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EBR-II DOSIMETRY TEST DATA ANALYSIS
(REACTOR RUNS 31E AND 31F)

June 1970

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(REACTOR RUNS 31E AND 31F)

By

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ABSTRACT

Data from the Pacific Northwest Laboratory (PNL) and Argonne National Laboratory (ANL) dosimetry test as analyzed by PNL, yield the result of 2.1×10^{15} n/cm²-sec for the total flux level at EBR-II Row 2 - midplane as measured in a dosimetry structural subassembly and at a stated reactor power of 50 MW. The value of 2.1×10^{15} n/cm²-sec now agrees to within ~10 to 15% with values based on reactor physics and power considerations and fuel burnup measurements. Previously reported larger differences are attributed mainly to use of incorrect cross sections data and in the stated value of low power for Run 25a. Results from a number of fast reactor tests require a reduction in previously reported EBR-II core spectral averaged cross section values of 25% for the ⁵⁴Fe(n,p) and 30% for ⁵⁸Ni(n,p) reaction to achieve consistency with results from fission reactions. Measured reaction rates were used to generate flux-spectra solutions for different radial and axial locations. For Row 2 - axially, flux-spectral differences were observed between low and high power and between the edge and center of a structural subassembly. It is concluded that fission and nonfission flux monitors are necessary in experimental subassemblies to secure accurate flux and fluence data for materials irradiation tests in EBR-II.

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EBR-II DOSIMETRY TEST DATA ANALYSIS
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I. INTRODUCTION

The design, characteristics, and operation of the Experimental Breeder Reactor II (EBR-II) as a primary irradiation facility for the Liquid Metal Fast Breeder Reactor program has been reported by Kirn.(1)

A cooperative dosimetry test involving the Pacific Northwest Laboratory (PNL) and Argonne National Laboratory (ANL) was conducted in EBR-II during reactor Runs 31E and 31F. The test was to resolve previously observed^(2,3) differences between predicted and measured integral fluxes, to provide measured values of energy-dependent integral fluxes as a function of radial and axial position, and to test and compare the results of different dosimetry methods. In order to accomplish this, the test was designed to answer the following questions:

- Do physics calculations, reactor power considerations, **burnup** measurements, and activation and fission dosimeters provide the same estimate of flux levels at different locations in EBR-II?
- Do structural subassemblies perturb the flux significantly?
- Can low-power flux measurements be extrapolated to predict high power flux-spectral values?
- What dosimetry methods are required for measuring **flux-spectra** and fluence in fast reactors such as **EBR-II**, and how accurate are the results?

This report provides a description of the test, the methods of analysis used by PNL personnel to interpret the dosimetry data, and the results of the PNL analysis. **Two other** reports

document the measurements of reaction rates^(4,5) for the 31E and 31F Runs and the selection of evaluated energy-dependent cross sections.⁽⁶⁾

11. SUMMARY AND CONCLUSIONS

The PNL analysis of data from the PNL-ANL EBR-II dosimetry experiment yields the result that the effective total flux at Row 2 - midplane in the dosimetry structural subassembly and at a stated reactor power of 50 MW, is 2.06×10^{15} n/cm²-sec.⁽⁷⁾ This results from SAND-II⁽⁸⁾ iteration runs using up to 17 reactions: $^{235}\text{U}(\text{n},\text{f})$, $^{238}\text{U}(\text{n},\text{f})$, $^{237}\text{Np}(\text{n},\text{f})$, $^{232}\text{Th}(\text{n},\text{f})$, $^{58}\text{Fe}(\text{n},\gamma)$, $^{54}\text{Fe}(\text{n},\text{p})$, $^{46}\text{Ti}(\text{n},\text{p})$, $^{47}\text{Ti}(\text{n},\text{p})$, $^{48}\text{Ti}(\text{n},\text{p})$, $^{45}\text{Sc}(\text{n},\gamma)$, $^{58}\text{Ni}(\text{n},\text{p})$, $^{63}\text{Cu}(\text{n},\gamma)$, $^{23}\text{Na}(\text{n},\gamma)$, $^{197}\text{Au}(\text{n},\gamma)$, $^{63}\text{Cu}(\text{n},\text{u})$, $^{27}\text{Al}(\text{n},\alpha)$, and $^{238}\text{U}(\text{n},\gamma)$. Separate analyses of fission and activation monitors are in agreement to within ~5%. Applying a maximum estimated 4% correction for the flux depression in a structural relative to a fueled subassembly, yields 2.14×10^{15} n/cm²-sec compared with 2.4 to 2.5×10^{15} n/cm²-sec based on reactor physics calculations, power considerations, and burnup analyses. Though this constitutes better agreement than found in previous tests, an unresolved 10 to 15% difference remains.

This better agreement results partly from changing the cross sections for the activation monitors. Results from this and other fast reactor tests required a 25 to 30% reduction in previously reported EBR-II core spectral averaged cross sections for the $^{54}\text{Fe}(\text{n},\text{p})$ and $^{58}\text{Ni}(\text{n},\text{p})$ reactions to achieve consistency with fission foil results.⁽⁷⁾ The energy-dependent cross sections, $\sigma(E)$, for most nonfission reactions used in this study were adjusted by SAND-II analysis of activation results from different fast reactor spectra, including EBR-II.⁽⁶⁾ Calculated spectral-averaged cross sections based on these adjusted $\sigma(E)$'s now agree within error limits assigned to measured values.

The reasons for previously reported^(2,3) differences between low and high-power total integral flux values have been clarified, but differences in spectral shapes have been observed between the low-power run, 31E, and the high-power run, 31F. Why these differences occur is not understood. For the 31F Run, local flux perturbation effects (as high as ~12%) have been observed within a structural subassembly. Such effects can cause significant depressions in the high-energy (>~2 MeV) part of the flux-spectrum^(9,10) and, if not accounted for, can introduce significant errors in total flux estimates based on individual flux monitors such as $^{54}\text{Fe}(n,p)$, $^{46}\text{Ti}(n,p)$, $^{63}\text{Cu}(n,\alpha)$, etc.

At distances up to ~59 cm above reactor midplane, some multiple foil results are up to 5 times greater than results from reactor physics calculations. This is not surprising since out-of-core regions are beyond the intended range of applicability of the diffusion code calculations used in this study. A transport or Monte Carlo code analysis should yield better results.

It is concluded that fission and nonfission flux monitors are necessary in experimental subassemblies to secure accurate flux and fluence data for materials irradiation tests in EBR-II.

III. TEST DESCRIPTION

A. FLUX MONITOR SELECTION

Monitor reactions are generated by neutrons from different energy ranges. Better accuracy in integral flux determinations can be achieved by the use of monitors with reactions that are sensitive to the energy regions comprising the major portion of the spectrum.⁽⁸⁾ While this is the primary criterion, selection of a set of monitors was also based on the following additional criteria.

- (1) Monitor materials must have melting points in excess of the anticipated experimental operating temperature and be compatible with the environment.
- (2) The half life of the product isotopes must be long enough to allow ample time for removal from the reactor and counting.
- (3) The activation cross sections for the materials must be well defined.
- (4) High purity materials must be used to reduce analytical problems created by the interference from activation of contaminants.

Using these criteria, a set of 16 monitor materials (21 reactions) were selected for this test. The set is described in Table 1 which gives the melting points, reaction product isotopes, product isotope half lives, and the upper (E_U) and lower (E_L) energy bounds of neutrons which produce 90% of the product isotope activity both at an in-core position and at a position near the top of Row 7.

All of the spectrum is covered as evidenced by the 90% response ranges given in Table 1 and illustrated by bars in Figures 1 and 2. Monitor coverage in-core extends from ~ 10 MeV down to around $\sim 10^{-2}$ MeV and less than 0.3% of the flux lies outside this range. At the experimental position near the top of Row 7, where there are fewer high energy neutrons, monitor coverage extends from ~ 10 MeV down to approximately 10^{-6} MeV. Therefore, the monitors shown will provide information on spectra from core center to positions well outside the core.

Not all the foils or foil reactions selected were used for the final analysis. For instance, based on a study by Armani⁽¹¹⁾ for fission reactions, interference from the

TABLE 1. Monitor Data

Monitor Material	Form	Melting Point, °F	Reaction (a)	Half Life	Energy Response Range of Monitor Reactions			
					Core Center-Midplane		Upper Blanket	
					E _L , MeV	E _u , MeV	E _L , MeV	E _u , MeV
KI	Pellet	1333	$^{127}\text{I}(n,2n)^{126}\text{I}$	13.1d	10.1	14.9	10.0	15.0
Ti	Wire	3272	$*^{48}\text{Ti}(n,p)^{48}\text{Sc}$	44.0h	6.7	13.6	6.8	13.9
Mg	Foil	1204	$^{24}\text{Mg}(n,p)^{24}\text{Na}$	15.0h	6.5	11.8	6.6	13.0
Al	Wire	1219	$*^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	15.0h	6.4	12.5	6.5	12.8
Cu	Foil	1981	$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$	5.268y	6.1	11.8	6.1	12.0
Ti			$*^{46}\text{Ti}(n,p)^{46}\text{Sc}$	84.0d	3.3	9.4	3.3	9.8
Ni	Wire	2651	$^{60}\text{Ni}(n,p)^{60}\text{Co}$	5.268y	2.6	9.8	2.5	9.9
Ti			$*^{47}\text{Ti}(n,p)^{47}\text{Sc}$	3.43d	2.0	6.7	1.8	6.4
Fe	Wire	2795	$*^{54}\text{Fe}(n,p)^{54}\text{Mn}$	313.5d	2.0	7.8	1.7	7.8
Ni			$*^{58}\text{Ni}(n,p)^{58}\text{Co}$	71.3d	1.9	7.3	1.6	7.2
Th	Foil	3353	$^{232}\text{Th}(n,f)^{140}\text{Ba}$	12.8d	1.4	6.9	1.4	6.8
U _{dep}	Foil	2071	$*^{238}\text{U}(n,f)^{140}\text{Ba}$	12.8d	1.4	6.1	1.2	5.1
N _p	Foil	1184	$*^{237}\text{Np}(n,f)^{140}\text{Ba}$	12.8d	4.8×10^{-1}	4.2	3.6×10^{-2}	2.5
Fe			$*^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	44.5d	3.8×10^{-2}	1.6	2.1×10^{-4}	2.4×10^{-1}
U _{enr}	Foil	2071	$*^{235}\text{U}(n,f)^{140}\text{Ba}$	12.8d	3.6×10^{-2}	2.7	3.4×10^{-5}	3.2×10^{-1}
NaCl	Pellet	1473	$*^{23}\text{Na}(n,\gamma)^{24}\text{Na}$	15.0h	2.6×10^{-2}	1.5	2.8×10^{-5}	2.3×10^{-1}
U _{dep}			$(b)^{238}\text{U}(n,\gamma)^{239}\text{Np}$	2.35d	2.2×10^{-2}	1.4	2.0×10^{-5}	1.1×10^{-1}
Sc	Foil	2192	$^{45}\text{Sc}(n,\gamma)^{46}\text{Sc}$	84.0d	1.7×10^{-2}	1.1	5.8×10^{-6}	2.2×10^{-1}
Au	Foil	1945	$*^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	2.7d	1.7×10^{-2}	0.96	4.8×10^{-6}	8.0×10^{-2}
Co	Foil	2723	$*^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	5.268y	1.4×10^{-2}	1.0	1.1×10^{-4}	4.3×10^{-3}
Cu			$*^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$	12.8h	8.0×10^{-3}	1.5	5.8×10^{-4}	1.6×10^{-1}

a. * denotes the 14 reactions used to determine flux-spectra at all r,z locations

b. $^{238}\text{U}(n,\gamma)^{239}\text{U} \xrightarrow[23 \text{ minutes}]{\beta} ^{239}\text{Np}$.

Neg 701853-1

FIGURE 1

90% Monitor Response Range in EBR-II Core

Neg 701853-5

FIGURE 2. 90% Monitor Response Range in EBR-II Axial Blanket

$^{98}\text{Mo}(n,\gamma)^{99}\text{Mo}$ reaction in the stainless steel of the capsule was found to invalidate all of the ^{99}Mo results at the out-of-core positions. The environmental affects were underestimated in the case of scandium and potassium iodide and these materials lost their integrity in the longer high-power run. Because of the inconsistency in the counting data, possibly a result of diffusion loss of the reaction products, the $^{24}\text{Mg}(n,p)^{24}\text{Na}$ and $^{35}\text{Cl}(n,a)^{32}\text{P}$ reactions were not used. The final set of monitor materials and 14 reactions used for most of the subsequent analyses in this report are starred in Table 1.

B. FLUX MONITOR ENCAPSULATION

As shown in Figure 3, the following design features of the monitor capsule helped to ensure the success of the test:

- (1) The Type 304 SS Container Body was small, approximately 1 in. long by 0.328-in. OD with an 0.032 in. wall. It enclosed 15 neutron monitors in fixed positions. This allowed pinpointing the position of any monitor to within 1/8-in. for flux gradient corrections.
- (2) Titanium spacers (0.001 in. thick, shown in Figure 3), placed between the bare foils, prevented cross contamination and facilitated separation of the materials after Irradiation. The bare monitors with the Ti spacers were stacked on the Type 304 SS Monitor Holder which comprises the Foil Pack as shown in the lower part of Figure 3.
- (3) Materials that might melt (such as magnesium and aluminum) and that might fracture (potassium iodide and sodium chloride) were encapsulated in 1/2 to 1 in. long by ~0.003 in. wall by 0.093 in. OD Type 304 SS, see

Neg 701853-16

FIGURE 3. Spectral Set - Foil and Capsule Assembly

Figure 3. Fission foils were also encapsulated in the same manner to prevent cross contamination or loss of fission products.*

- (4) The capsule was made rugged enough to withstand extensive mechanical abuse and yet be easily disassembled.
- (5) Tolerances were controlled to provide good heat transfer and to restrict the maximum temperature of any specimen to less than 1400 °F. Exceptions were potassium iodide and sodium chloride; their low thermal conductivity could result in higher temperatures within a specimen.

CAPSULE LOCATION AND IDENTIFICATION

The subassemblies used in the test were standard 19-pin structural subassemblies with solid stainless steel rods replacing the pins not used for the test. Subassemblies were located in the reactor at positions shown in Figure 4, with pin locations shown as black dots in each subassembly. Figure 5 shows the actual core configuration for Run 31F.(12) Figure 6 shows the radial distance (r) from the reactor vertical centerline for each pin and the average axial distance from the reactor midplane to each monitor capsule (z).

The letters S, G, E, SS, B, SP, and SPF shown in the boxes refer to the type of materials in the container. The S designates a spectral capsule shown in Figure 3. The data in this report came primarily from monitors in these capsules. The G or gradient wire capsules were used to help interpolate between axial positions. E designates capsules from other EBR-II experimenters for correlation with this test. The SS or self-shielding capsules contained various thicknesses of pure cobalt (also a dilute cobalt, aluminum-cobalt alloy),

* *For future tests, it is concluded that a foil that might be subject to reaction product loss by diffusion and/or atom recoil should be counted within its containment capsule -- either destructively or directly as required.*

Neg 701770-12

FIGURE 4. Relative Location of Dosimetry Subassemblies
During EBR-II Run 31F

KEY: SETH - SST THIMBLE C# - CONTROL ROD
 OSC - OSCILLATOR ROD S# - SAFETY ROD
 BETH - BERYLLIUM THIMBLE D - DRIVER FUEL
 SSCR - SST CONTROL ROD P - 1/2 DRIVER FUEL, 1/2 SST
 B - DEPLETED URANIUM
 X - EXPERIMENTAL SUBASSEMBLY

Neg 701770-10

FIGURE 5. EBR-II Loading Pattern Run 31F
 from 11-5-68 to 11-14-68

Neg 701853-11

FIGURE 6. Dosimetry Tests 31E and 31F Monitor Capsule Locations

copper, and scandium. The B or blank capsule was used for spacing and contained gradient wires. The SP or stable product capsule (nonfissionable material) and the SPF or stable product fissionable capsules were included to study the generation of stable fission products and helium.

IRRADIATION TIME HISTORY

The time history of the two tests were:

31E (Low-Power 84 kW)

At power 11-4-68 2301 MST

Shutdown 11-5-68 0001 MST

Effective constant power 62 min

31F (High-Power 50 MW)

At power 11-5-68 2351 MST

Shutdown 11-14-68 0400 MST

Effective Constant power 8 days and 6 hr.

Analysis of the reactor power curves indicates that use of the effective constant power values given above would introduce less than $\sim 0.5\%$ error in the saturated activities or fission rates for the foil reactions used in this test. Plots of the power level as a function of time during the 31E and 31F runs are given in Reference 5, based on data provided by Kirn.(12)

IV. METHODS OF DATA ANALYSIS

A. REACTION RATE MEASUREMENTS

The counting procedures, use of the irradiation time history data given in Section III-D, activation data reduction factors, foil sizes and weights, fission yields, tabulation and statistical comparison of all of the participating laboratory

results, and the discussion of the counting results and related counting problems for this test are reported by Kellogg and Zimmer in a separate document.⁽⁵⁾ Their results were used to generate an r, z matrix of reaction rates to determine energy-dependent integral fluxes.

Except for the $^{59}\text{Co}(n, \gamma)^{60}\text{Co}$ monitor, foils did not require corrections for neutron self-shielding. However, for the cobalt foil (0.0005 in. thick) corrections of up to $\sim 20\%$ were required.

To determine corrections to the cobalt reaction rate, three self-shielding capsules were included in the Row 2 pin ($r = 5.2$ cm) 7, 31, and 51 cm above midplane. Each capsule contained three each of four different cobalt monitors. These were 0.0005 in. foil, 0.005 in. foil, and 0.020 in. OD wire of pure cobalt and a 0.020 in. OD wire of an aluminum alloy containing 0.483% cobalt. The alloy was encapsulated, but did not melt in this experiment.

Comparing the reaction rates of the pure materials to the alloy, show that the correction factor for the 0.0005 in. foil varied from 1.0 (no correction) at 7 cm above midplane, to 1.15 at 31 cm, to 1.20 at 51 cm. The 1.20 figure is approximately the correction factor expected for an 0.0005 in. foil irradiated in a $1/E$ spectrum. Thus, the self-shielding correction for the 0.0005 in. cobalt should not exceed ~ 1.20 at any location in EBR-II. Both the 0.005 in. foil and 0.020 in. OD wire were found to have correction factors of about 1.15 at 7 cm, 1.85 at 31 cm, and 2.4 at 51 cm above core midplane. Again, 2.4 is roughly the correction expected in a $1/E$ spectrum.

B. ENERGY-DEPENDENT CROSS-SECTION SELECTION

Determination of flux and fluence from measured reaction rates requires the use of a consistent set of evaluated energy-dependent cross sections. Reported⁽¹³⁻¹⁵⁾ indications of inconsistencies in energy-dependent cross sections motivated the revision of the SAND-II reference cross-section library⁽⁶⁾ used to translate reaction rates to flux and fluence with the SAND-II code.^(8,16,17)

Twelve cross-section revisions (or additions) came from literature review and 11 more were re-adjusted by an iterative unfolding procedure.⁽¹⁸⁾ Five reference spectra were used together with three reference fission and one reference activation reaction to re-adjust the cross sections by the unfolding technique. [The five spectra were the FISSION,⁽¹³⁾ APFA-III-Godiva,⁽¹⁴⁾ EBR-II core center,⁽⁷⁾ and ECEL Cores 14 and 16.⁽¹⁷⁾ The four reference reactions were $^{235}\text{U}(n,f)$, $^{238}\text{U}(n,f)$, $^{237}\text{Np}(n,f)$, and $^{27}\text{Al}(n,\alpha)$.] Results of the unfolding for the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ reaction, which is of prime interest to this study, are presented in Figure 7. Additional results for this and the 22 remaining reactions are given in Reference 6.

The heavy solid line in Figure 7 is the new re-adjusted energy-dependent cross section. The $\sigma(E)$ used as a best input approximation is specified in the figure.⁽¹⁹⁾ The dashed curve is the solution using a constant input. Fairly good definition of the cross section is implied by the similarity in the solutions based on the constant and best input forms. The dot-dash curve is the previously used and reported⁽²⁾ $\sigma(E)$ data. Obviously, there is a difference between the new (solid line) and the old (dot-dash line) evaluations.

Neg 701853-2
 FIGURE 7. SAND-II Solution for $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ Reaction Cross Section

For comparison, a recent evaluation of the iron reaction cross section by Kamphouse et al.^(20,21) for measurements in EBR-II based on three threshold reactions is also shown in Figure 7. On an integral basis their results and the present results are quite consistent. That is, their higher values between ~ 2 and 3 MeV are partially compensated for by lower values above 3 MeV. Since they used only three reactions [$^{54}\text{Fe}(n,p)$, $^{58}\text{Ni}(n,p)$, and $^{46}\text{Ti}(n,p)$] in defining their $\sigma(E)$ curves, the overall agreement is good.

C. INTEGRAL FLUX-SPECTRA

1. Multiple Foil Method

a. General Discussion

These methods determine neutron flux-spectra from a set of measured reaction rates. A set of simultaneous equations relate detector reaction rates to the differential flux. The solution is a differential flux spectrum for which the calculated reaction rates agree (within acceptable limits of error) with the measured values. While a number of computer codes treat this problem, three in more common use are SAND-II.^(8,16,17) SPECTRA,⁽²²⁾ and RDMM.^(23,24) An ASTM document provides a brief description of each code.⁽²⁵⁾ The pertinent references give detailed information and user instructions.

The SAND-II code, which was utilized for the present study, has had wide use to determine integral flux-spectra in both thermal and fast reactors.^(13-15,17) The calculation procedure consists of selection of a trial spectrum, and iteration from that spectrum to an appropriate solution to the system of reaction rate equations. An appropriate solution must be physically reasonable as well as having the yielded calculated detector reaction rates which agree with the measured values.

b. SAND-II Solution Accuracy

The integral flux greater than a specified energy, $\Phi(>E)$, is defined by

$$\Phi(>E) \equiv \int_E^{\infty} \phi(E) \, dE, \quad (1)$$

where $\phi(E)$ is the differential neutron flux. The accuracy of $\Phi(>E)$, derived from the SAND-II code, is dependent upon uncertainties in cross sections, measured reactions rates, and solution definition (uniqueness)*. An estimate was required of the effect of these uncertainties on the accuracy of $\Phi(>E)$ values for the EBR-II Runs 31E and 31F dosimetry test.

Previous error estimate studies indicated that with a properly selected foil set, the combined errors from all sources should introduce no more than about a 10 to 30% absolute uncertainty in SAND-II derived integral flux values. (8,13-17) Of the three sources of uncertainty, lack of solution definition is, most often, the largest single contributor to the total error and required, therefore, separate study.

Using results based on the most reliable 14 foil reactions for the EBR-II 31F run (and without use of any specific knowledge of the expected spectrum), calculations established an estimated maximum error resulting from lack of solution definition of no more than about 5% in the core and bordering areas and about 25% in the blanket region (see following section).

**In the case of a set of multiple foil activation (reaction rate) integral equations, it is necessary to place bounds on the mathematical uncertainty of a given solution. On a purely mathematical basis, a single "unique" solution can be obtained only in the limit when the number of equations approaches infinity. For a finite set of foils, physical considerations help to establish expected bounds of the true solution, as discussed in the next section.*

Prior knowledge about the flux-spectrum, for example from a reactor physics calculation, could substantially reduce the 25% error.

It has been determined that the absolute error in measured reaction rates^(4,5) and cross sections^(6,8,17) for individual monitors (when integrated over any EBR-II spectrum) are about +7% (lo) and $\pm 10\%$ (lo), respectively.* A computerized technique to handle the propagation of multiple reaction errors is under development; but meanwhile a conservative expediency is used for the present study. The $\pm 7\%$ and $\pm 10\%$ errors are applied across the board for all neutron energies as indicated below and combined to give a 12% (la) value for all energies.

For this dosimetry test, then, a conservative estimate of the maximum (lo) error $\sigma_T(>E)$ in integral flux, $\Phi(>E)$, is defined by the relationship

$$\sigma_T(>E) = \pm[\delta(>E) + 12\%] . \quad (2)$$

$\delta(>E)$ is obtained from Figure 8, and is half the estimated maximum plus or minus percentage error resulting from lack of solution definition.** The assumption is that the solution definition error provides a conservative estimate of an equivalent statistical 2σ (at the 95% confidence level) uncertainty.

Use of a relationship such as Equation (2) is required to obtain error estimates for $\Phi(>E)$ that include all sources of uncertainty. When appropriate, Equation (2) has been used to obtain error estimates for integral flux values reported in subsequent sections.

* *These results are consistent with those obtained in the analysis of multiple foil results in previous thermal and fast reactor tests, see References 17, 26, and 27.*

** *In general, the largest values of $\delta(>E)$ are found for regions of inadequate foil coverage. It is expected, therefore, that if a "ratio" >1 is found, it will be equally possible to obtain a $\Phi(>E)$ solution with an equivalent value <1 .*

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FIGURE 8. Comparison of Derived and Known Integral Fluxes for Four Bounding r,z Locations

c. SAND-II Solution Uniqueness Study

How well data from 14 foil reactions (assuming no counting or cross-section errors) can define EBR-II spectra, will now be considered. Results for other thermal and fast reactor spectra are given elsewhere. (8,13-17)

In this analysis, spectra from 2DB calculations were used to calculate expected reaction rates. These reaction rate values were then used with empirically derived spectral inputs to see if the original spectra would be reproduced by the SAND-II procedure. How well the SAND-II derived spectra agreed with the original spectra determined the degree of definition (uniqueness) possible with the reactions used.

The degree of divergence between the SAND-II solutions and the original spectra depends upon the form of the input. For this study, the inputs were physically reasonable forms determined in the following empirical manner. For each spectrum, a fission spectrum form above 3 MeV, a constant from 3 MeV down to $\sim 10^{-4}$ MeV, and a form proportional to energy below $\sim 10^{-4}$ MeV was corrected using the $^{197}\text{Au}(n,\gamma)$ resonance, the $^{59}\text{Co}(n,\gamma)$ resonance, and the threshold monitors. The threshold monitors determined the magnitude above ~ 1 MeV, while the gold and cobalt resonances provided the magnitude below this energy. An E form was extrapolated below the gold resonance and straight line log-log interpolations joined the gold and cobalt resonances and the cobalt resonance and 1 MeV level to provide a physically reasonable input form.*

* For actual tests, this same procedure is recommended to define an input spectral form from a set of measured reaction rates if a form based on reactor physics calculations is not available.

The degree of solution definition was analyzed for four locations, Row 2 and Row 7 at midplane and at 59.2 cm above midplane. The ratio of the SAND-II solution integral flux divided by the integral flux for the starting 2DB spectrum versus neutron energy is shown in Figure 8. A constant ratio of 1 implies exact agreement between the unfolded and starting 2DB spectrum.

Figure 8 shows that the maximum differences in $\Phi(>E)$ along the reactor midplane ($z = 0$) and the top bounding position ($z = 59.2$ cm) are $\sim 5\%$ and $\sim 25\%$, respectively. Since there were no errors in the reaction rates or cross sections used in this analytical case, these uncertainties result from the lack of solution uniqueness associated with the use of a finite set of 14 foil reactions. The ratios in Figure 8 also indicate that the greatest uniqueness uncertainty exists in the energy range from ~ 0.05 to ~ 1 MeV. This is because the present monitors lack sufficient sensitivity to neutrons in this energy range.

2. Two Foil Reaction Rate Ratio Method

The expression used to calculate flux from the measured reaction rate of a single neutron flux monitor is

$$\Phi = A/\bar{\sigma}, \quad (3)$$

where

Φ = total value of integral neutron flux ($E > 10^{-10}$ MeV),

A = measured reaction rate (saturated activity or fission rate) per target atom

$\bar{\sigma}$ = spectral-averaged cross section,

$$\bar{\sigma} = \frac{\int_0^{\infty} \phi(E) \sigma(E) dE}{\int_0^{\infty} \phi(E) dE} \quad (4)$$

where $\phi(E)$ is the differential flux and $\sigma(E)$ is the energy-dependent cross section.

If Φ and A are measured," Equation (3) can be used to obtain a measured value, $\bar{\sigma}_m$, of $\bar{\sigma}$. If $\phi(E)$ and $a(E)$ are known, Equation (4) yields a calculated value, $\bar{\sigma}_c$, of $\bar{\sigma}$. The accuracy of $\bar{\sigma}_c$ depends upon the accuracy of differential quantities $\phi(E)$ and $\sigma(E)$ while the accuracy of $\bar{\sigma}_m$ depends only upon the accuracy of their integrals; i.e., the measured reaction rate and total integral flux. Measured and calculated values of $\bar{\sigma}$ are considered for single monitors in Section V-D.

This and previous^(9,20) studies have established that local perturbation effects between fueled and structural sub-assemblies can adversely effect the interpretation of single foil flux monitors. (See Section V-B-2.) At a very minimum, therefore, a monitor reaction sensitive to all neutrons and another sensitive to high energy neutrons are needed to measure integral flux-spectra in fast reactors.

Dudey et al.^(3,10) suggested using the $^{54}\text{Fe}(n,p)/^{55}\text{Fe}(n,\gamma)$ reaction rate ratio (as well as others) as a spectral hardness and flux level indicator for EBR-II spectra. To accomplish this, the authors related the measured $^{54}\text{Fe}/^{55}\text{Fe}$ reaction rate ratio to the median energy E_m and a value of $\bar{\sigma}$ for different EBR-II spectra. The median energies were obtained from reactor physics calculations, while the $\bar{\sigma}$ values were either calculated or measured.

In Section V-E of this report, results from the 31F dosimetry test have been analyzed in order to determine the uncertainty associated with the use of an empirically established relationship between values of $\bar{\sigma}$, E_m , $\bar{E}$ (the mean neutron energy), and the ratio of two measured reaction rates for the reaction ratios $^{54}\text{Fe}(n,p)/^{58}\text{Fe}(n,\gamma)$ and $^{238}\text{U}(n,f)/^{235}\text{U}(n,f)$. The mean neutron energy $\bar{E}$ is defined by

* Φ is not a directly measured quantity; however, a reasonable estimate of Φ is obtained by multiple foil measurements.

$$\bar{E} \equiv \int_0^{\infty} E \phi(E) dE \bigg/ \int_0^{\infty} \phi(E) dE, \quad (5)$$

and the median energy E_m is defined by

$$\int_{E_m}^{\infty} \phi(E) dE \bigg/ \int_0^{\infty} \phi(E) dE = 1/2. \quad (6)$$

3. Reactor Physics Calculations

Reactor physics calculations of the integral flux distribution in EBR-II for core 31F were made using the diffusion code, 2DB.⁽²⁹⁾ The code 2DB is a two-dimensional diffusion-burnup code designed primarily for fast reactor core analysis. Flux-spectra results from such a core analysis are not intended to be valid for out-of-core regions. The comparison of multiple foil derived flux-spectra with those by the 2DB diffusion code is given in Section V-B-3.

The normalized results of DTF-IV transport,⁽²⁸⁾ 2DB diffusion,⁽²⁹⁾ and Monte Carlo⁽²⁰⁾ calculations of the spectral form in Row 2 - Midplane are presented in Table 2. Below 10^{-2} MeV, the transport shape has been used to extend the Monte Carlo result. For the specified energies, the three reactor physics calculations agree to within $\sim 30\%$. The differences are a result of different assumptions used in the calculations and modeling of the core and blanket regions, variations in group structure and cross sections, and differences in EBR-II cores (only the diffusion code result is for core 31E-F). All three of these spectra were used as input forms for SAND-II analysis.

TABLE 2. Integral Flux Comparison Above Specified Neutron Energies

Neutron Energy, MeV	Normalized Integral Flux Above Specified Energy, n/cm ² -sec		
	Diffusion	Transport	Monte Carlo
10 ⁻³	1.000*	1.000*	1.000*
10 ⁻²	9.982 × 10 ⁻¹	9.978 × 10 ⁻¹	9.935 × 10 ⁻¹
10 ⁻¹	8.789 × 10 ⁻¹	8.945 × 10 ⁻¹	8.709 × 10 ⁻¹
0.5	4.579 × 10 ⁻¹	4.916 × 10 ⁻¹	4.482 × 10 ⁻¹
1.0	2.536 × 10 ⁻¹	2.802 × 10 ⁻¹	2.465 × 10 ⁻¹
1.5	1.651 × 10 ⁻¹	1.762 × 10 ⁻¹	1.567 × 10 ⁻¹
2.0	1.157 × 10 ⁻¹	1.215 × 10 ⁻¹	1.053 × 10 ⁻¹
2.5	8.118 × 10 ⁻²	8.433 × 10 ⁻²	7.436 × 10 ⁻²
3.0	5.775 × 10 ⁻²	6.000 × 10 ⁻²	5.139 × 10 ⁻²
3.5	4.010 × 10 ⁻²	4.196 × 10 ⁻²	3.526 × 10 ⁻²
4.0	2.782 × 10 ⁻²	2.934 × 10 ⁻²	2.434 × 10 ⁻²
5.0	1.409 × 10 ⁻²	1.490 × 10 ⁻²	1.146 × 10 ⁻²
6.0	6.660 × 10 ⁻³	7.034 × 10 ⁻³	5.986 × 10 ⁻³
7.0	3.190 × 10 ⁻³	3.365 × 10 ⁻³	3.278 × 10 ⁻³
8.0	1.555 × 10 ⁻³	1.641 × 10 ⁻³	1.639 × 10 ⁻³
9.0	7.532 × 10 ⁻⁴	7.946 × 10 ⁻⁴	8.127 × 10 ⁻⁴
10.0	0.00**	0.00**	0.00**

* The flux below 10⁻³ MeV is negligible.

** The flux above 10 MeV is negligible.

V. RESULTS

A. SAND-II MULTIPLE FOIL DERIVED INTEGRAL FLUX SPECTRA

Integral fluxes $\Phi(>E)$ were determined by the SAND-II code using 2DB diffusion code calculated spectra and 14 reaction rates as input. The reaction rates were taken from a r,z matrix derived from run 31F, see Section V-C-1. The multiple foil derived axial integral flux $\Phi(>E)$ distributions for four radial positions are shown for seven energies "E" in Figures 9 to 12.* These results can be used to define the approximate flux level at different r,z locations for interpreting and correlating irradiation effects data and for planning tests.

For the integral flux values given in Figures 9 to 12, the estimated maximum error range was determined for two positions each in Row 2 and Row 7 using Equation (2). These estimates are given in Table 3.

B. INTEGRAL FLUX COMPARISONS

1. High Power Flux Level Differences

The value of the central core (Row 2, midplane) flux obtained from this test is not in agreement with the flux as inferred from total reactor fission power. With an effective ^{235}U fission cross section of 1.4 ± 0.03 barns at the reactor center, an effective total neutron flux of 2.5×10^{15} n/cm²-sec at 50 MW is inferred from total reactor fission power.⁽¹⁾ Using a value of 200 MeV/fission as reported by Kirn,⁽¹²⁾ a 2DB reactor physics calculation performed at PNL yielded a value of

* *Tabulated and/or plotted integral flux values based on the 31F test for these and other r,z coordinates are available to EBR-II experimenters through the Fast Reactor Dosimetry Center at PNL.*

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FIGURE 9. EBR-II Run 31F - Radius = 5.2 cm,
Integral Flux **Above** Specified
Energies

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FIGURE 10. EBR-II Run 31F - Radius = 13.1 cm
Integral Flux Above Specified
Energies

Neg 701770-8

FIGURE 11. EBR-II Run 31F - Radius = 24.7 cm
Integral Flux Above Specified
Energies

Neg 701770-9

FIGURE 12. EBR-II Run 31F - Radius = 33.6 cm
Integral Flux Above Specified
Energies

TABLE. 3. Estimated Maximum Percent Errors
(One Standard Deviation). for SAND-II
Derived Integral Fluxes^(a) $>E_L$

E_L , MeV	r , cm			
	5.2	5.2	33.6	33.6
	z , cm			
	0.0	59.2	0.0	59.2
1.0×10^{-10}	12	17	12	14
1.0×10^{-2}	12	15	13	13
1.0×10^{-1}	14	16	14	22
4.0×10^{-1}	12	14	13	15
1.0	12	14	12	13
3.0	12	13	12	13
6.0	12	12	12	13

a. See Section IV-C-I-b

total flux of 2.38×10^{15} n/cm²-sec.⁽²⁹⁾ The value obtained from multiple foil measurements at the edge of the structural subassembly for Run 31F is 2.06×10^{15} n/cm²-sec.

Based on reactor power considerations, Kirn provided a value of the fission rate for Row 2 midplane for driver fuel for Run 31F of 0.88×10^{13} f/g ²³⁵U-sec, or 3.44×10^{-9} fission/nuclei-sec.^{(12)*} The value of 2.90×10^{-9} fissions per nuclei-sec obtained at the edge of the structural subassembly in the 31F dosimetry test is ~16% less. Measured ¹³⁷Cs saturated activity from a driver fuel subassembly showed about the same difference relative to that obtained at the edge of the structural assembly in the 31F dosimetry test. Therefore, there

* Kirn has reported a central value of 0.90×10^{13} fissions/g-sec at 50 MW for Run 39, Reference 1.

is also $\sim 16\%$ difference between the fission rate obtained from the dosimetry test and the fission rate derived from a driver fuel. (11,30)

The results of an analytical study performed by Miller and Jarka⁽⁹⁾ indicate no more than an $\sim 4\%$ depression in ^{235}U fission rate between a fuel and structural subassembly. Thus, there should be no more than $\sim 4\%$ depression in calculated flux between a fuel and structural subassembly since the ^{235}U fission cross section should not change significantly between them in the core.

Consequently, an increase in the value of 2.06×10^{15} by $\sim 4\%$ to compensate for the difference between a fueled and a structural subassembly, leaves a difference of ~ 10 to 15% between the predicted (based on reactor physics calculations, power considerations, and burnup measurements) and measured flux level at Row 2 midplane for the 31F dosimetry test. The 10 to 15% is within the probable accuracy limits for the monitoring and prediction methods, although portions may represent systematic errors in the methods, or a real difference between the local flux at the experiment and the non-perturbed flux elsewhere.

2. Structural Subassembly Perturbations

Subassembly X046 (see Figures 4 and 5) contained two nuclear duplicate dosimetry pins, the other 17 pins were solid stainless steel rods. Pin 18 was near the edge of the subassembly at 5.2 cm from the core centerline and Pin 10 was in the center of the subassembly at 5.9 cm.

Since both pins were irradiated simultaneously, variations from unity in the ratios of individual monitor reaction rates should indicate changes in flux level and spectrum. Table 4 gives the different monitors, their 90% response range, reaction rates, and ratios of the Pin 18/Pin 10 values.

TABLE 4. Structural Subassembly Perturbations
at Reaction Midplane - Row 2

Reaction	90% Response Range, MeV		Reaction Rate Ratio, Pin 18/Pin 10	Mean Ratio	Estimated Standard Error, 2 a
	E _L	E _u			
$^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$	0.008	1.5	0.98	1.00	4%
$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	0.014	1.0	1.01		
$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	0.017	0.96	0.94		
$^{23}\text{Na}(n,\gamma)^{24}\text{Na}$	0.026	1.5	0.99		
$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	0.038	1.6	1.00		
$^{235}\text{U}(n,f)^{140}\text{Ba}$	0.036	2.7	1.08	1.06	3%
$^{237}\text{Np}(n,f)^{140}\text{Ba}$	0.048	4.2	1.02		
$^{238}\text{U}(n,f)^{140}\text{Ba}$	1.4	6.1	1.02		
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	1.9	7.3	1.09		
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	2.0	7.8	1.08		
$^{47}\text{Ti}(n,p)^{47}\text{Sc}$	2.0	6.7	1.01	1.08	7%
$^{46}\text{Ti}(n,p)^{46}\text{Sc}$	3.3	9.4	1.05		
$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$	6.1	11.8	1.14		
$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	6.4	12.5	1.08		
$^{48}\text{Ti}(n,p)^{48}\text{Sc}$	6.7	13.6	1.01		

The entire set is divided into three groups* whose response energies are less than ~ 2 MeV, greater than ~ 2 MeV, and greater than ~ 6 MeV. The mean ratio for each group and the standard error of the mean, Table 4, show that the lower energy parts of the spectra at the two pins are about the same. For the higher energy groups, there is less activation in Pin 10 than in Pin 18; therefore, there were fewer high energy neutrons at the center of a structural subassembly than at the edge. This is further substantiated by similar results obtained at 15.2 cm above midplane (near the core top) where the mean ratios

* This grouping is consistent with the manner in which the Sand II code averages (or weights) the results of foils with similar energy response.

for the three groups with increasing energy are 0.97, 1.06, and 1.06 with 2σ standard errors of 4%, 3%, and 1%.

These experimental results are consistent with the analytical results of Miller and Jarka^(9,10) who find that the spectrum in a structural subassembly softens significantly above 2.23 MeV and that the flux depression may be as high as 25% from the center of a driver fuel subassembly to the center of a structural subassembly.

Using the reactions listed in Table 4, the integral flux above different lower energy limits has been determined for both Pins 18 and 10 and is given in Table 5. The ratios of integral fluxes from 10^{-2} MeV to 2 MeV show changes of less than $\sim 3\%$. However, for integral fluxes $E > 2.2$ MeV to > 6.0 MeV, the ratio changes from 1.04 to 1.12 with an estimated 2σ uncertainty of 5 to 7%, respectively. In terms of measured spectral averaged cross sections for $^{54}\text{Fe}(n,p)$ and $^{58}\text{Ni}(n,p)$, the edge-to-center flux-spectral change is reflected as an $\sim 6\%$ decrease for both reactions."

These indicated high energy changes must be included in the interpretation of high energy flux monitors such as iron and nickel which are commonly used by fuels and materials experimenters. Neglecting spectral differences between fueled and structural subassemblies could introduce errors of up to $\sim 25\%$ in total flux and fluence estimates based on these monitors.

* Analytical results based on a Monte Carlo calculation reported by Kamphouse et al.⁽²⁰⁾ indicate a decrease of ~ 5 to 6% in the spectral averaged cross sections for the $^{54}\text{Fe}(n,p)$ and $^{58}\text{Ni}(n,p)$ reactions, respectively, from the edge to center of a three region partially fueled subassembly in Row 2.

TABLE 5. Integral Flux Comparisons of Pins 18 and 10

Neutron Energy, E, MeV	Integral flux >E(a)		Ratio Pin 18/Pin 10
	Pin 18	Pin 10	
$>1 \times 10^{-10}$	2.068×10^{15}	2.059×10^{15}	1.00
$>1 \times 10^{-2}$	2.063×10^{15}	2.055×10^{15}	1.00
$>1 \times 10^{-1}$	1.817×10^{15}	1.801×10^{15}	1.01
>1.0	5.144×10^{14}	5.001×10^{14}	1.03
>2.0	2.103×10^{14}	2.035×10^{14}	1.03
>2.2	1.783×10^{14}	1.718×10^{14}	1.04
>2.3	1.645×10^{14}	1.581×10^{14}	1.04
>3.0	9.706×10^{13}	9.158×10^{13}	1.06
>5.0	2.588×10^{13}	2.333×10^{13}	1.11
>6.0	1.345×10^{13}	1.198×10^{13}	1.12

a. *Four significant figures are retained for purposes of comparison only.*

3. Multiple Foil and Reactor Physics Calculations

The comparison of multiple foil derived energy-dependent integral fluxes with those obtained from different calculations in core center and the PNL diffusion code, 2DB, calculations in- and out-of-core will now be considered. Combinations of up to 17 foil reaction rates,^(4,5) and the different input spectra listed in Table 2 were used to generate flux-spectral solutions for the Row 2 - midplane location in the structural subassembly.

Differences in integral flux solutions (indicated by the standard deviations, 7th row of Table 6) based on the most reasonable input spectra (from diffusion, transport, and Monte Carlo code calculations, see Table 2) define approximate limits for the shape of the true solution. The solution based on a constant spectrum input gives a further estimate of the

TABLE 6. Integral Flux Comparison - EBR-II Row 2 Midplane(a)

Input Spectral Approximation Based on	SAND-II Solution		Integral Flux Values $\Phi(E)$ (n/cm ² -sec) Above Specified Energies (MeV) $\times 10^{-15}$					
	Solution (b)		$10^{-3}(c)$	0.1	0.5	1.0	2.0	3.0
	K	SPD						
Monte Carlo Code (MC) (17 Foils)	26	3.1	2.01	1.78	0.933	0.535	0.232	0.105
Diffusion Code (DC) (17 Foils)	26	3.1	2.07	1.81	0.916	0.531	0.236	0.104
Transport Code (TC) (17 Foils)	28	3.0	2.06	1.82	0.915	0.538	0.236	0.103
Transport Code (TC) (4 Fission Foils)	12	2.9	2.11	1.85	0.909	0.528	0.237	0.119 ^(d)
Transport Code (TC) (13 Nonfission Foils)	50	2.0	2.07	1.83	0.955	0.566	0.238	0.103
Average Values (AV)			2.06	1.82	0.926	0.540	0.236	0.104
Standard Deviation %			± 1.7	± 1.4	± 2.0	± 2.8	20.97	20.92
constant Input ^(e) (CI) (17 Foils)	33	2.6	2.01	1.78	0.948	0.524	0.229	0.101
Absolute Deviation ^(f) Between (AV) and (CI) %			2.4	2.2	2.4	3.0	3.0	2.9
$\frac{\text{Measured (AV)}}{\text{Predicted}}$			0.87	0.87	0.85	0.89	0.86	0.76

a. Determined for a stated power of 50 MW, Reference 12.

b. The convergence criterion used was that the change between solution standard percent deviation (SPD) between successive iterations had stabilized to less than 1%, see Reference 16. The SPD is defined as:

$$SPD^{(K)} = \left[\frac{1}{n-1} \sum_{i=1}^n \left(\frac{A_i^{(K)} - A_i^m}{A_i^{(K)}} \right)^2 \right]^{1/2}$$

where "n" is the number of foil reactions, A_i^m is the measured reaction rate for the i^{th} foil, and $A_i^{(K)}$ is a calculated value using the solution differential flux at the K^{th} cycle of iteration and the energy-dependent cross section for the i^{th} foil. In general, iteration beyond a stability limit of 1% produces little change in integral flux.

c. The integral flux below 10^{-3} MeV is negligible compared to that above this energy.

d. This value was not used in obtaining the average value because of the lack of high energy foil coverage.

e. Constant above 10^{-2} MeV, with a form proportional to energy below this energy, since there are few neutrons below 10^{-2} MeV.

f. Defined as $100[|\Phi(E) - \bar{\Phi}(E)| / \bar{\Phi}(E)]$, where $\bar{\Phi}(E)$ is the average value (AV).

accuracy of this shape (indicated by the absolute deviations, 9th row of Table 6). These results indicate that the integral flux solution is largely independent of the input spectral form for the Row 2 - midplane position.

The present study (also see Sections IV-C-1-b and V-A) suggests an absolute uncertainty estimate of $\sim \pm 10\%$ (1 standard deviation) for the measured average integral flux values (AV), Table 6. These values are compared in the last row of Table 6 with theoretical predictions based on the 2DB diffusion code results,⁽²⁹⁾ with an energy release value of 200 MeV/fission as suggested by Kirn.⁽¹²⁾ The measured and predicted values for energies less than 2 MeV are found to agree within $\sim 15\%$. The larger difference above 3 MeV is presumed to reflect the additional effect of a high energy flux depression in the structural subassembly (see Section V-B-2).

In Figure 13, the ratios of absolute integral fluxes (SAND-II/2DB) at the four bounding locations are shown for energies from 10^{-5} MeV up to 10 MeV. The ratios remain constant below 10^{-5} MeV and hence are not shown.

The two midplane ratios indicate that the 2DB calculation and multiple foil derived total integral fluxes agree within 15%, and the integral flux $\Phi(>E)$ differ by no more than 35%. Outside of the core region, in the upper blanket where the 2DB calculation was not intended to be valid, the total integral fluxes differ by as much as $\sim 80\%$ and the integral flux $\Phi(>E)$ differ by as much as a factor of 5. Except for the midplane position integral fluxes below about 2 or 3 MeV, the 2DB calculated values of integral fluxes lie outside the maximum estimated 1 σ accuracy limits of the SAND-II results.

The shape of the ratio curves for the top of Rows 2 and 7 position have three common characteristics; approximately constant below 0.5 MeV, a peak between 1 and 2 MeV and a rise

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FIGURE 13. Comparison of SAND-II and 2DB Absolute Integral Fluxes for Bounding r,z Locations

above 7 MeV. The fact that the ratio is much greater than one above 0.5 MeV at the two out-of-core locations indicates that the measured spectra are significantly harder than the 2DB calculated spectra. In core, the decrease in the ratio above ~2 MeV indicates that the measured spectrum is somewhat softer (a result that is consistent with an expected flux depression in the high energy flux for a structural subassembly located next to fuel in the core).

These results emphasize that core calculations are not intended for use outside the core. For such regions, experimenters must select an adequate set of monitors to cover the spectrum if they are to secure reliable flux-fluence values. This conclusion is particularly relevant to the large number of structural material experiments in Rows 6 and 7 of EBR-II.

4. High- and Low-Power Flux Spectral Comparisons

a. Reactor Runs 31E and 31F

During the low-power 31E and high-power 31F dosimetry tests in EBR-II, two 19-pin subassemblies, X045 and X046, respectively, were irradiated in Reactor Position 2B1. Each 19-pin subassembly contained nuclear duplicate dosimetry pins in Pin Position 18. Each dosimetry pin contained an identical set of neutron monitors in five precisely located axial positions.

The ratio of the low- to high-power reaction rates of the individual monitors should be the same as the ratio of reactor powers if the low- and high-power spectra were identical. For the position $z = 0$, Table 7 lists the 14 reaction rates at high and low power and their ratio with estimated uncertainties. For the stated reactor powers, 84 kW and 50 MW, the ratio would be 1.68×10^{-3} . Except for two cases, the ratios range between

TABLE 7. Comparison of Low- and High-Power Reaction Rates^(a)
(Row 2, R = 5.2 cm, Z = 0.0 cm)

Reaction	Reaction Rate $\pm$ (1 σ % Std. Error)		Ratio: (Low/High) $\times 10^5$ (2 σ % Std. Error)
	Low Power	High Power	
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	$6.42 \times 10^{-14} \pm 1.7$	$3.77 \times 10^{-11} \pm 3.3$	1.70 ± 7.4
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	$8.80 \times 10^{-14} \pm 1.6$	$5.14 \times 10^{-11} \pm 1.0$	1.71 ± 3.8
$^{46}\text{Ti}(n,p)^{46}\text{Sc}$	$8.36 \times 10^{-15} \pm 4.6$	$4.80 \times 10^{-12} \pm 1.1$	1.74 ± 9.4
$^{47}\text{Ti}(n,p)^{47}\text{Sc}$	$1.59 \times 10^{-14} \pm 1.0$	$8.36 \times 10^{-12} \pm 3.0$	1.90 ± 6.4
$^{48}\text{Ti}(n,p)^{48}\text{Sc}$	$2.13 \times 10^{-16} \pm 3.2$	$1.22 \times 10^{-13} \pm 1.5$	1.75 ± 7.0
$^{235}\text{U}(n,f)^{140}\text{Ba}$	$5.28 \times 10^{-12} \pm 1.8$	$2.85 \times 10^{-9} \pm 2.6$	1.85 ± 6.4
$^{238}\text{U}(n,f)^{140}\text{Ba}$	$3.24 \times 10^{-13} \pm 0.6$	$1.81 \times 10^{-10} \pm 2.8$	1.79 ± 5.8
$^{237}\text{Np}(n,f)^{140}\text{Ba}$	$2.26 \times 10^{-12} \pm 2.5$	$1.26 \times 10^{-9} \pm 0.9$	1.79 ± 5.4
$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	$1.66 \times 10^{-14} \pm 1.6$	$8.75 \times 10^{-12} \pm 2.3$	1.90 ± 5.6
$^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$	$7.93 \times 10^{-14} \pm 6.3$	$4.47 \times 10^{-11} \pm 5.9$	1.77 ± 17.0
$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	$1.35 \times 10^{-12} \text{ (b)}$	$4.68 \times 10^{-10} \pm 6.3 \text{ (c)}$	2.88 ± 19.0
$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	$4.85 \times 10^{-16} \pm \text{(b)}$	$2.75 \times 10^{-13} \pm \text{(b)}$	1.76 ± 20.0
$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	$8.01 \times 10^{-14} \pm \text{(b)}$	$3.14 \times 10^{-11} \pm 3.9 \text{ (c)}$	2.55 ± 16.0
$^{23}\text{Na}(n,\gamma)^{24}\text{Na}$	$1.94 \times 10^{-15} \pm \text{(b)}$	$1.08 \times 10^{-12} \pm \text{(b)}$	1.80 ± 20.0

a. *Reactions/sec-nuclei*

b. *Only one monitor was available and no standard error could be calculated. Based on the precision obtained for the other reactions and the consistency in the data as compared with other axial and radial positions, the 1 σ standard deviation is estimated to be less than 7%.*

c. *Based on results from only two laboratories.*

1.70 and 1.90×10^{-3} . For the $^{197}\text{Au}(n,\gamma)$ and the $^{59}\text{Co}(n,\gamma)$ reactions, the ratios are significantly higher than 1.68×10^{-3} . Similar results were obtained at the other four axial positions.

These ratios lead to the conclusion of a definite difference in spectra between the high and low power tests. At the first axial position ($z = 0$ cm), SAND-II analysis of the Table 7 reaction rate data showed that there were fewer low-energy neutrons (below 10^{-2} MeV) at high power than at low power; i.e. 0.2% versus 1%. The 14 reactions shown in Table 7 yielded the integral flux solutions used to generate the ratios presented in Figure 14 for all five axial positions.*

Except for results at core center midplane ($z = 0$ cm), all other results show a definite downward trend for all energies. Therefore, there were fewer high-energy neutrons in the low-power spectrum as compared to the high-power spectrum. Why these differences occur is not understood. Until they are understood, the procedure of extrapolating low-power dosimetry data for high-power materials irradiation tests must be considered uncertain.

b. Comparison of Previous and Recent Dosimetry Tests

A previous report⁽²⁾ stated that the measured high-power flux in EBR-II was ~60% of the high-power flux estimated from an extrapolated low-power measurement. In addition, although not discussed in the reference, iron and nickel based high-power flux measurements were ~60% of a flux value based on reactor physics, power level and burnup analyses. An important purpose of the 31E and 31F dosimetry tests was to resolve these differences.

* *The reaction rate results obtained by one laboratory were used rather than the mean values given in Table 7, since only relative changes in reaction rate are of interest here. The errors associated with the relative measured reaction rates of one laboratory are considered smaller than those associated with the mean values given in Table 7.*

FIGURE 14. Comparison of Low- and High-Power Multiple Foil Derived Integral Fluxes at 5 Axial Positions, Row 2

Table 8 compares the previous data from the low-power physics test 25a and the high-power (45 MW) test with the present 31E and 31F tests. All data were taken from monitors located near the edge of the subassemblies. The reported total flux/MW shows significant difference between the previous and present results.

Based on the current set of BNW cross sections the data were re-evaluated. This resolved most of the difference between the old high-power results and the 31E and 31F results, but the 25a test results remained excessively high.

Two additional factors were considered. The 25a test was run in a modified driver subassembly in an 88-element core. The closeness to fuel and the smaller core would produce an estimated flux no more than ~5% higher than 31E and 31F. The old high-power data were run in an 80-element core and the flux would be ~5% higher compared to a 91-element core. Applying these corrections, the old high-power, 31E and 31F results agree even more closely, but the 25a result is still high.

Full power reactor power estimates are considered to be accurate to within $\pm 5\%$.⁽¹²⁾ The power estimates for the low-power tests, however, were taken from nuclear instrumentation and are more uncertain. If total flux is proportional to reactor power, the power estimate for 25a should be adjusted to 26 kW. This adjustment will resolve all previous differences.

Such an adjustment would not be valid if the spectrum in the 25a test were quite different from the spectrum in the 45 MW, the 31E and 31F tests. Normalized fluxes are given in Table 9. The major differences, as evidenced by the ratio

TABLE 8. Comparison of Previous ^(a) and Recent Dosimetry Data

Test	Previous Data		Present Data	
	25a	Old High-Power	31E	31F
Stated Power	20 kW	45 MW	84 kW	50 MW
Radial Blanket
Subassembly Type	Modified Fuel	Structural	Structural	Structural
Core Size-Elements	88	~80	91	91
Reported Flux (Total)/MW n/cm ² -sec-MW	5.06×10^{13}	3.33×10^{13}	4.20×10^{13}	4.12×10^{13}
Flux/MW Based on Current Cross Sections	$5.70 \times 10^{13(a)}$	$4.44 \times 10^{13(b)}$	$4.20 \times 10^{13(a)}$	$4.12 \times 10^{13(a)}$
Flux/MW Corrected for Subassembly and Core Size to 91 Elements	5.41×10^{13}	4.22×10^{13}	4.20×10^{13}	4.12×10^{13}
Total Integral Flux Level	1.08×10^{12}	1.90×10^{15}	3.53×10^{12}	2.06×10^{15}
Corrected Power Based on Both High Power and 31E	26 kW			
Flux/MW Corrected for All Factors	4.16×10^{13}	4.22×10^{13}	4.20×10^{13}	4.12×10^{13}

a. Based on multiple foil analysis

b. Based on $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ reaction.

of 25a/31F, are only in the high-energy part of the spectra, $E > 2$ MeV, which accounts for $\sim 10\%$ of the total integral flux. Such differences would not invalidate the adjustment based on relative activations.

TABLE 9. Comparison of Runs 25a and 31F Normalized Integral Fluxes Above Specified Energies

<u>EL, MeV</u>	<u>Integral Flux Above(a)</u> <u>Neutron Energy, EL</u>		<u>Ratio</u> <u>25a/31F</u>
	<u>25a</u>	<u>31F</u>	
$>1.0 \times 10^{-10}$	1.00000	1.00000	1.00
$>1.0 \times 10^{-4}$	1.00000	1.00000	1.00
$>1.0 \times 10^{-3}$	0.99998	0.99999	1.00
$>1.0 \times 10^{-2}$	0.99827	0.99791	1.00
$>1.0 \times 10^{-1}$	0.89370	0.87908	1.02
>1.0	0.26166	0.25152	1.04
>2.0	0.12023	0.10335	1.16
>3.0	0.05959	0.04770	1.25
>4.0	0.02775	0.02215	1.25
>5.0	0.01349	0.01103	1.22
>6.0	0.00591	0.00518	1.14

a. *Five significant figures are retained for purposes of comparison only.*

Low-power flux/MW measurements may be more uncertain than those for a high-power test. Extrapolation of low-power flux results to predict full power flux, therefore, must be done with due caution.

C. MEASURED REACTION RATES

1. Radial and Axial Distributions

During the 31F reactor run, monitor sets were located axially at reactor midplane and 6, 8, 15, and 25 in. above midplane in Rows 2, 4, 6, and 7. Rows 2, 4, and 6 are within

the reactor core, and Row 7 is in the inner blanket (depleted uranium) region. Activation data^(4,5) from the monitors have been incorporated into a computer library from which the saturated reaction rate and/or integral flux can be determined for any r,z position in the EBR-II core and blanket out to 38 cm radius from core centerline and up to 76 cm axial distance above midplane. The reaction rates derived from the matrix have an estimated uncertainty of approximately $\pm 10\%$ (1a) at all locations.

Measured reaction rates for each of the four rows as a function of axial position for a reactor power of 50 MW for 14 foil reactions are given in Figures 15 to 18. Measured values can be read to within $\sim 10\%$ from the plots. These data can be used for estimating activation levels for individual flux monitors in materials tests or the fission rates in fuel tests.

2. Reaction Rate Ratios

Relative changes in the ratio of reaction rates for different monitors indicate spectral changes. The quantitative effect of such changes is considered in Section V-E. Three ratios of reaction rates are plotted in Figures 19 and 20 for axial and radial measurements, respectively. The upper curve, in both figures, is the ratio for $^{235}\text{U}(n,f)$ ^{140}Ba to $^{54}\text{Fe}(n,p)$ ^{54}Mn . Approximately 90% of response of the $^{235}\text{U}(n,f)$ reaction in the core center spectrum lies between 0.038 and 3 MeV. The $^{54}\text{Fe}(n,p)$ ^{54}Mn response is between 2.1 and 7.9 MeV. The middle curve is the ratio $^{235}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$, the ^{238}U response is between 1.4 and 6.4 MeV. The bottom curve is the ratio of $^{54}\text{Fe}(n,p)$ ^{54}Mn to $^{58}\text{Fe}(n,\gamma)$ ^{59}Fe , the $^{58}\text{Fe}(n,\gamma)$ ^{59}Fe response is between 0.003 and 1.2 MeV. Where the ratios are constant, the neutron spectrum does not change.

Neg 701770-4

FIGURE 15a. Reaction Rates at 50 MW
Row 2 (R = 5.2 cm)

Neg 701867-6

FIGURE 15b. Reaction Rates at 50 MW
Row 2 (R = 5.2 cm)

Neg 701867-6

FIGURE 15c. Reaction Rates at 50 MW
Row 2 (R = 5.2 cm)

Neg 701770-5

FIGURE 16a. Reaction Rates at 50 MW
Row 4 (R = 13.1 cm)

Neg 701770-2

FIGURE 16b. Reaction Rates at 50 MW
Row 4 (R = 13.1 cm)

Neg 701867-5

FIGURE 16c. Reaction Rates at 50 MW
Row 4 (R = 13.1 cm)

Neg 701867-4

FIGURE 17a. Reaction Rates at 50 MW
Row 6 (R = 24.7 cm)

Neg 701867-3

FIGURE 17b. Reaction Rates at 50 MW
Row 6 (R = 24.7 cm)

Neg 701867-2

FIGURE 17c. Reaction Rates at 50 MW
Row 6 (R = 24.7 cm)

Neg 701867-1

FIGURE 18a. Reaction Rates at 50 MW
Row 7 (R = 33.6 cm)

Neg 701770-1

FIGURE 18b. Reaction Rates at 50 MW
Row 7 (R = 33.6 cm)

Neg 701867-7

FIGURE 18c. Reaction Rates at 50 MW
Row 7 (R = 33.6 cm)

Neg 0691743

FIGURE 19. Reaction Rate Ratios - Axial,
Normalized to 1.0 at Row 2
Midplane

Neg 0691617

FIGURE 20.

Reaction Rate Ratios - Radial,
 Normalized to 1.0 at Row 2
 Midplane

Thus, it is inferred from Figure 19 that for the EBR-II Core 31F the spectrum remains constant for approximately 4 in. above midplane, and from Figure 20 the spectrum remains essentially unmodified out to approximately Row 4 on the reactor midplane.

D. MEASURED AND CALCULATED SPECTRAL AVERAGED CROSS SECTIONS

In those cases where the neutron spectrum is known, a single monitor can be used to determine the magnitude of the flux level according to $\Phi = A/\bar{\sigma}$ [Equation (3), Section IV-C-2]. The accuracy of the flux value obtained will be dependent upon the accuracy of the measured value of the reaction rate A and the spectral averaged cross section $\bar{\sigma}$. Procedures and uncertainties associated with the determination of appropriate values of $\bar{\sigma}$ are considered here and in Section V-E.

Results from this and a number of fast reactor tests indicate that the previously reported^(2,3) EBR-II core spectral-averaged cross section, $\bar{\sigma}_c$, for the $^{54}\text{Fe}(n,p)$ and $^{58}\text{Ni}(n,p)$ reactions should be reduced by about 25 and 30%, respectively, to obtain consistency with fission reactions.

The expression used to obtain measured values of spectral averaged cross section, $\bar{\sigma}_m$, is from Equation (3),

$$\bar{\sigma}_m = A/\Phi, \quad (7)$$

where again A and Φ^* are the measured reaction rate and total value of integral flux, respectively. For the EBR-II run 31F, Row 2 - midplane, dosimetry test, mean values of A with assigned error limits are given in Table 7. A value of $\Phi = 2.06 \times 10^{15}$ n/cm²-sec, Table 6, was used to obtain the measured values of $\bar{\sigma}_m$, Table 10. The absolute uncertainties associated with

* Φ is not a directly measured quantity; however, multiple foil measurements yield a reasonable estimate of Φ .

TABLE 10. Measured ($\bar{\sigma}_m$) and Calculated ($\bar{\sigma}_c$) Spectral Averaged Cross Sections for EBR-II Row 2 - Midplane

Reaction	$\bar{\sigma}_m$ (mb)	$\bar{\sigma}_c$ (mb)			Reference (2)	Maximum % Difference of $\bar{\sigma}_c$'s from $\bar{\sigma}_m$ (b)
		Transport ^(a) Code	Diffusion ^(a) Code	Monte Carlo ^(a)		
$^{235}\text{U}(n,f)$	1383	1345	1357	1383	1440	3
$^{238}\text{U}(n,f)$	87.9	97.2	90.7	86.1	98.2	11
$^{237}\text{Np}(n,f)$	612	678	628	615	-	11
$^{58}\text{Fe}(n,\gamma)$	4.25	4.22	4.28	4.42	-	4
$^{54}\text{Fe}(n,p)$	18.3 ^(c)	21.8	20.8	18.5	24.1	19
$^{58}\text{Ni}(n,p)$	25.0 ^(c)	29.8	28.4	25.6	35.4	19
$^{46}\text{Ti}(n,p)$	2.33	3.12	2.97	2.62	3.8	34
$^{47}\text{Ti}(n,p)$	4.06	5.16	4.92	4.48	-	27
$^{48}\text{Ti}(n,p)$	0.0592	0.0676	0.0641	0.0660	-	14
$^{26}\text{Al}(n,\alpha)$	0.133	0.185	0.178	0.178	0.173	39
$^{45}\text{Sc}(n,\gamma)$	15.0 ^(d)	14.5	15.3	16.7	12.2	3
$^{59}\text{Co}(n,\gamma)$	15.2	13.6	14.3	16.0	-	11
$^{63}\text{Cu}(n,\gamma)$	21.7	21.3	21.8	27.1	25.5	25
$^{197}\text{Au}(n,\gamma)$	227	218	229	251	-	11
$^{23}\text{Na}(n,\gamma)$	0.524	0.493	0.508	0.553	-	6
$^{232}\text{Th}(n,f)$	23.0 ^(d)	23.7	22.2	20.9	-	9
$^{238}\text{U}(n,\gamma)$	150 ^(d)	143	147	156	-	5

a. Based on the spectra given in Table 2, Section IV-C-3.

b. Maximum percent difference of Transport, Diffusion, or Monte Carlo results (Column 3 to 5) from the measured values, Column 2.

c. Using their revised $\sigma(E)$'s, Kamphouse et al. report values that vary from 20.4 to 19.0 for Fe and from 30.1 to 28.2 for Ni over a partially fueled structural subassembly nuclear mockup for Row 2, near midplane, see Section IV-B, and Reference 20.

d. The measured value of A for this reaction does not appear in Table 7 and was obtained from Reference 5.

these $\bar{\sigma}_m$'s are estimated to be $\pm 10\%$ to 15% (1 standard deviation); i.e., the combined estimated absolute uncertainties in the values of A_m and Φ . Relative uncertainties of ratios of one $\bar{\sigma}_m$ to another are much smaller.

The SAND-II readjusted energy-dependent cross sections (see Section IV-B), and three $\phi(E)$'s based on a Transport, a 2DB Diffusion, and a Monte Carlo code calculation (Table 2) were used with Equation (4) to obtain values of $\bar{\sigma}_c$, Table 10. Values for $\bar{\sigma}_c$ using previously reported, but unadjusted $\sigma(E)$'s,⁽²⁾ are given in the next to last column for a $\phi(E)$ based on a previous transport code calculation.^(2,28)

Comparison of $\bar{\sigma}_c$ values for Fe and Ni in the next to last column with the measured values $\bar{\sigma}_m$ in the second column provides the basis for the suggested 25 to 30% reductions. Based on current energy-dependent cross sections, the maximum percent difference between any of the $\bar{\sigma}_c$ and $\bar{\sigma}_m$ values is given in the last column of Table 10. This comparison shows that $\bar{\sigma}_c$ values are sensitive to spectral differences near the core center.

A comparison of $\bar{\sigma}_m$ and $\bar{\sigma}_c$ values at an out-of-core position in Row 7 of the blanket region ($r = 33.6$ cm, $z = 59.2$ cm) is given in Table 11. The values of A were obtained from the r, z matrix of 14 reaction rates stored on magnetic tape. For comparison, results are also given for a Row 2, core midplane position ($r = 5.2$ cm, $z = 0.0$ cm).^{*} The spectra $\phi(E)$ for both locations were obtained from the same 2DB diffusion code calculation.⁽²⁹⁾ The measured values of Φ were obtained in both cases from SAND-II iteration runs using 14 reactions (from the r, z matrix) with the 2DB spectra as input.

^{*} These values will differ somewhat from those given in Table 10 which are based on the mean values of A , Table 7.

TABLE 11. Comparison of Measured and Calculated Spectrum Averaged Cross Section for an EBR-II Core and Out-of-Core Position (cm²)

Reaction	r = 5.2 cm, z = 0.0 cm			r = 33.6 cm, z = 59.2 cm		
	Measured ^(a)	Calculated	Measured/ calculated	Measured	Calculated	Measured/ Calculated
⁵⁴ Fe(n,p) ⁵⁴ Mn	1.78-26	2.08-26	0.86	6.28-28	3.61-28	1.74
⁵⁸ Ni(n,p) ⁵⁸ Co	2.47-26	2.84-26	0.87	8.46-28	4.97-28	1.70
⁴⁶ Ti(n,p) ⁴⁶ Sc	2.35-27	2.97-27	0.79	9.18-29	5.06-29	1.81
⁴⁷ Ti(n,p) ⁴⁷ Sc	4.13-27	4.97-27	0.83	2.16-28	8.75-29	2.47
⁴⁸ Ti(n,p) ⁴⁸ Sc	5.73-29	6.41-29	0.89	2.37-30	1.16-30	2.04
²³⁵ U(n,f)	1.45-24	1.36-24	1.07	3.09-24	2.79-24	1.11
²³⁸ U(n,f)	8.69-26	9.07-26	0.96	5.19-27	1.88-27	2.76
²³⁷ Np(n,f)	6.21-25	6.28-25	0.99	8.81-26	7.92-26	1.11
⁵⁸ Fe(n,γ) ⁵⁹ Fe	4.07-27	4.28-27	0.95	1.74-26	1.84-26	0.95
⁶³ Cu(n,γ) ⁶⁴ Cu	2.30-26	2.18-26	1.06	1.12-25	1.13-25	0.99
¹⁹⁷ Au(n,γ) ¹⁹⁸ Au	2.13-25	2.29-25	0.93	2.76-24	2.13-24	1.30
²⁷ Al(n,α) ²⁴ Na	1.34-28	1.76-28	0.76	4.97-30	3.18-30	1.56
⁵⁹ Co(n,γ) ⁶⁰ Co	1.47-26	1.43-26	1.03	7.40-25	4.51-25	1.64
²³ Na(n,γ) ²⁴ Na	5.24-28	5.08-28	1.03	2.01-27	1.87-27	1.07

a. These values are slightly different from those given in Table 10 which are based on the mean values of "A", Table 7, because the r,z matrix of reaction rates required the fitting of up to 20 straight line segments through all the available data for each row from z = 0 up to z = 76 cm.

The ratio of the measured-to-calculated values, Columns 4 and 7 of Table 11, shows agreement within about 20% for the central core position, but very poor agreement for the blanket position [with a maximum difference of almost a factor of 3 for the $^{238}\text{U}(\text{n},\text{f})$ reaction]. Thus, unless the spectral form is well defined and local perturbation effects are considered, calculated spectral averaged cross sections should not be used to obtain values of total integral flux.

E. ESTIMATION OF SPECTRAL-AVERAGED CROSS SECTIONS AND AVERAGED ENERGIES FROM REACTION RATE RATIOS

A spectral index such as the ratio $^{238}\text{U}(\text{n},\text{f})/^{235}\text{U}(\text{n},\text{f})$ or $^{54}\text{Fe}(\text{n},\text{p})/^{58}\text{Fe}(\text{n},\gamma)$ can be used to show variations in neutron spectra (see Section IV-C-2).^(3,10) Such a spectral index could be used to relate to spectral-averaged cross sections, $\bar{\sigma}$; mean neutron energy, $\bar{E}$; and median neutron energy, E_m .

The values $\bar{\sigma}$, $\bar{E}$, E_m , and activity ratios were determined using SAND-II results and data taken from a matrix of activities constructed from the EBR-II run 31F data. Data are included at 13 axial positions up to 76 cm above midplane in six radial positions out to 33.6 cm. A least-square-fit, tenth-order polynomial was fit to all of the data for a given quantity. The resulting curves are shown in Figures 21 to 28.* Overall, the percent difference between the least-square curve and the multiple foil derived data is smallest for the $^{238}\text{U}/^{235}\text{U}$ index.

For a core configuration similar to 31F, $\bar{\sigma}$ may be used to estimate total fluxes in EBR-II to within 33% of the integral flux determined by multiple foils. However, this method is subject to limitations such as uncertainties from changes in the reactor. The spectral indices in Figures 21 to 28 are

* The spectral averaged cross sections of ^{235}U and ^{58}Fe are plotted as $1/\bar{\sigma}$ rather than as $\bar{\sigma}$ because $1/\bar{\sigma}$ was more convenient for making a Least-square polynomial fit.

Neg 701853-12

FIGURE 21. $^{54}\text{Fe}(n,p)$ $\bar{\sigma}$ Versus Reaction Rate Ratio $^{54}\text{Mn}/^{59}\text{Fe}$

Neg 701853-15

FIGURE 22. $^{58}\text{Fe}(n,p) \bar{\sigma}$ Versus Reaction Rate Ratio $^{54}\text{Mn}/^{59}\text{Fe}$

Neg 701853-14

FIGURE 23. Mean Energy Versus Reaction Rate Ratio $^{54}\text{Mn}/^{59}\text{Fe}$

Neg 701853-13

FIGURE 24. Medium Energy Versus Reaction Rate Ratio $^{54}\text{Mn}/^{59}\text{Fe}$

FIGURE 25. $^{238}\text{U}(n,f) \bar{\sigma}$ Versus Reaction Rate Ratio $^{238}\text{U}/^{235}\text{U}$

Neg 701853-8

FIGURE 26. $^{235}\text{U}(n,f) \bar{\sigma}$ Versus Reaction Rate Ratio $^{238}\text{U}/^{235}\text{U}$

FIGURE 27. Mean Energy Versus Reaction Rate Ratio $^{238}\text{U}/^{235}\text{U}$

Neg 701853-9

FIGURE 28. Medium Energy Versus Reaction Rate Ratio $^{238}\text{U}/^{235}\text{U}$

specific to the reactor operating conditions for Run 31F (in a structural subassembly, with a radial uranium blanket, and at a reactor power of 50 MW). Limited data from a structural experiment in X030 (radius = 27.3 cm) indicate that indices for Run 31F are applicable in the core region ($z < \sim 25$ cm) for a reactor configuration with a stainless steel radial blanket. Above this core region, the X030 index versus $\bar{\sigma}$ curve is higher than the 31F curve, by $\sim 50\%$. Consequently, using the least-square-fit curves in Figures 21, 22, 25, and 26 to determine $\bar{\sigma}$'s could result in unacceptably large errors.

By the reaction rate ratio method, the hardness of the neutron energy spectrum may be estimated from the values $\bar{E}$ and E_m (Figures 23, 24, 27, and 28); however, the method does not provide a precise determination of the energy-dependent integral flux $\Phi(>E)$.

In conclusion, indices reported here may be used judiciously to determine total integral flux from $\bar{\sigma}$, or to determine $\bar{E}$ and E_m for reactor configurations similar to Run 31F. The indices have limitations since they do not adequately determine integral fluxes for all energy limits.

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