FISSION PRODUCT AND REACTOR DOSIMETRY STUDIES
AT
COUPLED FAST REACTIVITY MEASUREMENTS FACILITY

March 1978
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FISSION PRODUCT AND REACTOR DOSIMETRY STUDIES
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By
Y. D. Harker
J. W. Rogers
D. A. Millsap

Compiled by
Y. D. Harker

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ABSTRACT

The Coupled Fast Reactivity Measurements Facility (CFRMF) plays a special role in the Idaho National Engineering Laboratory's general program to provide basic technological data for the development of fission reactors as safe and economical sources of energy for domestic use. Because of its unique features the CFRMF has been used to provide key integral data in the areas of fast neutron fission product capture effects and fast reactor dosimetry development. This report has been prepared to provide in one document a specification of the CFRMF and the results of measurements performed in it. The data are expected to be used in the evaluation and testing of neutron cross-section files pertaining to fission products, dosimetry materials, and other special applications.
# CONTENTS

ACKNOWLEDGMENTS .............................................. ii

ABSTRACT ....................................................... iii

I. INTRODUCTION .................................................. 1

II. PHYSICAL DESCRIPTION OF THE CFRMF ......................... 4

III. CFRMF DIMENSIONAL SPECIFICATION FOR NEUTRONIC CALCULATIONS.. 12
    1. IMPORTANCE OF DIMENSIONAL SPECIFICATIONS .............. 12
    2. THE CALCULATIONAL MODEL ............................... 12
    3. INFORMATION FOR FURTHER CALCULATIONS ................. 15

IV. FISSION-PRODUCT CAPTURE CROSS SECTIONS .................. 26
    1. SOME NOTES ON RESULTS .................................. 26
    2. NORMALIZATION OF ACTIVATION DATA ..................... 27

V. DOSIMETRY REACTION RATE MEASUREMENTS AND CALCULATIONS .... 36

VI. THEORETICAL CALCULATIONS .................................. 39

VII. REFERENCES .................................................. 43

# FIGURES

1. Cutaway pictorial diagram showing general assembly of the CFRMF .............................................. 5

2. Midplane cross-sectional diagram of CFRMF core, control devices, and chamber locations ....................... 6

3. Midplane cross-sectional diagram of filter unit which forms the CFRMF fast zone ............................... 7

4. CFRMF power level control servo system used for maintaining a fixed neutron level ............................ 9

5. Diagram of CFRMF $r,z$ Calculational Model (Not drawn to scale) .............................................. 14

6. Generalized Diagram for Clad Regions ........................ 15

7. CFRMF Neutron Spectrum calculated from transport theory .. 42
TABLES

I. rz Model Dimensions ........................................ 17
II. Region Specifications ........................................ 18
III. Composition Specifications ................................. 20
IV. Material Specifications ....................................... 22
V-A. Detailed Specifications for Selected Compositions  
     (Cylindrical Geometry) ................................. 24
V-B. Detailed Specifications for Selected Compositions  
     (Slab Geometry) ........................................ 25
VI. Fission Product Reaction Rate ................................ 30
VII. CFRMF Measured Reaction Rates ............................ 37
VIII. Comparison of Measured and Calculated Reaction Rates  
     in CFRMF ................................................ 38
IX. CFRMF Central Neutron Spectrum ............................ 40
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I. INTRODUCTION

Nuclear data are used to support design and operation of reactor systems in the following ways:
1. establishing criteria for core and shielding design,
2. defining long-range fissile material requirements,
3. predicting and optimizing reactor system performance,
4. establishing requirements for nuclear waste processing.

Since the development of nuclear energy began, there have been programs in basic nuclear data measurements to provide and improve nuclear data in support of the advancement of nuclear power. Because the requirements and degree of sophistication in nuclear power development are continuing to advance as the industry matures, it is necessary to continue to maintain forward-looking programs in base technology development. The physics programs at the Idaho National Engineering Laboratory (INEL) are a part of this national effort in base technology development.

The information and data presented in this report represent the culmination of a particular phase in base technology development in the areas of:
1. cross-section data pertaining to fission-product-class nuclides,
2. reaction-rate data for materials utilized in reactor dosimetry.

In this report integral neutron-cross-section and reaction-rate data obtained from measurements performed in the Coupled Fast Reactivity Measurement Facility (CFRMF) are presented. Because the CFRMF neutron spectrum spans the intermediate neutron energy range (1 keV \( \leq E_n \leq 1 \) MeV) and is "a typical fast reactor type spectrum," integral cross-section and reaction-rate data obtained in this neutron field are important to base technology development pertaining to fast reactors. Principal activities involving measurements in the CFRMF are concerned with capture effects in fission product nuclides, reaction-rate determinations in selected dosimetry materials, and capture and fission effects in actinide nuclides.
Fission products are a class of nuclides which contain the largest number of nuclides for which nuclear data are desired and for which the least is known about the values of the neutron cross sections in the intermediate-neutron-energy range. The neutron capture effect due to the presence of these nuclides in the reactor fuel is known to be a very important factor in limiting the lifetime of the fuel. Such a limitation has a direct effect on the economics of the reactor operation. Secondly, the presence of these nuclides is a serious consideration in the handling and processing of irradiated fuel.

Reactor dosimetry has a significant part in evaluating reactor performance and in analyzing experiments related nuclear fuel damage. Dosimetry technology is being advanced through a coordinated multi-laboratory effort called the Interlaboratory LMFBR Reaction Rate (ILRR) program. In this program, reaction-rate measurements are being performed in well characterized neutron fields in the U. S. and Europe. Samples irradiated in these fields are analyzed by participating laboratories and the results are intercompared first to assess the state-of-the-art in determining reaction rates and second to improve on dosimetry methods. The CFRMF was selected as the first facility to be used in this program, and because of its well characterized neutron environment and its ability to operate at moderate flux levels (up to \(10^{12}\) n/cm\(^2\)-sec), it has been the most extensively used facility in the ILRR program.

Integral cross-section and reaction-rate data obtained from measurements in the CFRMF are used in evaluating and testing cross-section information to be included or already contained in the Evaluated Nuclear Data File (ENDF/B). Evaluation and testing is performed by members of the Cross Section Evaluations Working Group (CSEWG) which is a multi-U.S.-laboratory effort coordinated through the National Nuclear Data Center (NNDC) of Brookhaven National Laboratory (BNL). Personnel involved in the INEL physics programs are members of this working group. Part of our CSEWG commitment is to provide supporting documentation to maintain the CFRMF as a "benchmark" for integral testing data contained in ENDF/B. The issuance of this report is a part of our commitment to CSEWG and also to the ILRR.
The following areas are covered:

1. A physical description of the CFRMF and its control system are presented. In this section a brief description is given of the CFRMF which is sufficient to provide a suitable assessment of the facility for data evaluation and testing purposes. Greater detail concerning the facility and the extensive characterization program conducted in it are contained in the references [1-5].

2. A specification of the CFRMF suitable for R-Z transport calculations is contained. This specification is provided to assist anyone who undertakes to perform a transport calculation.

3. Results of capture cross-section measurements on selected fission product nuclides are presented. Fifty-two reactions involving fission-product nuclides are presented along with brief descriptions of sample configurations.

4. Results of reaction-rate measurements on dosimetry materials are presented. These data were obtained through the ILRR program and are presented in this report so that such data may be used to evaluate dosimetry cross-section data and to establish the degree of characterization achieved for the CFRMF.

5. As stated earlier, an extensive characterization program for the CFRMF has been in progress and part of this effort has been the use of reactor physics calculations. Reported in this section is the neutron spectrum derived from one-dimensional calculations using multi-group transport theory and ENDF/B IV cross-section data. This spectrum is recommended for use in data testing; however, a new recommended spectrum will be presented for each new version of ENDF/B.
II. PHYSICAL DESCRIPTION OF THE CFRMF

The Coupled Fast Reactivity Measurements Facility (CFRMF) at the Idaho National Engineering Laboratory (INEL) is a zoned-core critical assembly with a fast-neutron spectrum zone in the center of an enriched 235\textsubscript{U}, water-moderated thermal "driver," and the core is contained in a large pool about 4.5 m beneath the surface. The concept, preliminary considerations, and most details of the CFRMF have been documented\textsuperscript{[1-5]}. The CFRMF was originally designed as a high-precision static-reactivity measurements facility operating at a flux level of approximately \(10^8\) neutrons per square centimetre per second (n/cm\(^2\cdot\text{s}\)), but has more recently found much use in the fast breeder reactor program as a fast-neutron field for the irradiation of materials of interest at flux levels up to approximately \(10^{12}\) n/cm\(^2\cdot\text{s}\).

Figure 1 is a cutaway pictorial diagram of the CFRMF. As shown in Figure 2, which is a midplane cross-sectional diagram of the CFRMF, the core has quadrantal symmetry with respect to structural assembly and fuel loading. The thermal driver zone fuel elements are conventional plate elements of aluminum-clad, fully enriched (approximately 93\%) 235\textsubscript{U}. The fueled portion of the core is 60.96 cm long, and each element is 8.183 cm square. The fast zone is so constructed that it is kept water-free and so that it can be used to "filter" or "tailor" the neutron energy spectrum. Figure 3 is a horizontal midplane cross-sectional diagram of the fast zone assembly. The 0.317-cm-thick stainless steel housing has a water-free seal and is only thick enough to provide the strength necessary to support the large mass of uranium. The 0.635-cm-thick 50wt\% boral side and end plates completely shroud the assembly except that each boral end plate has a centered hole 4.20 cm in diameter, through which the experiment access tube passes. Outside each boral end plate is a 2.697-cm-thick aluminum spacer through which the access tube also extends. The lower aluminum plate positions the uranium block; the upper plate provides symmetry.

The 238\textsubscript{U} (99.7\%) block (14.52 cm square x 60.69 cm long and weighing approximately 217 kg) with a 5.295-cm hole drilled through its axial
Fig. 1 Cutaway pictorial diagram showing general assembly of the CFRMF.
Fig. 2. Midplane cross-sectional diagram of CFRMF core, control devices, and chamber locations.
Fig. 3. Midplane cross-sectional diagram of filter unit which forms the CFRMF fast zone.
center is vertically aligned with the core fuel. The 0.635-cm-thick annulus of enriched $^{10}$B (90%) and 0.0889-cm-thick annulus of enriched $^{235}$U (93%) are clad with 0.0305-cm-thick stainless steel and they slip-fit together into the $^{238}$U block. The $^{235}$U and $^{238}$U materials are solid metal, and the stainless steel is either Type 304 or 321. The $^{10}$B is crystalline powder and is vibrocompacted to a density of 1.355 g/cm$^3$. The stainless steel access tube, with 0.147-cm-thick walls in the core region, slip-fits inside the $^{235}$U annulus and will accept objects with effective diameters up to 3.78 cm. The distance from the bottom of the access tube to the core midplane is 31.11 cm; it is approximately 6.1 m from the core midplane to the top of the access tube above the pool surface. Asymmetry exists in the fast zone vertical plane because above the top aluminum spacer is a 15.75-cm cube-shaped void capped with a 2.54-cm-thick stainless steel cover plate through which the access tube passes. Below the bottom aluminum spacer is the 1.27-cm-thick stainless steel end plate of the housing, which rests on the aluminum-grid support structure of the reactor, with a metal-to-water ratio of approximately 0.6 by volume.

The materials in the fast zone assembly filter or tailor the thermal neutron energy spectrum of the driver zone in the following general manner. The boron in the boral plates attenuates the neutrons below approximately 1 keV by absorption. The $^{238}$U block degrades the energy of the fission neutrons by inelastic scattering. The $^{10}$B annulus attenuates the neutrons below approximately 1 keV even further and ensures that all thermal neutrons are removed. The $^{235}$U annulus has no significant effect on the real flux spectrum other than supplying some fast-fission neutrons, but this annulus does affect the importance function (adjoint flux) which is of concern in reactivity measurements.

When the CFRMF functions as a high-precision irradiation facility, the power-level control system must reproducibly establish the desired neutron level and energy spectrum without changing the flux distribution at the location of the experiment. Control device locations are shown in Figure 2. The safety rods are not used for normal operation control.
Fig. 4. CFRMF power-level-control servo system used for maintaining a fixed neutron level.
purposes and are completely out-of-core during irradiations. The shim rod, a 1-mm-thick, 15.24-cm x 60.96-cm cadmium plate on a flat surface parallel to the core, is always set at the same position for related irradiations and can be repositioned to within ±5 x 10^-4 of a degree of rotation (equivalent to 1.5 x 10^-7 Δk/k), which causes no detectable changes in the flux distribution in the experiment region.

The power of the CFRMF is maintained at a constant and preset level by means of a conventional dc servo system, shown in Figure 4. This servo system is composed of a boron-lined compensated ion chamber (CIC), appropriate amplifiers, a reference voltage source, and a torque motor-driven regulating rod (a 1-mm-thick, 6.0-cm x 60.96-cm cadmium plate on a curved surface 6.35 cm in diameter and parallel to the core). The signal applied to the first-stage amplifier is the difference between a current proportional to the reactor power level as produced by the servo CIC and a current obtained from the reference voltage source. The power-level error signal is then amplified and used to drive the regulating rod to correct for any power-level deviation. The regulating rod is capable of holding the power level steady with a total motion corresponding to a reactivity of ±8.4 x 10^-8 Δk/k at low power levels, where thermal heating causes no problems. At the power levels normally used for irradiations, the reactivity changes observed are approximately 1 x 10^-4 Δk/k; such changes also do not produce detectable changes in the in-core flux distribution. The servo CIC, reference voltage, and first-stage dc amplifier determine the reproducibility and linearity of multiples of the power level. The servo CIC positioning reproducibility has been measured to <±0.5%. During long-term (several years) observations, chambers such as the CFRMF servo CIC in neutron fields equal to or greater than that of the CFRMF have been so stable that no changes in stability have been detected. Routine checks ensure that the compensation and bias voltages of the servo CIC are stable and at the proper levels.

The reference voltage unit, which is used to preset the level at which the power is controlled, consists of a voltage power supply potentiometer and two switchable precision resistors (100x and 1000x). The
stability of this device is ±0.01% or better over the normal ambient temperature range. The linearity of the CFRMF power level control system has been checked against foil activations and fission rates in the fast zone center. A factor-of-10 change (0.6 to 6.0 kW) on the power level setpoint produces a change of 9.6 or a nonlinearity of 4%. This nonlinearity, however, is reproducible and stable. Measurements of the reference voltage supply output at each of these setpoints shows no more than 1% nonlinearity for each range of the system. Consequently, the nonlinearity of the power level control system is likely due to imperfect CIC compensation.
III. CFRMF DIMENSIONAL SPECIFICATIONS FOR NEUTRONIC CALCULATIONS

1. IMPORTANCE OF DIMENSIONAL SPECIFICATIONS

The central measurement position of the Coupled Fast Reactivity Measurements Facility (CFRMF) at the INEL is being used to obtain a variety of integral measurement data (reaction rates, reactivities, etc.) for reactor materials of interest to the Liquid Metal Fast Breeder Reactor and other fast reactor programs. Accurate knowledge of the neutron spectrum in the measurement position of the CFRMF is of major importance in the analysis of these data. In order to provide this knowledge an extensive program of spectrum measurements and calculations was initiated during the construction of the facility and has continued until the present. Because measurement techniques, cross-sectional evaluations, and reactor physics codes are continually being refined and improved, a sizeable effort in measurements and calculations is maintained in order to give the most up-to-date characterization possible of the central spectrum of the CFRMF.

During design and critical loading of the core, efforts were made to retain transverse and axial symmetry. However, practical limitations have necessitated certain departures from this ideal. Because of these departures from symmetry, radial and axial measurements and calculations must be made in order to determine the magnitude and nature of any effects that existing asymmetry may have on the spectrum in the central measurement position. In this section, we present, as accurately as possible, the dimensional specifications necessary to making r,z calculations for the study of axial flux distributions in the central spectrum of the CFRMF.

2. THE CALCULATIONAL MODEL

Since the CFRMF thermal driver zone and several important fast zone structures are rectangular in shape, it is impossible to give an exact r,z representation of the reactor. Furthermore there are sufficient
structural complexities that even a cylindrical version of the transverse dimensions requires considerable homogenization in order to make a calculation practicable. The calculational model presented here has homogenized most of the thermal driver region, and gives a fuel composition which preserves the masses of uranium, aluminum, and water actually present in the reactor. However, it should be noted that in fact the core contains fuel elements in the driver with six different fuel compositions. This homogenization of the fuel region will therefore give a somewhat inaccurate source distribution in the driver. This will affect the eigenvalue but should not bias axial flux distributions to any appreciable extent. The fast zone is, however, more explicitly represented. The axial representation has included all pertinent features above and below the core both for the fast region and the thermal driver zone which might influence the axial flux distributions of the central measurement position.

The details of the r,z calculational model are given in Tables I through V. Table I lists the dimensions of the interfaces used to define the regions of the model and their coded number designations (arabic numerals for the radial direction, roman numerals for axial direction). To identify each region numerically, the combined number of the axial and radial interfaces forming the upper left-hand boundary of each region is used to designate that region. Figure 5, the reactor model, shows this clearly. In this figure, which is not drawn to scale, all regions are represented as either simple cylinders or cylindrical shells.

Table II is a tabulation of the various regions in the model, including a brief description of the reactor components contained in each region, and the code number for the composition contained in each region. Table II is a tabulation of the various regions in the model, including a brief description of the reactor components contained in each region. Table III gives the volume fractions of the various materials contained in each composition code. Table IV lists the elements or isotopes contained in each of the materials along with their atom densities and the parameters used to calculate those atom densities. The material
Fig. 5. Diagram of CFRMF r,z Calculational Model (Not drawn to scale)
atom densities given were calculated by the relation

\[ N = \rho \frac{M}{M_T} \frac{N_A}{A} \]  

(1)

where

- \( N \) = material atom density (atom/cm\(^3\))
- \( \rho \) = material density (g/cm\(^3\))
- \( M/M_T \) = elemental or isotopic mass divided by the total mass of the material
- \( N_A \) = Avogadro's number (atoms/mole)
- \( A \) = atomic weight of the element or isotope g/mole).

When a composition atom density rather than material atom density is desired, the results of the expression above must be multiplied by the respective volume fractions given in Table III. If an isotope occurs in more than one material of a particular composition, then the composition atom density for that isotope is found by adding the material atom densities weighted by the appropriate volume fractions.

3. INFORMATION FOR FURTHER CALCULATIONS

If more calculations for resonance and/or thermal cross sections are desired, refer to the information given in Table V where additional information for the fuel element plates and the heavily shielded compositions is given. Figure 6 aids in interpreting the information given in the table.
For crosschecks, other information may also be of use. The material volume fractions given in Table III can be derived from the data given in Table V for composition numbers 3, 4, 6, and 8b; but the dimensional data for compositions 3 and 4 are for cylindrical shells, while compositions 6 and 8b are for slabs (as is 18.). Composition 8b is that of a single fuel plate. There are 15 plates in each fuel element and 32 fuel elements in the core. The cross-sectional area occupied by each fuel element is 69.4512 cm$^2$. Total fuel loading is 5698.9 g U-235 (93.16% enriched). The total cross-sectional area of the four cruciform-shaped safety rods is 86.2837 cm$^2$. The area of the assembly to the reflector is equivalent to 36 fuel elements plus the area of the safety rods. The weight of enriched uranium in the U-235 sleeve is 1494.7 g, and the weight of enriched boron in the $^{10}$B sleeve is 437.05 g.
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* Composition Number 8 contains 12.039% 8a (fuel element side plates) and 87.961% 8b (Homogenized fuel plates).
### TABLE IV
MATERIAL SPECIFICATIONS

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<th>Material Density (g/cm³)</th>
<th>ENDF/B Number</th>
<th>Element or Isotope</th>
<th>Elemental or Isotopic Mass Fraction</th>
<th>Elemental or Isotopic Mass (gm/mole) (Carbon-12 Scale)</th>
<th>Atom Density (Atoms/cm³) x 10⁻²⁴</th>
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TABLE V-A

DETAILED SPECIFICATIONS FOR SELECTED COMPOSITIONS (CYLINDRICAL GEOMETRY)

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<th>Radius In Core (cm)</th>
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Boron-10 Sleeve (Composition No. 4)

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<th>Radius In Core (cm)</th>
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f | 2.69748 |
### TABLE V-B
DETAILED SPECIFICATIONS FOR SELECTED COMPOSITIONS (SLAB GEOMETRY)

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IV. FISSION PRODUCT CAPTURE CROSS SECTIONS

Fission product capture sections have been measured by activation techniques using the Coupled Fast Reactivity Measurements Facility (CFRMF) as the irradiation source. A description of the CFRMF and the characterization of its neutron field is given in Section II and References 1-5. Activation measurements are performed by exposing a well-characterized sample along with flux monitors to the neutron environment in the test region of the CFRMF. Reaction rates are thus derived by measuring the activity of the reaction products by means of gamma spectroscopy. Run-to-run normalization and normalization relative to a standard are described in the following subsection.

1. SOME NOTES ON RESULTS

The results of these measurements are summarized in Table VI. No self-shielding corrections have been made to the data; the table includes a brief description of each sample, however, so that such corrections can be made if desired. Calculated integrals based on ENDF/B IV fission product data and the spectrum (SCAMP 75/271) given in Section V are also included in the table. Once again, no attempt has been made to account for sample self-shielding in the calculations.

Error estimates are included with the measured reaction rates. These estimates include all uncertainties contributing to the uncertainty of determining a reaction rate from an irradiation experiment of this type. The major contribution to the error in each case is the estimated uncertainty associated with the decay data of each reaction product. At this time, the errors are conservative estimates based on a preliminary review of the decay data used. Future review of the literature is expected to alter these estimates.

In the majority of comparisons between measured and calculated reaction rates, the measured values are smaller than the respective
calculated values. In many cases the differences are of a magnitude which suggests self-shielding effects. There will be further study to determine the magnitude of self-shielding for a selected set of reactions.

From Table VI it appears that agreement between measurements and calculations is in the range of 10-25%. In Section V, similar comparisons are presented for materials (reactions) utilized in reactor dosimetry. There the overall agreement is in the range of 5%. This agreement substantiates that the activation method is capable of accuracies of at least this magnitude when the data used in analysis (primarily decay parameters) are of sufficient quality. If, therefore, the decay data for the reaction products are improved through additional measurements or improved evaluations, the accuracy of the data presented here will benefit.

2. NORMALIZATION OF ACTIVATION DATA

Each irradiation experiment performed in the CFRMF for the purpose of measuring fission product capture by activation utilized at least one gold flux monitor foil (0.127) mm thick). In early activation experiments in the CFRMF, gold capture was considered the standard to which all activation measurements would be referenced. Relative reaction rates were determined on this basis by dividing the saturated reaction rate of the reaction under study by the gold monitor saturated reaction rate associated with that measurement.

As a result of this program's involvement in the Interlaboratory LMFBR Reaction Rate (ILRR) program, improved techniques of standardization became available, one of which was the determination of the integrated flux of the CFRMF using multifoil dosimetry. This technique uses the gold flux monitors associated with each experiment to provide run-to-run normalization, which can then be traced to an ILRR experiment for which an integrated flux was determined. Using this approach, spectral
average cross sections were determined by dividing the measured saturated reaction rate of the reaction under study by the integrated flux level determined for that irradiation.

Further study revealed that using gold as a standard and relying on dosimetry-determined integrated fluxes is inferior to using the fission rate of $^{235}\text{U}$ as the standard. As a result of CFRMF involvement in the ILRR program this rate has been determined to a high degree of accuracy ($\sim 1\%$) using a National Bureau of Standards fission chamber. On an absolute basis this measured rate is many times more accurate than the rate measured for gold capture or for the integrated flux determined from multifoil dosimetry measurements. Another advantage of using $^{235}\text{U}(n,f)$ as a standard is that the calculated spectral average fission cross section for the CFRMF neutron spectrum is much less dependent on the spectral shape chosen. Thus the conversion of the measured relative reaction rates to absolute spectral average cross sections using $[\sigma_f(235)\text{U}]_{\text{calc.}}$ is not as dependent on spectrum as the same procedure using $[\sigma_c(197)\text{Au}]_{\text{calc.}}$ or the conversion using the flux derived from multifoil dosimetry. For these reasons, it was decided to convert all activation data to a $^{235}\text{U}(n,f)$ standard. The following steps describe the method by which this conversion was accomplished.

1. The gold monitors used in the ILRR irradiations were thinner (0.0508 mm) and located differently than the gold monitors used in our activation measurements involving fission products. Thus an experiment was performed to correlate the results obtained from the two types of monitor configurations. The results of that correlation run are

$$\text{ILRR Gold Monitor}$$

reaction rate $= 4.785 \times 10^{-14}$ captures/sec atom ($\pm 0.06\%$) \hspace{1cm} (2)

$$\text{Activation Gold Monitor}$$

reaction rate $= 4.859 \times 10^{-14}$ capture/sec atom ($\pm 0.31\%$) \hspace{1cm} (3)
2. From the ILRR data the following parameters have been determined:

\[
\phi \sigma_f^{(235\text{U})} = 1.221 \times 10^{-13} \text{fissions/sec\cdot atom (± 1.4%)}.
\]  

\[\text{(5)}\]

A reaction rate for ILRR Au monitor =

\[3.008 \times 10^{-14} \text{ capture/sec\cdot atom (± 1.0%)}\]

\[\text{(6)}\]

3. Based on items 1 and 2, the fission product flux level monitor data are converted to \(^{235}\text{U} \) fission rates by applying the relationship

\[
\phi \sigma_f^{(235\text{U})} = 4.00 \times \text{[Fission Products Gold Monitor Capture Rate]} \ (± 1.7\%).
\]

\[\text{(7)}\]

4. For those data recorded relative to gold capture, the following relationship is applied to convert to \(^{235}\text{U(n,f)} \) standard:

\[
\frac{\phi \sigma_c(A,Z)}{\phi \sigma_f^{(235\text{U})}} = \left[ \frac{\phi \sigma_c(A,Z)}{\phi \sigma_c^{(197\text{Au})}} \right] \quad \text{4.00 (± 1.7%)}
\]

\[\text{(8)}\]
<table>
<thead>
<tr>
<th>Reaction</th>
<th>Sample Description</th>
<th>Sample Mass</th>
<th>Measured[a]</th>
<th>Calculated[b]</th>
<th>Measured/Calculated</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{87}$Rb(n,$\gamma$)$^{88}$Rb</td>
<td>RbCl powder in a disk shape 1 cm diameter x 1.6 mm thick</td>
<td>62.6 mg</td>
<td>$8.50 \times 10^{-3}$($\pm 12%$)</td>
<td>$7.46 \times 10^{-3}$</td>
<td>1.14</td>
</tr>
<tr>
<td>$^{89}$Y(n,$\gamma$)$^{90}$Y</td>
<td>Metal chips</td>
<td></td>
<td>$2.20 \times 10^{-4}$($\pm 7.7%$)</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>$^{93}$Nb(n,$\gamma$)$^{94}$Nb</td>
<td>Nb Metal foil .79 cm square x .13 mm thick</td>
<td>69 mg</td>
<td>$1.15 \times 10^{-1}$($\pm 50%$)</td>
<td>$7.65 \times 10^{-2}$</td>
<td>1.50</td>
</tr>
<tr>
<td>$^{99}$Tc(n,$\gamma$)$^{100}$Tc</td>
<td>Tc powder in a disk shape 1 cm diameter x 1.6 mm thick</td>
<td>99.2 mg</td>
<td>$1.72 \times 10^{-1}$($\pm 11%$)</td>
<td>$1.74 \times 10^{-1}$</td>
<td>.99</td>
</tr>
<tr>
<td>$^{98}$Mo(n,$\gamma$)$^{99}$Mo</td>
<td>Mo metal foil $\geq$1 cm diameter x 0.13 mm thick</td>
<td>99.0 mg</td>
<td>$3.81 \times 10^{-2}$($\pm 11%$)</td>
<td>$4.36 \times 10^{-2}$</td>
<td>.87</td>
</tr>
<tr>
<td>$^{100}$Mo(n,$\gamma$)$^{101}$Mo</td>
<td>Mo metal foil 1 cm diameter x 0.13 mm thick</td>
<td>99.0 mg</td>
<td>$2.24 \times 10^{-2}$($\pm 8.7%$)</td>
<td>$3.11 \times 10^{-2}$</td>
<td>.72</td>
</tr>
<tr>
<td>$^{102}$Ru(n,$\gamma$)$^{103}$Ru</td>
<td>Ru powder in a disk shape 1 cm diameter x 1.6 mm thick</td>
<td>87.5 mg</td>
<td>$5.55 \times 10^{-2}$($\pm 8.1%$)</td>
<td>$7.84 \times 10^{-2}$</td>
<td>.71</td>
</tr>
<tr>
<td>$^{104}$Ru(n,$\gamma$)$^{105}$Ru</td>
<td>Ru powder in a disk shape 1 cm diameter x 1.6 mm thick</td>
<td>75.4 mg</td>
<td>$5.18 \times 10^{-2}$($\pm 6.3%$)</td>
<td>$5.50 \times 10^{-2}$</td>
<td>.94</td>
</tr>
<tr>
<td>$^{103}$Rh(n,$\gamma$)$^{104}$Rh</td>
<td>Rh wire 0.254 mm diameter</td>
<td>6.3 mg</td>
<td>$2.02 \times 10^{-2}$($\pm 12%$)</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Reaction</td>
<td>Sample Description</td>
<td>Sample Mass</td>
<td>$\bar{\sigma}_C(A,Z) / \bar{\sigma}_f(235,92)$</td>
<td>Measured[a]</td>
<td>Calculated[b]</td>
</tr>
<tr>
<td>--------------------------</td>
<td>-------------------------------------</td>
<td>-------------</td>
<td>-----------------------------------------------</td>
<td>---------------------</td>
<td>---------------</td>
</tr>
<tr>
<td>$^{103}$Rh($n,\gamma$)$^{104}$Rh</td>
<td>Rh wire 0.254 mm diameter</td>
<td>6.3 mg per cm length</td>
<td>$2.15 \times 10^{-1}$ ($\pm 26%$)</td>
<td>- - -</td>
<td>- - -</td>
</tr>
<tr>
<td>$^{103}$Rh($n,\gamma$)$^{104}$Rh</td>
<td>- - -</td>
<td>$\sum$</td>
<td>$2.35 \times 10^{-1}$ ($\pm 24%$)</td>
<td>$2.64 \times 10^{-1}$</td>
<td>.89</td>
</tr>
<tr>
<td>$^{107}$Ag($n,\gamma$)$^{109}$Ag</td>
<td>Ag metal foil $\approx$ 1 cm diameter x .13 mm thick</td>
<td>100 mg</td>
<td>$2.75 \times 10^{-1}$ ($\pm 15%$)</td>
<td>$2.65 \times 10^{-1}$</td>
<td>1.04</td>
</tr>
<tr>
<td>$^{109}$Ag($n,\gamma$)$^{119}$Ag</td>
<td>Ag metal foil $\approx$ 1 cm diameter x .13 mm thick</td>
<td>100 mg</td>
<td>$3.11 \times 10^{-1}$ ($\pm 12%$)</td>
<td>- - -</td>
<td>- - -</td>
</tr>
<tr>
<td>$^{109}$Ag($n,\gamma$)$^{119}$Ag</td>
<td>Ag metal foil $\approx$ 1 cm diameter x .13 mm thick</td>
<td>100 mg</td>
<td>$1.80 \times 10^{-2}$ ($\pm 6.6%$)</td>
<td>- - -</td>
<td>- - -</td>
</tr>
<tr>
<td>$^{109}$Ag($n,\gamma$)$^{110}$Ag</td>
<td>- - -</td>
<td>$\sum$</td>
<td>$3.29 \times 10^{-1}$ ($\pm 11%$)</td>
<td>$1.92 \times 10^{-1}$</td>
<td>1.71</td>
</tr>
<tr>
<td>$^{108}$Pd($n,\gamma$)$^{109}$Pd</td>
<td>Pd metal foil $\approx$ 1 cm diameter x .13 mm thick</td>
<td>98.1 mg</td>
<td>$7.81 \times 10^{-2}$ ($\pm 25%$)</td>
<td>- - -</td>
<td>- - -</td>
</tr>
<tr>
<td>$^{110}$Pd($n,\gamma$)$^{111}$Pd</td>
<td>Pd metal foil $\approx$ 1 cm diameter x .13 mm thick</td>
<td>98.1 mg</td>
<td>$3.06 \times 10^{-3}$ ($\pm 10%$)</td>
<td>- - -</td>
<td>- - -</td>
</tr>
<tr>
<td>$^{115}$In($n,\gamma$)$^{116}$In</td>
<td>In metal foil .13 mm thick</td>
<td>63.7 mg</td>
<td>$1.67 \times 10^{-1}$ ($\pm 7.9%$)</td>
<td>$1.92 \times 10^{-1}$</td>
<td>.94</td>
</tr>
<tr>
<td>$^{121}$Sb($n,\gamma$)$^{122}$Sb</td>
<td>Sb powder in a disk shape 1 cm diameter x 1.6 mm thick</td>
<td>132.1 mg</td>
<td>$1.74 \times 10^{-1}$ ($\pm 6.3%$)</td>
<td>$1.93 \times 10^{-1}$</td>
<td>.90</td>
</tr>
<tr>
<td>Reaction</td>
<td>Sample Description</td>
<td>Sample Mass</td>
<td>$^\gamma_C (A,Z) / ^\gamma_C(235,92)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>--------------------------</td>
<td>-------------------------------------------------------------------------------------</td>
<td>-------------</td>
<td>--------------------------------------</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{123}$Sb($n,\gamma)^{124}$Sb</td>
<td>Sb powder in a disk shape 1 cm diameter x 1.6 mm thick</td>
<td>132.1 mg</td>
<td>$9.65 \times 10^{-2}(\pm 8.9%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{122}$I($n,\gamma)^{123}$I</td>
<td>AgI powder evenly distributed over 1 cm x 1.5 cm area</td>
<td>(127I) 5.51 mg</td>
<td>$1.85 \times 10^{-1}(\pm 19%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{128}$I($n,\gamma)^{130}$I</td>
<td>Ag$^{129}$I powder evenly distributed over 1 cm x 1.5 cm area</td>
<td>(129I) 5.29 mg</td>
<td>$1.14 \times 10^{-1}(\pm 8.6%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{132}$Xe($n,\gamma)^{133}$Xe</td>
<td>$^{132}$Xe implanted in Al foil over area 13.1 cm x 5 cm</td>
<td>(132Xe) 286.8 $\mu$g</td>
<td>$1.52 \times 10^{-3}(\pm 15%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{132}$Xe($n,\gamma)^{133}$Xe</td>
<td>$^{132}$Xe implanted in Al foil over area 13.1 cm x 5 cm</td>
<td>(132Xe) 286.8 $\mu$g</td>
<td>$2.45 \times 10^{-2}(\pm 8.0%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{132}$Xe($n,\gamma)^{133}$Xe</td>
<td>$^{132}$Xe implanted in Al foil over area 13.1 cm x 5 cm</td>
<td>$\sum$ 286.8 $\mu$g</td>
<td>$2.60 \times 10^{-2}(\pm 7.6%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{134}$Xe($n,\gamma)^{135}$Xe</td>
<td>$^{134}$Xe implanted in Al foil over area 13.2 cm x 5 cm</td>
<td>(134Xe) 106.0 $\mu$g</td>
<td>$9.29 \times 10^{-3}(\pm 6.6%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{133}$Cs($n,\gamma)^{134}$Cs</td>
<td>Cs$_2$SO$_4$ powder in a disk 1 cm diameter x 1.6 mm thick</td>
<td>69.6 mg</td>
<td>$3.47 \times 10^{-2}(\pm 16%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{133}$Cs($n,\gamma)^{134}$Cs</td>
<td>CsNO$_3$ powder in bottom of quartz vial-hemispherical 6.4 mm diameter</td>
<td>9.01 mg</td>
<td>$1.72 \times 10^{-1}(\pm 6.4%)$</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reaction</td>
<td>Sample Description</td>
<td>Sample Mass</td>
<td>Measured$^a$</td>
<td>Calculated$^b$</td>
<td>Measured/Calculated</td>
</tr>
<tr>
<td>----------</td>
<td>--------------------</td>
<td>-------------</td>
<td>--------------</td>
<td>----------------</td>
<td>---------------------</td>
</tr>
<tr>
<td>$^{150}$Nd(n,$\gamma$)$^{151}$Nd</td>
<td>Nd$_2$O$_3$ powder in a disk 1 cm diameter x 1.6 mm thick</td>
<td>60.82 mg</td>
<td>$6.02 \times 10^{-2}$ (± 21%)</td>
<td>$9.15 \times 10^{-1}$</td>
<td>.66</td>
</tr>
<tr>
<td>$^{152}$Sm(n,$\gamma$)$^{153}$Sm</td>
<td>Sm$_2$O$_3$ powder in a disk 1 cm diameter x 1.6 mm thick</td>
<td>87.39 mg</td>
<td>$1.82 \times 10^{-1}$ (± 4.9%)</td>
<td>$1.88 \times 10^{-1}$</td>
<td>.97</td>
</tr>
<tr>
<td>$^{154}$Sm(n,$\gamma$)$^{155}$Sm</td>
<td>Sm$_2$O$_3$ powder in a disk 1 cm diameter x 1.6 mm thick</td>
<td>87.39 mg</td>
<td>$7.33 \times 10^{-1}$ (± 4.5%)</td>
<td>$9.78 \times 10^{-1}$</td>
<td>.75</td>
</tr>
<tr>
<td>$^{151}$Eu(n,$\gamma$)$^{152m1}$Eu</td>
<td>$^{151}$EuO$_3$ powder in hemispherical bottom of quartz vial (6.4 mm diameter) 4 mm deep</td>
<td>15.33 mg</td>
<td>$6.75 \times 10^{-1}$ (± 8.4%)</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>$^{151}$Eu(n,$\gamma$)$^{152m2}$Eu</td>
<td>$^{151}$EuO$_3$ powder in polyethylene container</td>
<td>15.49 mg</td>
<td>$1.39 \times 10^{-3}$ (± 9.5%)</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>$^{151}$Eu(n,$\gamma$)$^{152g}$Eu</td>
<td>$^{151}$EuO$_3$ powder in hemispherical bottom of quartz vial (6.4 mm diameter) 4 mm deep</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>$^{151}$Eu(n,$\gamma$)$^{152}$Eu</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>$^{153}$Eu(n,$\gamma$)$^{154}$Eu</td>
<td>$^{153}$EuO$_3$ powder in hemispherical bottom of quartz vial (6.4 mm diameter) 4 mm deep</td>
<td>14.87 mg</td>
<td>$9.49 \times 10^{-1}$ (± 7.9%)</td>
<td>$8.90 \times 10^{-1}$</td>
<td>1.07</td>
</tr>
<tr>
<td>$^{158}$Gd(n,$\gamma$)$^{159}$Gd</td>
<td>Gd metal foil 1 cm diameter x 0.051 mm thick</td>
<td>30 mg</td>
<td>$1.12 \times 10^{-1}$ (±12%)</td>
<td>$1.15 \times 10^{-1}$</td>
<td>.98</td>
</tr>
</tbody>
</table>

$^a$ Measured

$^b$ Calculated

TABLE VI (Continued)
<table>
<thead>
<tr>
<th>Reaction</th>
<th>Sample Description</th>
<th>Sample Mass</th>
<th>$\sigma_c(A,Z)/\sigma_f(235,92)$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{133}$La(n,$\gamma$)$^{140}$La</td>
<td>La$_2$O$_3$ powder in a disk 1.27 cm diameter x 3.2 mm thick</td>
<td>416 mg</td>
<td>$1.14 \times 10^{-2}$ (± 6.3%)</td>
</tr>
<tr>
<td>$^{140}$Ce(n,$\gamma$)$^{143}$Ce</td>
<td>CeO$_2$ powder in a disk 1.27 cm diameter x 3.2 mm thick</td>
<td>431 mg</td>
<td>$5.45 \times 10^{-4}$ (± 8.7%)</td>
</tr>
<tr>
<td>$^{142}$Ce(n,$\gamma$)$^{143}$Ce</td>
<td>CeO$_2$ powder in a disk 1.27 cm diameter x 3.2 mm thick</td>
<td>431 mg</td>
<td>$1.16 \times 10^{-2}$ (± 11%)</td>
</tr>
<tr>
<td>$^{141}$Pr(n,$\gamma$)$^{142}$Pr</td>
<td>Pr powder in a disk 1 cm diameter x 1.6 mm thick</td>
<td>79.8 mg</td>
<td>$4.75 \times 10^{-2}$ (± 10%)</td>
</tr>
<tr>
<td>$^{147}$Pm(n,$\gamma$)$^{148}$Pm</td>
<td>$^{147}$Pm$_2$O$_3$ powder evenly distributed over 1 cm diameter area</td>
<td>13.6 mg</td>
<td>$2.21 \times 10^{-1}$ (± 13%)</td>
</tr>
<tr>
<td>$^{147}$Pm(n,$\gamma$)$^{148}$Pm</td>
<td>$^{147}$Pm$_2$O$_3$ powder evenly distributed over 1 cm diameter area</td>
<td>13.6 mg</td>
<td>$2.69 \times 10^{-1}$ (± 18%)</td>
</tr>
<tr>
<td>$^{147}$Pm(n,$\gamma$)$^{148}$Pm</td>
<td>- - -</td>
<td>$\sum$</td>
<td>$4.90 \times 10^{-1}$ (± 12%)</td>
</tr>
<tr>
<td>$^{146}$Nd(n,$\gamma$)$^{147}$Nd</td>
<td>Nd$_2$O$_3$ powder in a disk 1.25 cm diameter x 1.7 mm thick</td>
<td>84.63 mg</td>
<td>$4.15 \times 10^{-2}$ (± 14%)</td>
</tr>
<tr>
<td>$^{148}$Nd(n,$\gamma$)$^{149}$Nd</td>
<td>Nd$_2$O$_3$ powder in a disk 1 cm diameter x 1.6 mm thick</td>
<td>60.82 mg</td>
<td>$6.67 \times 10^{-2}$ (± 9.1%)</td>
</tr>
<tr>
<td>Reaction</td>
<td>Sample Description</td>
<td>Sample Mass</td>
<td>Measured$^a$</td>
</tr>
<tr>
<td>------------------</td>
<td>-------------------------------------</td>
<td>-------------</td>
<td>-----------------</td>
</tr>
<tr>
<td>$^{160}$Gd(n,γ)$^{161}$Gd</td>
<td>Gd metal foil 1 cm diameter x 0.051 mm thick</td>
<td>30 mg</td>
<td>$5.75 \times 10^{-2} (\pm 8.5%)$</td>
</tr>
<tr>
<td>$^{161}$Tm(n,γ)$^{170}$Tm</td>
<td>Tm metal foil .13 mm thick</td>
<td>78 mg</td>
<td>$2.93 \times 10^{-1} (\pm 12%)$</td>
</tr>
<tr>
<td>$^{181}$Ta(n,γ)$^{182}$Ta</td>
<td>Ta metal foil .05 mm thick</td>
<td>142.05 mg</td>
<td>$4.70 \times 10^{-4} (\pm 22%)$</td>
</tr>
<tr>
<td>$^{181}$Ta(n,γ)$^{182}$Ta</td>
<td>Ta metal foil .05 mm thick</td>
<td>142.05 mg</td>
<td>$3.29 \times 10^{-1} (\pm 9.0%)$</td>
</tr>
<tr>
<td>$^{181}$Ta(n,γ)$^{182}$Ta</td>
<td>- - -</td>
<td></td>
<td>$\sum = 3.29 \times 10^{-1} (\pm 9.0%)$</td>
</tr>
<tr>
<td>$^{186}$W(n,γ)$^{187}$W</td>
<td>W metal foil .127 mm thick</td>
<td>258 mg</td>
<td>$9.00 \times 10^{-2} (\pm 13%)$</td>
</tr>
</tbody>
</table>

$^a$ $\sigma_f(235,92)$ was measured using a National Bureau of Standards fission chamber; run-to-run normalization is based on gold flux monitors included with each irradiation run.

$^b$ Calculated ratios are based on ENDF/B IV cross-section data. The neutron spectrum used in obtaining spectral averages was derived from one-dimension transport theory (SCAMP-72/271) from ENDF/B IV data (section VI).

$[\sigma_f(235,92)]$ calc. = 1.595 barn/atom.
V. DOSIMETRY REACTION RATE
MEASUREMENTS AND CALCULATIONS

Integral reaction rates have been obtained for fissionable and non-fissionable dosimeter materials irradiated in the CFRMF neutron field. Fission chamber, solid-state track recorder, gamma-ray spectrometry, and helium mass spectrometry measurements and analysis techniques were used to determine these reaction rates. Several laboratories were involved in the planning, preparation, and execution of these measurements; the results presented here represent the efforts of many people associated with these laboratories. Table VII summarizes the measured reaction rates in CFRMF at the 6-kW power level, which corresponds to $7.9 \times 10^{10}$ n/cm$^2$·sec ($\pm$ 1.9%).

Using the calculated neutron spectrum from Section V, reaction rates have been calculated for all measured reaction rates. Table VIII lists these reaction rates relative to the $^{235}$U fission rate and compares them by ratios of measured to calculated rates.
<table>
<thead>
<tr>
<th>Reaction</th>
<th>Reaction Rate [reactions/sec·atom] x 10^{15}</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{235}\text{U}(n,f)$</td>
<td>122.1 ± 1.4%</td>
</tr>
<tr>
<td>$^{238}\text{U}(n,f)$</td>
<td>5.96 ± 1.9%</td>
</tr>
<tr>
<td>$^{239}\text{Pu}(n,f)$</td>
<td>142.3 ± 1.6%</td>
</tr>
<tr>
<td>$^{237}\text{Np}(n,f)$</td>
<td>43.5 ± 1.9%</td>
</tr>
<tr>
<td>$^{45}\text{Sc}(n,\gamma)^{46}\text{Sc}$</td>
<td>1.845 ± 1.8%</td>
</tr>
<tr>
<td>$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$</td>
<td>0.481 ± 1.5%</td>
</tr>
<tr>
<td>$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$</td>
<td>7.197 ± 2.3%</td>
</tr>
<tr>
<td>$^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$</td>
<td>3.56 ± 4.5%</td>
</tr>
<tr>
<td>$^{115}\text{In}(n,\gamma)^{116}\text{In}$</td>
<td>22.1 ± 2.5%</td>
</tr>
<tr>
<td>$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$</td>
<td>33.3 ± 1.0%</td>
</tr>
<tr>
<td>$^{238}\text{U}(n,\gamma)^{239}\text{U}$</td>
<td>17.5$^{[a]}$ ± 2.4%</td>
</tr>
<tr>
<td>$^{27}\text{Al}(n,\gamma)^{24}\text{Na}$</td>
<td>0.0127 ± 1.2%</td>
</tr>
<tr>
<td>$^{27}\text{Al}(n,p)^{27}\text{Mg}$</td>
<td>0.0687 ± 2.0%</td>
</tr>
<tr>
<td>$^{46}\text{Ti}(n,p)^{46}\text{Sc}$</td>
<td>0.205 ± 2.0%</td>
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<tr>
<td>$^{47}\text{Ti}(n,p)^{47}\text{Sc}$</td>
<td>0.328 ± 4.0%</td>
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<tr>
<td>$^{48}\text{Ti}(n,p)^{48}\text{Sc}$</td>
<td>0.00541 ± 2.0%</td>
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<tr>
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<tr>
<td>$^{10}\text{B}(n,\text{total}^4\text{He})$</td>
<td>145.8 ± 1.6%</td>
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[a] Corrected for resonance self-shielding
### TABLE VIII
COMPARISON OF MEASURED AND CALCULATED REACTION RATES IN CFRMF

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<th>Reaction</th>
<th>Measured(^{[a]})</th>
<th>Calculated</th>
<th>Measured/Calculated</th>
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<td>1.00 (Ref.)</td>
<td>1.00 (Ref.)</td>
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<td>(^{237}\text{Np}(n,f))</td>
<td>0.356</td>
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<td>(^{45}\text{Sc}(n,\gamma)) (^{26}\text{Sc})</td>
<td>0.0151</td>
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<td>(^{58}\text{Fe}(n,\gamma)) (^{59}\text{Fe})</td>
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<td>0.261</td>
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<tr>
<td>(^{238}\text{U}(n,\gamma)) (^{239}\text{U})</td>
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<td>1(^{a})B(n, total (^{4})He)</td>
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\(^{[a]}\) Relative to \(^{235}\text{U}\) fission rate. Spectral averaged cross sections can be obtained by multiplying listed values by \(\sigma_f(92.235) = 1.595\) barns/atom.

\(^{[b]}\) Corrected for resonance self-shielding.
VI. THEORETICAL CALCULATIONS

In the effort to characterize the CFRMF spectrum, transport, Monte Carlo and resonance theory, computerized techniques have been applied. The versatility of the transport code (SCAMP, ID, P1, S6) with its many options and relatively short computation time made it quite useful for most of the calculations. Monte Carlo calculations were used to test for smoothing effects in the cross-section processing, and only minor differences were observed. The resonance theory spectrum also showed only minor differences in comparison with the transport theory in the resonance energy region. Extensive tests of the cylindrical model to give the best representation of the complex CFRMF assembly have shown the best compromise between neutron energy group structure and material regions to be 71 energy groups with 0.25 lethargy spacing from 21 MeV down. Both cell and full-core models have been used in the calculations, with only minor differences resulting. ENDF/B versions III and IV have both been used, and version IV data produce more relative flux above 0.5 MeV due primarily to the version IV changes in $^{238}$U inelastic scattering. At this time the recommended spectrum for CFRMF is the full-core cylindrical model calculation using ENDF/F version IV cross-section data. This spectrum is tabulated in Table IX. In Figure 7, this spectrum is compared with a similar spectrum derived using ENDF/B III data. The spectrum derived from ENDF/B data spectrum is recommended because it agrees well in shape with the measurements in the energy regions where the various measurements are considered to be most credible and because it is impossible to make spectrum measurements over the entire energy range of interest.
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<th>$E_i$ (eV)</th>
<th>$\phi(U_i)$</th>
<th>$\frac{U_i}{\int_{U_i} \phi(U)du}$</th>
<th>$i$</th>
<th>$U_i$</th>
<th>$E_i$ (eV)</th>
<th>$\phi(U_i)$</th>
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<td>2.98 x 10^{-12}</td>
<td>70</td>
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<td>.532</td>
<td>1.21 x 10^{-13}</td>
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<td>71</td>
<td>17.00</td>
<td>.414</td>
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<td>1.01 x 10^{-14}</td>
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Fig. 7. CFRMF Neutron Spectrum calculated from transport theory.
VII. REFERENCES


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