, no useful measurements were obtained in this facility. Future measurements will be made in any new facility before operating it at an appreciable power level, or a separate low power converter system would be maintained for spectral measurements.

B. Activation Techniques

The usual techniques of measuring fluxes by foil activation have been well developed and will not be discussed here. However, such techniques

have frequently been applied primarily to thermal reactors and the underlying assumptions must be carefully analyzed in view of the present application to fast flux spectra.

Activation monitors can be used to measure neutron flux in the resonance region by techniques already developed by Foell and Scoville⁽²⁵⁾. The procedure is a combination activation and reactivity measurement procedure. It depends on having a series of isotopes with single absorption resonances whose characteristics are well known and which are large compared to scattering effects. Information is obtained in this way which makes it possible to derive the statistical weight function.

The flux above 1 MeV may be determined from a number of standard threshold detectors. Other foil activation techniques will be studied for applicability to flux measurement in the range of interest of this program.

C. Standard Fast Flux Spectrum

It would be desirable to have available a reference fast spectrum which is amenable to calculation and over which various differential cross sections could be averaged. In the thermal reactor field there are the 2200 m/sec and 1/E flux averaged cross sections which have become standard. Perhaps integral quantities measured in the various configurations of the CFRMF would be of some use as reference fast spectra which are tractable to calculation. Various differential cross sections could be averaged over this spectrum giving one number to be tabulated for each and which would serve as an index to the properties of the material in question. Since no "standard" flux spectrum appears a natural choice (such as the 1/E in thermal reactors), a choice would have to be made rather arbitrarily with the requirements that it be well known, or representative of a fast power reactor, reproducible to fairly good precision and that measured and calculated values be in good agreement.

The type of facility proposed in the CFRMF is one that should not be too hard to produced elsewhere. A thermal critical facility with a flux trap in the center is not uncommon, and thus, the input spectrum should be constant. The material constituting the fast core insert could be specified to establish the center fast flux spectrum. Once the value of a material is established in the standard spectrum which is well known, it should be possible to translate it via calculation to other spectra.

VI. PROPOSED PROGRAM

The proposed program is to some degree an iteration among calculating, designing and building a facility, and procuring and experimenting with appropriate samples. The overall objective is to provide advanced understanding of cross section burnout and calculational methods which are needed for the physics measurements of fast reactors.

Calculations will be pursued both in support of sample measurements and to guide in facility design. The capabilities developed here make it possible to compare results of reactivity calculations and measurements obtained with specific samples. Agreement would generate confidence in the differential cross section data for use in reactor design calculations, and in the interpretation of reactivity components involved. Design of the next models converter tubes of the CFRMF core, and of the irradiation capsule will be chosen initially based on calculations.

Because of their simpler design, the converter tube types will be used first. Preliminary work has already been done and results reported above. The contemplated changes in the next generation of converter tube have been discussed (Section II-D). However, calculational study and subsequent design of the CFRMF cores will continue until a satisfactory model is obtained.

From the cost angle, convenience in use and flux spectra obtainable the smaller 4-element model, it now appears, will be developed as the most acceptable at this time. Without undue delays the first CFRMF model could be completed by the end of the fiscal year.

The program of sample measurement has two broad categories: the specimens of normal material and the irradiated specimens. The cold materials will consist of structural or moderator materials as illustrated in the measurements summarized in Table I. These can be handled directly in air. Also included are fresh (unirradiated) fuel elements and fertile materials. In as far as possible, a library of such pure materials and of representative fast reactor fuel specimens will be developed. In order to handle these interchangeably, they will be in some standard geometry or in a capsule of standard geometry.

The effect of high flux irradiation on specimens is of particular interest. Reactivity effects of burnup of fuel and buildup of fission products is of prime importance. Irradiations will be started as soon as possible depending on availability of samples, irradiating facilities, and the necessary pre-irradiation measurements in the ARMF converter or CFRMF. Typical fuel elements of fast reactors are desired as well as other forms of these materials.

A program of calibration measurements must be made initially to be able to interpret the results obtained with less well known materials. Some of the samples alluded to in the above paragraph will be used for this purpose. With the help of calculational support, the extent to which various interactions contribute to the total reactivity will be determined. An effort will be made to determine a functional dependence of reactivity due to neutron scatter. The reactivity effects of material combinations

likewise must be studied to determine to what extent the individual contributions are additive. The sensitivity of various materials and to what extent self-shielding is experienced must also be determined.

The indicated needs in calibration and understanding of reactivity mechanisms is most applicable in the case of burnup samples. Irradiation invariably introduces other nuclei of which the amounts and species may not be well known. This is particularly true in the case of fission products produced in burnup of fuel.

Limited capabilities are also available for making doppler measurements. The present procedure employs a small capsule into which the sample is placed and heated electrically while reactivity changes are recorded by the ARMF. Such measurements made in a fast flux spectrum should be of value. Present techniques could not handle irradiated samples, however.

VII. EVALUATION AND CONCLUSIONS

The flux spectra calculated for several models of the CFRMF are in good agreement with those of EBR-II, and well within the range of spectra found in the literature. Calculations show that it is possible to vary the spectrum over a considerable range by changes in core composition. The adjoint spectrum generally is not found in published papers. However, the information given about the ERMINE experiment⁽¹⁴⁾ allows calculation of this parameter. It is relatively flat, the maximum being only a factor of two greater than the minimum. This spectrum resembles in part the adjoint spectrum of CFRMF 16-B (Figure 14) in the range where the real flux is significant.

The reactivity obtained with samples of reasonable size is adequate for good reproducibility in sample measurements. The sensitivity should

be ample to detect small changes in samples due to burnup. The overall reactivity effect is enhanced in CFRMF 4-A because the negative slope of the adjoint causes a negative scatter reactivity which adds to that from absorption. In general, because of the low absorption cross sections the reactivity caused by neutron scatter may be the major contributor in non-fissile isotopes.

The capability of making measurements of this type on highly irradiated samples is probably unique. The hot sample changer, which makes it possible to handle irradiated samples, has performed very satisfactorily in actual tests. The proposed facilities will be designed to attach directly to the HSC.

The cost of the proposed CFRMF 4-A should be small compared to that of a separate fast reactivity measurement facility. Engineering cost estimates will be obtained after the core composition has been established.

Calculations of the flux converter tube case have given results in fair agreement with experiment except perhaps in two cases. The observed discrepancies generally may be attributed to uncertainties in the sample scaling factor in material cross sections, and in the flux spectrum. The necessity of a finite group structure may introduce errors due to in-group scattering which, though measured, is lost to calculation.

A considerable effort will be necessary in the way of calibration studies to interpret the CFRMF reactivity data. Some support from calculational determined reactivities may be necessary, particularly for scatter reactivity. Although some materials are very nearly pure scatterers the reactivity coefficient relating scatter reactivities among various materials has not been established.

The concept of standard fast flux spectra should be given further study. With a definite flux distribution in energy a single standard spectrum might be sufficient. However because of the span covered and the structure observed a better characterization of a material would certainly result in two or three different standard spectra. These should be chosen to emphasize different regions by position of peak flux and shape of adjoint.

In summary it is concluded that:

1. A fast flux core region in the center of the ARMF can give results of value to the fast reactor research effort.
2. Such results can be produced with a minimum delay time and monetary cost.
3. Calculational and experimental efforts complement each other to produce the most in useful results.
4. A satisfactory high intensity fast flux irradiation facility in a thermal test reactor is entirely feasible and can be put in operation in a short time at nominal cost.
5. A standard fast flux that is reproducible and amenable to calculational techniques can readily be obtained in the ARMF facility and could form a practical basis for characterizing material response usable in any fast reactor flux.

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