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REV. 1

MASTER

REVIEW OF MICROSCOPIC INTEGRAL CROSS SECTION DATA IN FUNDAMENTAL REACTOR DOSIMETRY BENCHMARK NEUTRON FIELDS

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1. INTRODUCTION AND SCOPE

This paper is intended to review and critically discuss microscopic integral cross section measurement and calculation data for fundamental reactor dosimetry benchmark neutron fields.[1,2] The wording "fundamental" is used to indicate that, from the three categories[3] of benchmark neutron fields: (1) standard, (2) reference, and (3) controlled-environment, only the first two will be considered here. Specifically the review covers the following fundamental benchmarks:

- χ_{82}; the spontaneous californium-252 fission neutron spectrum standard field.[4,5,6,7]
- χ_{25}; the thermal-neutron induced uranium-235 fission neutron spectrum standard field.[8,9]
- ΣΣ; the (secondary)[10] intermediate-energy standard neutron field at the center of the Mol-ΣΣ,[11] NISUS,[12] and ITN-ΣΣ[13] facilities.
- CFRMF; the reference neutron field at the center of the Coupled Fast Reactor Measurement Facility.[14]
- BIG-10; the reference neutron field at the center of the 10% enriched uranium metal, cylindrical, fast critical.[15]
- ISNF; the (primary) Intermediate-Energy Standard Neutron Field.[16]

The restriction to standard and reference fields in the selection of data to be reviewed here does not mean that the work done in numerous controlled environments is deemed unuseful. It is believed, however, that controlled-environment data should, with some exceptions, generally serve to confirm nuclear data trends rather than to assess them; partly because of the poorer quality of integral results and neutron spectral characterization for these fields; partly also because controlled environment fields are not usually built with the primary purpose to validate and improve nuclear data, while this is one of the primary objectives for fundamental benchmarks. One of the main exceptions to the above is for data development and testing where

high flux-fluence exposures are required for stable and long half-life reaction products in controlled environment facilities such as EBR-II.

The present paper is a follow-on study of three recent publications[17,18,19] dealing with the same subject. Some redundancy in the subject material is unavoidable, but it is kept to a minimum. First, an updating of experimental results is presented; these results are compared with calculated values using the ENDF/B-IV dosimetry cross section file;[20] and finally a review is presented of the current accuracy and consistency of differential-energy cross sections and neutron spectral distributions in terms of integral reaction rate predictions. The steps involved in this review are:

(1) All measured integral reaction rates are transformed into spectral average cross sections by normalizing them with total fluxes derived through a flux transfer from a calibrated californium-252 fission neutron flux.

(2) The neutron spectra for all benchmark fields, except ^{252}Cf, are subject to an ad hoc adjustment which best matches measured spectral average cross sections for a selected set of category I* reactions. The latter have been chosen generally from among reactions with differential cross sections known to better than average accuracy and for which observed integral responses in a Cf spectrum agree well with an evaluated ^{252}Cf spectrum shape from spectrometry.

(3) On the basis of these adjusted spectral shapes, bias factors (measured to computed average cross section ratios) are defined for non-category I reactions (category II)* for each benchmark. If available studies of the evaluated differential-energy cross section of a given category II reaction suggest that the cross section shape is well defined and within uncertainties, an average bias factor is derived for each reaction. This bias factor is treated as a cross section scale normalization correction for that reaction.

In McElroy's et al companion paper to this meeting, category I reactions, supplemented by the category II reactions that can be normalized as explained above, are used to infer improved adjusted spectral shapes in

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* Category I and II reactions are defined in Reference [1].

all benchmarks. It is recognized that the procedure followed in the two papers disregards assigned benchmark spectra based on spectrometry and/or calculation in energy ranges where these are deemed less reliable. The results of the investigation in this and McElroy el al paper are provisional and will have to be compounded with similar results obtained using assigned benchmark spectra when a number of them will have been updated by accounting for the most recent data, for instance ENDF/B-IV reactor physics computations. In this iterative way only is it possible to achieve final consistent cross sections for reactions in both categories I and II.

2. DATA DEVELOPMENT - PRESENTATION AND BRIEF DISCUSSION OF MEASURED INTEGRAL CROSS SECTIONS

2.1 NORMALIZATION OF INTEGRAL DATA

In general, integral measurements in benchmark neutron fields do not yield integral cross sections, but absolute reaction rates per nucleus per sec $\int_0^\infty \sigma_r^i(E)\ \phi_k(E)\ dE$, where $\sigma_r^i(E)$ is the evaluated differential-energy cross section for the r^{th} type reaction and i^{th} isotope, and $\phi_k(E)$ is the energy-dependent neutron flux spectrum of the benchmark field, ϕ_k.

The integral, or average, microscopic cross section is equal to the absolute reaction rate divided by the total absolute flux:

$$\bar{\sigma}_r(i,\phi_k) = \int_0^\infty \sigma_r^i(E)\ \phi_k(E)\ dE \Big/ \int_0^\infty \phi_k(E)\ dE. \qquad (1)$$

As the total flux is most often unknown, the results of such measurements are quoted as reaction rates or reaction rate ratios, e.g., integral cross section ratios. If the integral cross section however is independently known for one (or a few) reaction(s), normalization of all data is straightforward.

Another easy normalization, as done for instance by use of the SAND-II Code[21], consists of defining the total flux as the weighted mean ratio of the measured to the computed quantities, right-hand side of relation (1). The accuracy of such normalization depends on the accuracy with which the benchmark spectral shape and category I and II cross sections are known[17] and is subject to the errors involved in absolute reaction rate

determinations.

The normalization adopted for this paper (and a previous one[18]*) involves a flux transfer, using the $^{239}Pu(n,f)$ reaction and the NBS californium-252 source, along lines outlined by Grundl et al.[22,29] In this technique, all of the errors listed above either disappear or are substantially reduced. The californium source was chosen for this purpose because its strength has been established to ±1.1% (1σ)[23] in the NBS Manganous Sulfate Bath Facility relative to the internationally compared standard Ra-Be photoneutron source, NBS-1. The absolute flux at the NBS californium-252 facility is derived directly from the source strength and a distance measurement; the uncertainty of the total free-field flux is estimated to be ±1.4% (1σ). The $^{239}Pu(n,f)$ reaction was used for the transfer because the reaction cross section is among the better known reactions and because of its relatively flat shape in the energy range of interest. The $^{239}Pu(n,f)$ reaction displays almost a constant average cross section in the benchmark neutron fields: the computed values are 1789, 1781, 1754 and 1735 mb for the χ_{82}, χ_{25}, CFRMF and ΣΣ benchmarks, respectively; this is, in the worst case, a difference of only 3%. Interrelated NBS fissionable deposits of plutonium-239 have been exposed in all these neutron fields in the NBS double absolute ionization fission chamber;[24] therefore, the flux transfer is rather direct and its accuracy is of the order of ±0.8% or better.

Not surprisingly, this absolute flux normalization departs by as much as 6% for two of the benchmarks, CFRMF and ΣΣ, from the ones initially derived[2,25] by means of the SAND-II approach using both category I and II reaction cross sections, but agrees better with more recent application of this approach using just the category I type reactions; i.e., using an improved selection of the most reliable detector reactions.[17] The use of the NBS flux transfer method is the reason why the absolute values of integral data tabulated in this paper are significantly different from the ones in previous compilations.**[18,25]

2.2 SURVEY OF INTEGRAL DATA

In this section, a survey and brief discussion is provided of

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* Tables VI and VII of this reference.

** For ΣΣ and CFRMF, see Tables IV and V of Reference [18].

the status of the development of microscopic integral cross sections in the fundamental benchmark neutron fields identified in Section 1. The emphasis here is on new data made available since - or data not covered at - the first ASTM-EURATOM 1975 symposium.[18]

χ_{82}

A large array of new integral activation cross sections has been reported recently[7] by a Hungarian group. This work seems extensive and generally agrees with earlier data, but it is poorly documented and the quoted uncertainties are large, of the order of 10%; many investigated reactions are not part of the ENDF/B-IV file and are therefore not considered for the present study. Only a few new measurements performed by this group are listed in the revised tabulation, Table I.

The capture cross section of gold recently measured by Green[6] using a californium source similar to the NBS one[4] is now preferred to previous data, which are much higher. It is wondered if the higher integral values do not bear some relationship to the materials and design of the individual sources. It is noted that the higher data are consistent with the proton recoil spectrometry observation[28] of a large excess of neutrons in χ_{82} below 1 MeV.

In conclusion, Table I gathers the experimental data presently considered as recommended for χ_{82}, and compares them with calculated values using the NBS χ_{82} spectral evaluation[29] and the ENDF/B-IV cross section file.[20]

χ_{25}

New measurements of integral fission cross sections for ^{235}U, ^{239}Pu, ^{238}U, ^{237}Np, and ^{232}Th* have been performed in the Mol Cavity Fission Neutron Spectrum Standard Field. They are reported in a contributed paper[30] to this meeting and agree very well with the revised evaluated χ_{25} data.[18]

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* For the first time, fission cross sections for ^{233}U and ^{241}Pu have been obtained; they are $\bar{\sigma}_f(^{233}U,\chi_{25}) = (1881 \pm 64)$mb and $\bar{\sigma}_f(^{241}Pu,\chi_{25}) = (1614 \pm 60)$mb.

TABLE I. MICROSCOPIC INTEGRAL CROSS SECTIONS IN THE ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRUM

REACTION	INTEGRAL CROSS SECTION (mb)		
	MEASURED $\bar{\sigma}_r(i,x_{82})$	CALCULATED[a] $\int_0^\infty \sigma_r^i(E)x_{82}(E)dE$	
		$\sigma_r^i(E)$: ENDF/B-IV	$\sigma_r^i(E)$: SAND II
$^{115}In(n,\gamma)^{116m}In$	125.3 ± 4.3 [26]	130.3	141.4
$^{197}Au(n,\gamma)^{198}Au$	79.9 ± 2.9 [6]	79.9	82.2
$^{235}U(n,f)$	1203 ± 30 [4]	1241	1239
$^{239}Pu(n,f)$	1804 ± 45 [27]	1789	1819
$^{237}Np(n,f)$	1332 ± 37 [27]	1351	1305
$^{103}Rh(n,n')^{103m}Rh$	757 ± 53 [7]	--	--
$^{115}In(n,n')^{115m}In$[b]	198 ± 5 [5]	191.1	190.7
$^{238}U(n,f)$	320 ± 9 [27]	315.4	313.7
$^{47}Ti(n,p)^{47}Sc$	18.9 ± 0.4 [5]	23.84	18.58
$^{58}Ni(n,p)^{58}Co$	118 ± 3 [5]	115.0	114.2
$^{54}Fe(n,p)^{54}Mn$	84.6 ± 2 [5]	89.1	87.1
$Ti(n,x)^{46}Sc$	13.8 ± 0.3 [5]	12.52	13.69
$^{27}Al(n,p)^{27}Mg$	5.1 ± 0.5 [7]	5.14	4.80
$^{56}Fe(n,p)^{56}Mn$	1.45 ± 0.035 [5]	1.475	1.549
$^{27}Al(n,\alpha)^{24}Na$	1.006 ± 0.022 [5]	1.059	1.024
$^{48}Ti(n,p)^{48}Sc$	0.42 ± 0.01 [5]	0.265	0.383
$^{55}Mn(n,2n)^{54}Mn$	0.58 ± 0.06 [7]	0.528	--
$^{59}Co(n,2n)^{58}Co$	0.57 ± 0.06 [7]	0.379	--
$^{63}Cu(n,2n)^{62}Cu$	0.30 ± 0.03 [7]	--	0.214

(a) $x_{82}(E)$: NBS evaluation[29] $\int_0^\infty x_{82}(E)dE = 1.$

(b) For ^{115m}In γ ray branching ratio of 45.9%.

(c) A value of 205 ± 9 mb. is reported in [26].

In the same facility, Williams and Hannan from the University of London Reactor Center (ULRC) have recently remeasured the fission spectrum average cross sections for the reactions $^{115}In(n,n')^{115m}In$, $^{58}Ni(n,p)^{58}Co$, $^{64}Zn(n,p)^{64}Cu$, $^{56}Fe(n,p)^{56}Mn$, $^{24}Mg(n,p)^{24}Na$ and $^{27}Al(n,\alpha)^{24}Na$. The preliminary results of this work support the χ_{25} evaluated data,[18] except for the zinc reaction.

In conclusion, recent interlaboratory work in χ_{25} provides added confidence in the previous evaluation[18] of microscopic integral cross sections for this benchmark. The results of this evaluation are compared in Table II with the values computed using different sets of available differential data.

ΣΣ

NBS-type absolute fission chambers have been used as probes to validate the spectral integrity and effective identity of the central neutron field at the Mol-ΣΣ facility[34], at the ITN-ΣΣ facility[13] (Bucharest, Rumania) and at the NISUS facility[35] (ULRC, London, Great Britain). The observed integral fission cross section ratios are displayed in Table III.

Further intercomparison of NISUS and Mol-ΣΣ has been done by the ULRC experts for 8 activation reactions.

The conclusion is that the three neutron fields are indeed neutronically equivalent in terms of integral reaction rate measurements.

In Table IV are compared ΣΣ average activation cross sections* as independently measured by the CEN-SCK, Mol experts[18] and by the ULRC, London experts. Except for the $^{27}Al(n,\alpha)^{24}Na$ reaction, the agreement is very gratifying. The ∿12% discrepancy for aluminum is difficult to understand (in χ_{25}, the Mol and ULRC data agree with each other) and requires additional work. At this stage, the ULRC datum is preferred to the CEN-SCK because it better matches the systematical trends observed when comparing the results of the different benchmarks with each other.

Table V gathers ΣΣ experimental integral cross section data as presently

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* Data normalization as outlined in Section 2.1.

TABLE II. MICROSCOPIC INTEGRAL CROSS SECTIONS IN THE URANIUM-235 THERMAL NEUTRON INDUCED FISSION NEUTRON SPECTRUM

REACTION	EFFECTIVE THRESHOLD (MeV)	INTEGRAL CROSS SECTION (mb)						
		MEASURED	COMPUTED $\int_0^\infty \sigma_r^i(E)\,\chi_{25}(E)\,dE$					
			$\sigma_r^i(E)$: ENDF/B-IV $\chi(E)$:			$\sigma_r^i(E)$: SAND-II (c) $\chi(E)$:	$\sigma_r^i(E)$: RECENT LITERATURE $\chi(E)$: MAXWELLIAN $\bar{E}$ = 1.97 MeV	
		$\bar{\sigma}_r(i,\chi_{25})$	MAXWELLIAN $\bar{E}$ = 1.97 MeV	NBS EVAL. (b) $\bar{E}$=1.98 MeV	WATT $\bar{E}$ = 2.00 MeV	WATT $\bar{E}$=2.00 MeV	[31]	[32]
$^{115}In(n,\gamma)^{116m}In$		134.5 ± 6	137.1	135.9	135.0	146.4	--	--
$^{197}Au(n,\gamma)^{198}Au$		83.5 ± 5	85.5	84.6	83.0	85.5	--	--
$^{63}Cu(n,\gamma)^{64}Cu$		9.30 ± 1.4	11.07	10.99	10.82	10.87	--	--
$^{235}U(n,f)$		1203 ± 30	1243	1241	1241	1241	--	--
$^{239}Pu(n,f)$		1811 ± 60	1782	1781	1785	1817	--	--
$^{237}Np(n,f)$	0.6	1312 ± 50	1320	1320	1337	1293	1320	--
$^{103}Rh(n,n')^{103m}Rh$ (a)	0.8	733 ± 38	--	--	--	--	720	--
$^{115}In(n,n')^{115m}In$	1.2	189 ± 8	180.9	182.2	185.7	185.3	170	173 [33]
$^{232}Th(n,f)$	1.4	81 ± 5.4	68.6	69.0	70.2	71.3	70.8	--
$^{238}U(n,f)$	1.5	305 ± 10	294.0	295.8	301.6	300.4	276	293
$^{47}Ti(n,p)^{47}Sc$	2.2	19.0 ± 1.4	21.2	21.4	21.7	17.2	--	21.0
$^{31}P(n,p)^{31}Si$	2.4	35.5 ± 2.7	--	--	--	33.0	--	--
$^{58}Ni(n,p)^{58}Co$	2.8	108.5 ± 5.4	100.5	101.6	102.8	102.2	98.1	99.4
$^{64}Zn(n,p)^{64}Cu$	2.8	29.9 ± 1.6	--	--	--	--	42.9	32.8
$^{32}S(n,p)^{32}P$	2.9	66.8 ± 3.7	63.3	64.1	65.0	60.9	64.8	--
$^{54}Fe(n,p)^{54}Mn$	3.1	79.7 ± 4.9	76.9	77.7	78.4	76.3	--	74.3
$Ti(n,x)^{46}Sc$	3.9	11.8 ± 0.75	10.08	9.99	9.92	11.28	--	--
$^{27}Al(n,p)^{27}Mg$	4.4	3.86 ± 0.25	4.16	4.12	4.10	3.84	--	3.72
$^{56}Fe(n,p)^{56}Mn$	6.0	1.035 ± 0.075	1.119	1.053	1.035	1.085	--	--
$^{59}Co(n,\alpha)^{56}Mn$	6.8	0.143 ± 0.010	0.163	0.148	0.146	--	--	--
$^{63}Cu(n,\alpha)^{60}Co$	6.8	0.500 ± 0.056	0.386	0.352	0.347	0.473	--	--
$^{24}Mg(n,p)^{24}Na$	6.8	1.48 ± 0.082	--	--	--	1.498	--	--
$^{27}Al(n,\alpha)^{24}Na$	7.2	0.705 ± 0.040	0.780	0.693	0.684	0.663	--	--
$^{48}Ti(n,p)^{48}Sc$	7.6	0.300 ± 0.018	0.195	0.173	0.169	0.236	--	--
$^{93}Nb(n,2n)^{92m}Nb$	10.2	0.475 ± 0.032	--	--	--	--	--	0.38 --
$^{127}I(n,2n)^{126}I$	10.5	1.05 ± 0.065	1.574	1.186	1.149	0.686	--	--
$^{55}Mn(n,2n)^{54}Mn$	11.6	0.244 ± 0.015	0.348	0.245	0.232	--	--	--
$^{63}Cu(n,2n)^{62}Cu$	12.4	0.122 ± 0.012	--	--	--	0.0846	--	--
$^{90}Zr(n,2n)^{89}Zr$	∿13	0.247 ± 0.017	--	--	--	0.0795	--	--
$^{58}Ni(n,2n)^{57}Ni$	∿13.5	0.00577 ± 0.00031	0.00448	0.00282	0.00254	0.00239	--	--

(a) Measured and calculated results rescaled to be consistent with a branching ratio of 45.9% for the ^{115m}In 336.2 KeV gamma ray.
(b) Segment-adjusted spectrum.
(c) SAND-II edited tape associated to NBS DETAN code.

TABLE III. INTEGRAL FISSION CROSS SECTION RATIOS MEASURED BY MEANS OF NBS-TYPE ABSOLUTE FISSION CHAMBERS[a] IN THE ΣΣ NEUTRON FIELD AT THE CENTER OF THE MOL-ΣΣ, ITN-ΣΣ AND NISUS FACILITIES

CROSS SECTION RATIO	FACILITY		
	MOL-ΣΣ	ITN-ΣΣ	NISUS
$\frac{\bar{\sigma}_f(^{239}Pu,\Sigma\Sigma)}{\bar{\sigma}_f(^{235}U,\Sigma\Sigma)}$	1.173(±2.1%)	1.169(±2.3%)	1.175(±2.3%)
$\frac{\bar{\sigma}_f(^{238}U,\Sigma\Sigma)}{\bar{\sigma}_f(^{235}U,\Sigma\Sigma)}$	0.0564(±2.5%)	0.0566(±2.5%)	0.0568(±2.7%)
$\frac{\bar{\sigma}_f(^{237}Np,\Sigma\Sigma)}{\bar{\sigma}_f(^{235}U,\Sigma\Sigma)}$	0.381(±2.8%)	0.380(±3.0%)	0.383(±3.0%)

(a) Interlaboratory results[34] in Mol-ΣΣ are 1.000: 1.167(±2%): 0.0561(±1.5%): 0.388(±2.5%) for ^{235}U: ^{239}Pu: ^{238}U: ^{237}Np, respectively.

TABLE IV. COMPARISON OF INDEPENDENT MICROSCOPIC INTEGRAL CROSS SECTION MEASUREMENTS AT THE CENTER OF THE MOL-ΣΣ FACILITY

REACTION	$\bar{\sigma}_r(i,\Sigma\Sigma)$ mb CEN-SCK	ULRC	DIFFERENCE
$^{197}Au(n,\gamma)^{198}Au$	401±10	404±13	+0.7%
$^{115}In(n,\gamma)^{116m}In$	237± 9	243± 8	+2.5%
$^{115}In(n,n')^{115m}In$	56.0±1.4	56.0±1.4	0
$^{58}Ni(n,p)^{58}Co$	26.5±0.8	26.2±0.9	-1.1%
$^{56}Fe(n,p)^{56}Mn$	0.260±0.008	0.261±0.010	+0.4%
$^{27}Al(n,\alpha)^{24}Na$	0.173±0.005	0.153±0.005	-12.3%

TABLE V. MICROSCOPIC INTEGRAL CROSS SECTIONS IN BENCHMARK NEUTRON FIELD ΣΣ

Reaction	$\bar{\sigma}_r(i,\phi)$ Measured (mb.)	$\int_0^\infty \sigma_r^i(E)\,\phi(E)dE$ [a] Calculated (mb.)	Measured/ Calculated
$^{59}Co(n,\gamma)^{60}Co$	--	41.5	--
$^{58}Fe(n,\gamma)^{59}Fe$	--	5.44	--
$^{55}Mn(n,\gamma)^{56}Mn$	36.0 ± 2.0	--	--
$^{63}Cu(n,\gamma)^{64}Cu$	36.2 ± 2.0	38.9	0.932
$^{197}Au(n,\gamma)^{198}Au$	402 ± 10	373.5	1.076
$^{238}U(n,\gamma)^{239}U$	[174 ± 7 [b]]	222	(0.784)
$^{10}B(n,\alpha)^{7}Li$	--	1518	--
$^{45}Sc(n,\gamma)^{46}Sc$	--	19.0	--
$^{115}In(n,\gamma)^{116m}In$	240 ± 9	285	0.842
$^{6}Li(n,\alpha)^{3}H$	--	923.5	--
$^{235}U(n,f)$	1512 ± 55	1525	0.991
$^{239}Pu(n,f)$	1764 ± 65	1735	1.017
$^{237}Np(n,f)$	586.5 ± 20	607	0.966
$^{103}Rh(n,n')^{103m}Rh$	281 ± 8.5	--	--
$^{115}In(n,n')^{115m}In$	56.0 ± 1.4 [c]	55.2	1.014
$^{238}U(n,f)$	84.8 ± 2.5	81.2	1.044
$^{47}Ti(n,p)^{47}Sc$	--	5.15	--
$^{58}Ni(n,p)^{58}Co$	26.5 ± 0.8	23.3	1.139
$^{54}Fe(n,p)^{54}Mn$	--	17.2	--
$Ti(n,x)^{46}Sc$	--	2.07	--
$^{27}Al(n,p)^{27}Mg$	0.983 ± 0.10	0.869	1.131
$^{56}Fe(n,p)^{56}Mn$	0.260 ± 0.008	0.230	1.130
$^{27}Al(n,\alpha)^{24}Na$	0.153 ± 0.005	0.152	1.007
$^{48}Ti(n,p)^{48}Sc$	--	0.0370	--

(a) $\sigma_r^i(E)$: ENDF/B-IV dosimetry file[20]; $\phi(E)$: as recommended in [11], normalized $\int_0^\infty \phi(E)dE=1$.

(b) Uncorrected for spectral shielding effect (~20%) in reactor constituents.

(c) For ^{115m}In γ branching ratio of 45.9%.

recommended* and compares them with predictions based on the spectral shape published earlier[11] and the ENDF/B-IV dosimetry file.[20]

CFRMF

Previous preliminary $^{10}B(n,\alpha)^{7}Li$ and $^{6}Li(n,\alpha)^{3}He$ reaction rate data** in CFRMF[36] have been updated. New measurements have been completed by Farrar and the initial results of his analysis are used here.

The uranium-238 capture rate data for CFRMF have been corrected by Harker for neutron spectrum shielding effects in the natural uranium block constituting the central zone of the reactor. To this end, Rabble,[37] a multiregion resonance absorption cross section cell code, with space-and-energy-dependent slowing-down sources and ultra-fine energy group structure ($\Delta u = 0.001$), has been used to prepare properly shielded coarse group-averaged cross sections for CFRMF. Corrections to observed uranium-238 capture rates determined in this way are of the order of 20%.***

Except for these three reaction rates, the CFRMF data remain unchanged and are compared**** in Table VI with the values computed from differential data.

BIG-10

Big-10 reaction rate data were not previously available; the data,***** quoted for the first time in this paper, Table VII, are still of a preliminary nature and do not represent a concensus of the Interlaboratory LMFBR Reaction

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* Data normalization as outlined in Section 2.1.

** More specifically the measurements are for the total helium production, but the difference between the n, total helium and n,α reaction production of helium is negligibly small.

*** A similar correction should be applied to measured uranium-238 capture rates in $\Sigma\Sigma$, but has not yet been computed.

**** Experimental data normalized as outlined in Section 2.1.

*****Analytical flux depression corrections by Hansen have been applied to relate the response of a real detector in its real environment to the response of the corresponding infinitesimal detector at the center of a cavity-free Big-10. Such corrections never exceed 3.5%.

TABLE VI. MICROSCOPIC INTEGRAL CROSS SECTIONS IN BENCHMARK NEUTRON FIELD CFRMF

Reaction	$\bar{\sigma}_r(i,\phi)$ Measured (mb.)	$\int_0^\infty \sigma_r^i(E)\ \phi(E)dE$ [a] Calculated (mb)	Measured/ Calculated
$^{59}Co(n,\gamma)^{60}Co$	91.6 ± 3.6	85.3	1.074
$^{58}Fe(n,\gamma)^{59}Fe$	6.12 ± 0.22	6.11	1.002
$^{55}Mn(n,\gamma)^{56}Mn$	-	-	-
$^{63}Cu(n,\gamma)^{64}Cu$	45.4 ± 2.6	47.5	0.956
$^{197}Au(n,\gamma)^{198}Au$	424 ± 14	416	1.019
$^{238}U(n,\gamma)^{239}U$	223 ± 11 [b]	232	0.961
$^{10}B(n,\alpha)^{7}Li$	1814 ± 60 [c]	1694	1.071
$^{45}Sc(n,\gamma)^{46}Sc$	23.5 ± 0.9	20.1	1.166
$^{115}In(n,\gamma)^{116m}In$	281.5 ± 11	303	0.929
$^{6}Li(n,\alpha)^{3}H$	948 ± 39 [c]	988.5	0.959
$^{235}U(n,f)$	1557 ± 53	1590	0.979
$^{239}Pu(n,f)$	1783 ± 60	1754	1.016
$^{237}Np(n,f)$	551 ± 21	547	1.007
$^{115}In(n,n')^{115m}In$	51.0 ± 3.0 [d]	47.5 [d]	1.074
$^{238}U(n,f)$	75.6 ± 3.0	69.4	1.089
$^{47}Ti(n,p)^{47}Sc$	4.18 ± 0.2	4.70	0.889
$^{58}Ni(n,p)^{58}Co$	24.0 ± 0.8	21.9	1.093
$^{54}Fe(n,p)^{54}Mn$	17.5 ± 0.6	16.5	1.061
$Ti(n,x)^{46}Sc$	2.61 ± 0.10	2.15	1.214
$^{27}Al(n,p)^{27}Mg$	0.874 ± 0.033	0.887	0.985
$^{56}Fe(n,p)^{56}Mn$	-	0.238	-
$^{27}Al(n,\alpha)^{24}Na$	0.161 ± 0.005	0.162	0.994
$^{48}Ti(n,p)^{48}Sc$	0.0688 ± 0.003	0.0385	1.787

(a) $\sigma_r^i(E)$: ENDF/B-IV dosimetry file[20]; $\phi(E)$: ENDF/B-III S_N computation[14], normalized $\int_0^\infty \phi(E)\ dE = 1$.
(b) Corrected for spectral shielding effect (21%) in reactor constituents.
(c) New measurements.
(d) For ^{115m}In γ branching ratio of 45.9%.

TABLE VII. MICROSCOPIC INTEGRAL CROSS SECTIONS IN BENCHMARK NEUTRON FIELD BIG-10: PRELIMINARY [a]

Reaction	$\bar{\sigma}_r(i,\phi)$ Measured (mb.)	$\int_0^\infty \sigma_r^i(E)\ \phi(E)dE$ [b] Calculated (mb)	Measured/ Calculated
$^{59}Co(n,\gamma)^{60}Co$	12.94 ± 0.4	12.59	1.028
$^{58}Fe(n,\gamma)^{59}Fe$	4.27 ± 0.21	3.09	1.382
$^{55}Mn(n,\gamma)^{56}Mn$	-	-	-
$^{63}Cu(n,\gamma)^{64}Cu$	23.1 ± 0.9	24.8	0.932
$^{197}Au(n,\gamma)^{198}Au$	228 ± 6	219	1.041
$^{238}U(n,\gamma)^{239}U$	149.5 ± 4.5	149.2	1.002
$^{10}B(n,\alpha)^{7}Li$	1378 ± 28	1208	1.141
$^{45}Sc(n,\gamma)^{46}Sc$	17.86 ± 0.55	15.87	1.125
$^{115}In(n,\gamma)^{116m}In$	-	232.5	-
$^{6}Li(n,\alpha)^{3}H$	967 ± 19	966	1.001
$^{235}U(n,f)$	1361 ± 18 [39]	1368	0.995
$^{239}Pu(n,f)$	1632 ± 33 [39]	1605	1.017
$^{237}Np(n,f)$	433.5 ± 11 [39]	440	0.985
$^{115}In(n,n')^{115m}In$	35.65 ± 1.1 [c]	32.64 [c]	1.092
$^{238}U(n,f)$	50.9 ± 1.1 [39]	46.00	1.107
$^{47}Ti(n,p)^{47}Sc$	2.96 ± 0.13	3.13	0.945
$^{58}Ni(n,p)^{58}Co$	16.87 ± 0.34	14.56	1.159
$^{54}Fe(n,p)^{54}Mn$	12.26 ± 0.31	10.95	1.120
$Ti(n,x)^{46}Sc$	1.81 ± 0.06	1.40	1.290
$^{27}Al(n,p)^{27}Mg$	-	0.580	-
$^{56}Fe(n,p)^{56}Mn$	-	0.151	-
$^{27}Al(n,\alpha)^{24}Na$	0.110 ± 0.007	0.102	1.078
$^{48}Ti(n,p)^{48}Sc$	0.0487 ± 0.0020	0.0244	1.993

(a) Based only on HEDL, NBS and AI measurements.

(b) $\sigma_r^i(E)$: ENDF/B-IV dosimetry file[20]; $\phi(E)$: ENDF/B-III S_N computation, normalized $\int_0^\infty \phi(E)\ dE = 1$.

(c) For ^{115m}In γ branching ratio of 45.9%.

Rate (ILRR) program[2], currently responsible for the work performed in the U. S. dosimetry benchmarks.

The experimental average cross sections listed in Table VII have been normalized as outlined in Section 2.1. The fission cross sections are based on measurements[38] by means of NBS double absolute fission chambers[24] by Gilliam, Grundl et al. The $^{10}B(n,\alpha)$ and $^{6}Li(n,\alpha)$ cross sections result from helium production rate measurements by Farrar. All other cross sections have been obtained by radiometric high resolution Ge(Li) counting performed at the Hanford Engineering Development Laboratory (HEDL) and they do not include the results obtained by the other independent laboratories; the HEDL data however depart by less than ±2% from the weighted mean of all results, except in the case of the $^{27}Al(n,\alpha)^{24}Na$ reaction for which HEDL is high by approximately 4%.

ISNF

The Intermediate-Energy Standard Neutron Field (ISNF)[16], developed jointly by NBS and CEN-SCK, and in operation at NBS since 1975, is the most recent of the benchmark fields in the ILRR "family". Therefore, only two fission cross section ratios have yet been measured in this environment and the results are preliminary. They are briefly discussed in the Section 3.2.

3. DATA TESTING

3.1 PITFALLS AND PROBLEMS IN CONVENTIONAL DATA TESTING

A major conclusion from recent[17,18] and current work[19] is that integral cross section measurements for dosimetry reactions in standard and reference benchmark neutron fields show unsatisfactory departures from those computed, not only because of differential-energy cross section inadequacies, but also because the spectral shapes characteristic of these benchmarks are usually inaccurate in the energy ranges not covered or poorly covered by differential neutron spectrometry techniques;

for example - below ∿250 KeV and above ∿10 MeV for χ_{82} and χ_{25}
- below ∿10 KeV and above ∿2 MeV for ΣΣ, CFRMF and Big-10.

Even in the well covered energy ranges, the reliability sometimes remains

questionable, as is presently the case for χ_{25} between 3 and 6 MeV,[18,19] and for CFRMF between 100 and 400 KeV.[39]

Computed neutron spectra for ΣΣ, CFRMF and Big-10 are affected to a large degree by uncertainties in the uranium-238 nuclear data, most noticeably inelastic and elastic cross sections, depending on the energy range.

Consequently a direct confrontation, Table VIII, of measured integral cross sections for the various benchmarks and their computed values using spectra based only on neutron spectrometry and transport theory does not allow the dosimetry file to be unambiguously tested and adjusted. Instead, the overall approach recommended in 1973[1] must be followed; e.g., adjust the benchmark spectra on the basis of integral microscopic cross sections for a selected category I reaction set and use this improved spectral characterization to adjust differential-energy cross sections for the other reactions, labelled category II.

The impact of spectral adjustment is illustrated for fundamental fission cross sections by Tables IX and X, which present measured, computed and measured-to-computed cross section ratios before and after adjustment. It is seen that the changes in the ratios due to the adjustment are very significant; they generally remain, however, within the uncertainties of the spectral shape characterization. A striking example is the uranium-235 to uranium-238 measured-to-computed ratio in CFRMF: before spectral adjustment, this ratio is 0.899 while it becomes 0.970 with adjustment; a new computation of the CFRMF spectrum by Harker et al., using ENDF/B-IV data, provides a new ratio of 0.983, in agreement with the adjustment.

TABLE VIII RATIO OF MEASURED[a] TO COMPUTED[b] INTEGRAL CROSS SECTIONS IN DOSIMETRY BENCHMARK NEUTRON FIELDS

REACTION	NEUTRON FIELD				
	X_{82}	X_{25}	ΣΣ	CFRMF	BIG-10
$^{59}Co(n,\gamma)^{60}Co$	-	-	-	1.074	1.028
$^{58}Fe(n,\gamma)^{59}Fe$	-	-	-	1.002	1.382
$^{63}Cu(n,\gamma)^{64}Cu$	-	0.846	0.932	0.956	0.932
$^{197}Au(n,\gamma)^{198}Au$	1.000	0.987	1.076	1.019	1.041
$^{238}U(n,\gamma)^{239}U$	-	-	(∿0.95[c])	0.961	1.002
$^{10}B(n,\alpha)^{7}Li$	-	-	-	1.071	1.141
$^{45}Sc(n,\gamma)^{46}Sc$	-	-	-	1.166	1.125
$^{115}In(n,\gamma)^{116m}In$	0.962	0.990	0.842	0.929	-
$^{6}Li(n,\alpha)^{3}H$	-	-	-	0.959	1.001
$^{235}U(n,f)$	0.969	0.969	0.991	0.979	0.995
$^{239}Pu(n,f)$	1.008	1.017	1.017	1.016	1.017
$^{237}Np(n,f)$	0.986	0.994	0.966	1.007	0.985
$^{115}In(n,n')^{115m}In$	1.036	1.037	1.014	1.074	1.092
$^{238}U(n,f)$	1.015	1.031	1.044	1.089	1.107
$^{47}Ti(n,p)^{47}Sc$	0.793	0.888	-	0.889	0.945
$^{58}Ni(n,p)^{58}Co$	1.026	1.068	1.139	1.093	1.159
$^{54}Fe(n,p)^{54}Mn$	0.949	1.026	-	1.061	1.120
$Ti(n,x)^{46}Sc$	1.102	1.181	-	1.214	1.290
$^{27}Al(n,p)^{27}Mg$	0.992	0.937	1.131	0.985	-
$^{56}Fe(n,p)^{56}Mn$	0.983	0.983	1.130	-	-
$^{27}Al(n,\alpha)^{24}Na$	0.950	1.017	1.007	0.994	1.078
$^{48}Ti(n,p)^{48}Sc$	1.585	1.734	-	1.787	1.993

(a) Normalized by $^{239}Pu(n,f)$ transfer from californium, text Section 2.1.

(b) $\int_0^\infty \sigma_r^i(E)\,\phi(E)\,dE$; $\sigma_r^i(E)$:ENDF/B-IV file; $\phi(E)$: as recommended, 1975, normalized $\int_0^\infty \phi(E)\,dE = 1$.

(c) Applying the spectral shielding correction computed for CFRMF.

TABLE IX. FUNDAMENTAL FISSION CROSS SECTION RATIOS
IN DOSIMETRY BENCHMARK NEUTRON FIELDS

Neutron Field	$\frac{^{235}U(n,f)}{^{238}U(n,f)}$			$\frac{^{239}Pu(n,f)}{^{235}U(n,f)}$			$\frac{^{239}Pu(n,f)}{^{238}U(n,f)}$
	Measured	Computed*	Ratio	Measured	Computed*	Ratio	Ratio
X_{82} (a)	3.76(±1.7%)	3.935	0.955	1.500(±1.6%)	1.442	1.040	0.993
X_{25} (a)	3.94(±2.0%)	4.195	0.939	1.505(±2.2%)	1.435	1.049	0.985
$\Sigma\Sigma$ (b)	17.8(±1.5%)	18.78	0.948	1.167(±2.0%)	1.138	1.025	0.972
CFRMF (c)	20.6(±1.4%)	22.91	0.899	1.145(±1.5%)	1.103	1.038	0.933
Big-10 (c)	26.8(±1.7%)	29.65	0.904	1.198(±1.5%)	1.173	1.021	0.923
ISNF (c)	10.8(±<3%)	11.85	0.911	1.15(±<3%)	1.114	1.033	0.941

(a) $\phi(E)$ for computation : NBS evaluation.[29]

(b) $\phi(E)$: reference.[11]

(c) $\phi(E)$: discrete-ordinates transport theory calculation based on ENDF/B-III.

(*) $\sigma_f(E)$: ENDF/B-IV.

TABLE X. RATIO OF MEASURED-TO-COMPUTED INTEGRAL FISSION CROSS SECTIONS IN DOSIMETRY BENCHMARK NEUTRON FIELDS BEFORE (b) AND AFTER (a) ADJUSTMENT OF THE SPECTRAL SHAPES

NEUTRON FIELD	$\left[\frac{^{235}U(n,f)}{^{238}U(n,f)}\right]_{Meas.} / \left[\frac{^{235}U(n,f)}{^{238}U(n,f)}\right]_{Calc.}$		$\left[\frac{^{239}Pu(n,f)}{^{235}U(n,f)}\right]_{Meas.} / \left[\frac{^{239}Pu(n,f)}{^{235}U(n,f)}\right]_{Calc.}$		$\left[\frac{^{239}Pu(n,f)}{^{238}U(n,f)}\right]_{Meas.} / \left[\frac{^{239}Pu(n,f)}{^{238}U(n,f)}\right]_{Calc.}$	
	b	a*	b	a*	b	a
χ_{82}	0.955	0.955**	1.040	1.040**	0.993	0.993**
χ_{25}	0.939	0.958	1.049	1.051	0.985	1.007
$\Sigma\Sigma$	0.948	0.980	1.025	1.028	0.972	1.007
CFRMF	0.899	0.970	1.038	1.030	0.933	0.999
BIG-10	0.904	0.970	1.021	1.030	0.923	0.999

* A decrease of the order of at least 5% in the U^{235} $\sigma(E)$ in the 0.01 to 1 MeV range would establish ratios much closer to unity.

** As indicated in the text, no adjustment was necessary for Cf^{252}.

3.2 SELECTION OF A CATEGORY I REACTION SET

The selection of a category I reaction set is in itself a challenging task. Reactions which are considered as standards by differential cross section measurers seem to fall naturally into such a category, yet there are notable exceptions: for fast neutron spectra, the principal information given by $^{235}U(n,f)$ and $^{239}Pu(n,f)$ reactions is on total fluence and one selects $^{239}Pu(n,f)$ for category I as the better fluence monitor and not on the basis of uncertainties in the differential cross section data. Again, the information given by either $^{234}U(n,f)$ or $^{237}Np(n,f)$ reactions is essentially equivalent and one selects $^{237}Np(n,f)$ for category I merely because more integral data are available.

Except for the addition of $^{197}Au(n,\gamma)^{198}Au$, the category I reactions selected for this study coincide with those of Vlasov et al[19], namely: $^{197}Au(n,\gamma)^{198}Au$, $^{239}Pu(n,f)$, $^{237}Np(n,f)$, $^{238}U(n,f)$, $^{58}Ni(n,p)^{58}Co$, $^{56}Fe(n,p)^{56}Mn$, $^{27}Al(n,\alpha)^{24}Na$, $^{63}Cu(n,2n)^{62}Cu$, and $^{58}Ni(n,2n)^{57}Ni$.*

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* Uncertainties of the order of 10% are currently acceptable in the very high energy range of response of the last two reactions.

3.3 ACCURACY OF THE ENDF/B-IV DOSIMETRY FILE

This brief discussion of the accuracy of selected reactions on the ENDF/B-IV dosimetry file will be based primarily on Table XI data, in which measured-to-calculated integral cross section ratios are quoted for SAND-II adjusted benchmark spectral shapes for all fields, except for the use of the unadjusted NBS evaluated spectrum for californium-252 (see Sections 3.1 and 3.2).

The table is divided by a horizontal line that separates threshold and non-threshold reactions; the reactions are arranged, approximately, in order of increasing $\Sigma\Sigma$ field energy response.

Also factored into the discussion is a careful outlook at the status of differential-energy cross sections.[19,20]

Non-threshold reactions*

$^{59}Co(n,\gamma)^{60}Co$: The CFRMF result suggests improper spectral characterization in the 0.1 - 1 keV range; indeed, the measured resonance integral, which accounts for more than 50% of the reaction rate in CFRMF, is reasonably well known and agrees with the calculated ENDF/B-IV value.[20] The BIG-10 results support the current file evaluation at higher energies; therefore this reaction can be used for characterizing some benchmark spectra rather than the reverse: it is a future category I candidate.

$^{58}Fe(n,\gamma)^{59}Fe$: In view of the above comments on cobalt and since the file evaluated cross section relies on a Hauser-Feshbach calculation which was lowered by 10% to establish better agreement with CFRMF integral datum,[20] the CFRMF iron result is inconclusive while the BIG-10 result suggests a serious inconsistency at higher energy; further, the measured resonance integral value is ∿33% lower

* Recommendations for new integral measurements for the non-threshold reactions in the ISNF fields are not re-stated here because they have already been made as a part of planned ILRR program work.

TABLE XI RATIO OF MEASURED[a] TO COMPUTED[b] INTEGRAL CROSS SECTIONS IN DOSIMETRY BENCHMARK NEUTRON FIELDS: SAND II ADJUSTED SPECTRAL SHAPES[c]

REACTION	NEUTRON FIELD				
	X_{82}	X_{25}	ΣΣ	CFRMF	BIG-10
$^{59}Co(n,\gamma)^{60}Co$	-	-	-	1.205	1.013
$^{58}Fe(n,\gamma)^{59}Fe$	-	-	-	1.000	1.364
$^{63}Cu(n,\gamma)^{64}Cu$	-	0.834	0.874	0.937	0.915
$^{197}Au(n,\gamma)^{198}Au$	1.000	0.972	1.002	1.000	1.019
$^{238}U(n,\gamma)^{239}U$	-	-	(0.907)[d]	0.956	0.988
$^{10}B(n,\alpha)^{7}Li$	-	-	-	1.053	1.108
$^{45}Sc(n,\gamma)^{46}Sc$	-	-	-	1.158	1.102
$^{115}In(n,\gamma)^{116m}In$	0.962	0.996	0.807	0.929	-
$^{6}Li(n,\alpha)^{3}H$	-	-	-	0.949	0.980
$^{235}U(n,f)$	0.969	0.968	0.975	0.974	0.987
$^{239}Pu(n,f)$	1.008	1.017	1.002	1.003	1.017
$^{237}Np(n,f)$	0.986	1.000	1.003	0.999	1.015
$^{115}In(n,n')^{115m}In$	1.036	1.019	0.988	1.007	1.034
$^{238}U(n,f)$	1.015	1.010	0.995	1.004	1.018
$^{47}Ti(n,p)^{47}Sc$	0.793	0.852	-	0.810	0.840
$^{58}Ni(n,p)^{58}Co$	1.026	1.020	1.008	0.999	1.017
$^{54}Fe(n,p)^{54}Mn$	0.949	0.977	-	0.967	0.975
$Ti(n,x)^{46}Sc$	1.102	1.155	-	1.125	1.129
$^{27}Al(n,p)^{27}Mg$	0.992	0.917	0.991	0.914	-
$^{56}Fe(n,p)^{56}Mn$	0.983	1.004	1.050	-	-
$^{27}Al(n,\alpha)^{24}Na$	0.950	1.022	1.000	0.999	1.014
$^{48}Ti(n,p)^{48}Sc$	1.585	1.714	-	1.686	1.859

(a) Normalized by $^{239}Pu(n,f)$ transfer from californium, text Section 2.1.

(b) $\int_0^\infty \sigma_r^i(E)\ \phi^*(E)\ dE$; $\sigma_r^i(E)$: ENDF/B-IV file; $\phi^*(E)$: SAND-II adjusted spectra[c], normalized $\int_0^\infty \phi^*(E)\ dE = 1$.

(c) Except for X_{82}: NBS evaluation[29]

(d) Applying the spectral shielding correction computed for CFRMF.

that the calculated value[20]; a χ_{25} (or χ_{82}) and a new resonance integral measurement are recommended for this reaction.

$^{63}Cu(n,\gamma)^{64}Cu$: The $\Sigma\Sigma$ datum might be slightly inaccurate due to uncertainties in the spectrum or in self-shielding corrections; the measured resonance integral is ~13% below the ENDF/B-IV calculated value; further, the increasing discrepancy from CFRMF through BIG-10 to χ_{25} seems to indicate a non-surprising cross section shape inadequacy above 1 keV; here, a more thorough evaluation effort of differential data is needed as well as a new resonance integral measurement.

$^{197}Au(n,\gamma)^{198}Au$: Category I, satisfactory.

$^{238}U(n,\gamma)^{239}U$: Satisfactory in terms of dosimetry applications but difficult to apply reliably in systems containing large amounts of uranium-238.

$^{10}B(n,\alpha)^{7}Li$: The integral versus differential data discrepancies are sizeable for such a supposedly standard cross section; they decrease slightly from Table VIII to Table XI; e.g., the spectral adjustments have a positive influence; the measured and calculated resonance integral values are in agreement.[20] In view of the importance of this reaction and the fact that only one integral experimental approach has been used, further independent integral measurements are recommended.

$^{45}Sc(n,\gamma)^{46}Sc$: The measured versus computed discrepancies are important but not surprising; the measured and calculated resonance integrals agree but the measured value has a ~10% uncertainty;[20] in the energy range of relevance for BIG-10, differential measurements are very sparse, while the CFRMF datum is influenced by the complex resonance structure above

2 keV; additional differential and integral measurements are needed in this case.

$^{115}In(n,\gamma)^{116m}In$: The differential-energy cross section seems more or less acceptable if it is assumed that the ΣΣ datum is in error; the measured and calculated resonance integrals are in agreement;[20] the ΣΣ measurement should be further validated by independent group(s).

$^{6}Li(n,\alpha)^{3}H$: The BIG-10 datum supports the evaluated file very well, but the discrepancy in CFRMF is significant and puzzling as it is for $^{10}B(n,\alpha)$; there is no measured resonance integral value reported in Reference [20]; further independent integral measurements are recommended; it is most relevant to indicate here that $^{6}Li(n,\alpha)$ spectrometry[39] in CFRMF suggests a spectral depletion in the 100-400 keV range: if such an effect were real, it would help to explain and resolve the inconsistencies in present integral observations.

$^{235}U(n,f)$: Further differential measurements and evaluations are required to establish this as a category I reaction; see discussion in Section 3.2.

$^{239}Pu(n,f)$: Category I, seems satisfactory but further differential measurements below 100 keV may be necessary; in view of the $^{235}U(n,f)$ problem, a reevaluation is needed for confirmation.

Threshold reactions

The threshold reactions are not discussed individually here, but bias factors as defined in Section 1 of this paper and in Reference [19] are tentatively recommended, wherever applicable. This is done on the basis of a review of the data in Table XI as well as that in column 4, Table 1, of Reference [19].

For some reactions, such as $^{63}Cu(n,\alpha)^{60}Co$ and $^{48}Ti(n,p)^{48}Sc$, the

evidence of differential-energy cross section shape inadequacies precludes the definition of a bias factor.

For the other reactions investigated in this work, the bias factors are as follows:

$^{237}Np(n,f)$	:	Category I
$^{115}In(n,n')^{115m}In$	:	1.017 ± 0.025
$^{232}Th(n,f)$	:	1.15
$^{238}U(n,f)$	:	Category I
$^{47}Ti(n,p)^{47}Sc$	:	0.825 ± 0.03
$^{58}Ni(n,p)^{58}Co$	:	Category I
$^{32}S(n,p)^{32}P$	:	0.987
$^{54}Fe(n,p)^{54}Mn$	:	0.967 ± 0.018
$Ti(n,x)^{46}Sc$	:	1.128 ± 0.026
$^{56}Fe(n,p)^{56}Mn$	:	Category I
$^{59}Co(n,\alpha)^{56}Mn$	:	0.973
$^{27}Al(n,\alpha)^{24}Na$	:	Category I
$^{127}I(n,2n)^{126}I$	:	0.778
$^{55}Mn(n,2n)^{54}Mn$	:	0.803
$^{63}Cu(n,2n)^{62}Cu$	:	Category I (0.90)*
$^{90}Zr(n,2n)^{89}Zr$	:	1.715
$^{58}Ni(n,2n)^{57}Ni$	:	Category I (1.12)*.

It is recommended that these bias factors be considered for use in the definition of an adjusted and improved ENDF/B dosimetry cross section file. It is believed that such an improved file would have an integral consistency to better than ±5% for the designated reactions as a result of the application of the data testing approach undertaken in this paper. It is also important to note that if a bias factor for a key fluence monitor such as

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* For the very high energy range, accuracies of the order of ±10% are presently acceptable for category I reactions.

$Ti(n,x)^{46}Sc$ had been defined by more conventional, direct data testing procedures, its value would have varied between 1.10 and 1.29; e.g., ±8%, depending on the benchmark field considered; with SAND-II adjustments, this ±8% spread is reduced to ±2-3%, which is a very significant improvement.

It must be recalled that this forced consistency ignores possible inadequacies of Category I reaction cross sections and also possible systematic errors in the integral measurements. Discrepancies between observed and expected integral results in the benchmarks are interpreted as spectrum errors alone. Thus, to be complete, this approach to spectrum characterization for neutron dosimetry must take into account as an additional error component the departures of the SAND-II adjusted spectra from the assigned spectra based on spectrometry and calculation. Alternatively a compromise spectrum may be first defined followed by the derivation of bias factors for both categories of reactions.

4. CONCLUSIONS AND RECOMMENDATIONS

The principal conclusions related to dosimetry cross section data development and testing based on this study and those presented at the Petten symposium[18] may be stated as follows:

- For category I and the best known category II ENDF/B-IV threshold reactions, integral and differential cross section data are generally consistent to within ±5% (1σ). When this is not the case, the deviations can be interpreted in an ad hoc procedure as errors in benchmark neutron field flux spectra.
- Adjustment of the benchmark neutron spectra by multiple foil unfolding on the basis of category I reactions significantly improves the overall integral versus differential data consistency, as is to be expected. These adjustments are often within the bounds of experimental uncertainties for the benchmark spectra. When this is not the case, the results suggest the existence of real biases.
- The spectral components of current benchmark neutron fields are not sufficiently well known and distinguishable to allow energy dependent adjustment of non-threshold category II cross sections. Present results do provide a good basis, however, for the future direction of evaluation work and measurements.

- When integral-differential discrepancies for category II threshold reactions are expressed as cross section rescaling or normalization factors, such bias factors can be established with uncertainties smaller than 2-3% when category I multiple foil unfolded spectral shapes are employed in place of assigned spectra based on spectrometry and calculation. Without this ad hoc adjustment biases of up to 10% or more for key fluence monitors are observed. The consequent improvement of spectrum characterization for dosimetry will depend upon the extent and reliability of the departures between unfolded and assigned spectra.

In summary, the data development and testing approach, first applied to the development of the SAND-II cross section file,[40] and subsequently recommended by the IAEA 1973 panel[1] has been further investigated and rescaled energy-dependent cross sections have been derived for category II threshold reactions. For the first time, some specific recommendations for further study of non-threshold reactions in the ENDF/B-IV file have been delineated. A few sustained problems still exist and a vigorous and well planned and coordinated international interlaboratory effort will be required to resolve them. These are:

- Uncertainties in the evaluated ENDF/B-IV differential-energy cross sections for key standard reactions such as $^{235}U(n,f)$, $^{10}B(n,\alpha)$, $^{6}Li(n,\alpha)$, $^{239}Pu(n,f)$, $^{197}Au(n,\gamma)$, and $^{58}Ni(n,p)$.
- Uncertainties in the low-energy spectrum tails (<10 keV) for all current reactor dosimetry benchmarks. More analytical work and sensitivity studies will be needed, as well as dedicated integral measurement comparisons with the Intermediate-Energy Standard Neutron Field (ISNF), to define the spectral shapes.
- Lack of a sufficient set of high-accuracy, redundant, interlaboratory microscopic integral cross section measurements; particularly for the californium-252 fission spectrum and in the intermediate-energy standard neutron fields ISNF and ΣΣ.
- Suggested uncertainties[19] regarding the shape of the ^{235}U thermal neutron induced fission neutron spectrum in the energy ranges of <250 keV, above 8 MeV and possibly also ∿ 2-6 MeV.

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