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PROJECTIONS OF ENDF/B VERSION V PERFORMANCE FOR FAST AND THERMAL REACTORS USING SENSITIVITY COEFFICIENTS*

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ABSTRACT

As part of our participation in the Cross Section Evaluation Working Group, the authors have generated, compiled, and documented an extensive set of sensitivity coefficients based upon ENDF/B-IV for use in conjunction with the data testing of the ENDF/B-V nuclear data file. The effort also included the application and testing of a computer retrievable sensitivity format (SENPRO) through the use of computer programs for editing, mode changing, and folding with projected changes to the nuclear data file.

Proposed reductions to $^{235}U(\bar{\nu})$ and $^{235}U(n,f)$ in the fast energy range have significant impact for uranium fueled fast critical assemblies. The long-standing LMFBR $^{28}c/^{49}f$ calculated overprediction is not resolved by proposed Version 5 cross section modifications for $^{238}U(n,\gamma)$ and $^{239}Pu(n,f)$. The upward evaluation for the $^{239}Pu(n,f)/^{235}U(n,f)$ ratio improves criticality predictions for Pu fueled fast assemblies. For thermal reactors, changes to the ^{238}U resonance parameters significantly reduce the long-standing $^{28}\rho$ discrepancy. Reduced resonance capture in the 1 eV ^{240}Pu resonance has significant implications for LWR fuel cycle studies.

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INTRODUCTION

As part of our participation in the Cross Section Evaluation Working Group, the authors have generated,[1-5] compiled,[1-11] and documented[12] an extensive set of sensitivity coefficients based upon ENDF/B-IV for use in conjunction with the data testing of the ENDF/B-V nuclear data file. The sensitivities were generated for many of the measured reactor parameters (i.e. k, reaction rate ratios, central worths, etc.) in several of the fast and thermal CSEWG benchmark assemblies. The effort also included the application and testing of a computer retrievable sensitivity format (SENPRO),[1] through the use of computer programs for editing,[13] mode changing,[13] folding with projected changes to the nuclear data file,[13] and nuclear data induced uncertainty determination.[14]

After some years of reevaluation, the preliminary ENDF/B-V file has recently been assembled and phase I review (checking of clerical errors, inconsistencies, etc.) initiated. The purpose of this paper is to describe the existing sensitivity coefficient library and associated computer programs, and the use of this library to project calculated reactor parameters for CSEWG benchmarks based upon proposed Version IV/Version V differences. Section I will describe the available sensitivity coefficients for fast reactors and Section II will describe those for thermal reactors. The available computer software is described briefly in Section III. Some of the important proposed data changes for U/Pu systems are described in Section IV. Finally, Sections V and VI indicate the assessment of such changes with regard to calculation of CSEWG fast and thermal reactor data testing benchmarks. It is important to note that, at this time, much of the ENDF/B-V file is still preliminary and subject to change before general release in approximately the Fall of 1978. Also, due to the broad scope of this effort, only illustrative examples are provided herein. More detailed documentation of this ongoing effort is planned prior to the ENDF/B-V release, as approved by the CSEWG Data Testing Subcommittee.

I. FAST REACTOR SENSITIVITY COEFFICIENTS

Cross section sensitivity coefficients were generated using ENDF/B-IV for CSEWG fast reactor benchmarks[15] ZPR-6/7, ZPR-6/6A, ZPR-3/48, ZPR-3/56B, ZPR-3/11, ZPR-9/31, as well as the fast metal spheres, GODIVA and JEZEBEL. The two principal codes used to generate these coefficients were FORSS[16] and VARI-1D.[17]

With this large volume of sensitivity data at hand, it is not practical to present graphs and tables of energy-dependent sensitivities for every assembly/response/nuclide/reaction type/Legendre moment/incident energy/and final energy. Nor have we found this to be necessary! To within a reasonable approximation, the energy-dependent sensitivities of ZPR-6/7, ZPR-6/6A, ZPR-3/48, ZPR-3/56B, ZPR-3/11, and ZPR-9/31 are similar and typified by those illustrated for ZPR-6/7. The degree of "reasonableness" of this approximation is such that energy-dependent sensitivities for different assemblies but identical responses, and comparable cross sections agree in many cases to within ∿ 30% for sensitivities whose absolute magnitudes are relatively large ($\geq$ 0.1), and within a factor of 3 for relatively small magnitudes ($\leq$ 0.01). One must examine the principal fissile species (e.g., ^{239}Pu and ^{235}U) and moderator (e.g., ^{16}O and ^{12}C when comparing sensitivities of assemblies with sharply different constituents. The similitude of sensitivity shapes can be used for applying available sensitivities of GODIVA to JEZEBEL by associating ^{235}U with ^{239}Pu, ^{238}U with ^{240}Pu, etc. It should be noted that this approximate similarity of the energy-dependent sensitivity coefficients for different assemblies, which has been employed here for convenience, has several glaring exceptions, notably the worth sensitivities (e.g., central Na worth) and sensitivities of some of the threshold reactions (e.g., ^{238}U inelastic and fission).

As a specific example we consider ZPR-6/7, which is a large (3100 liter) plutonium oxide fueled fast critical assembly with a soft spectrum and other characteristics representative of mixed Pu/U LMFBR designs. Table 1 summarizes integral sensitivities of several measured responses to the important reaction types. Only those integral sensitivities are listed whose absolute magnitude exceeds .02. The integral sensitivity coefficient is the percent change in the performance parameter resulting from a uniform percent change in the cross section over all energy. The calculated and calculation/experiment values given in Table 1 are nominal results (based on ENDF/B-IV, without requiring criticality) intended only to give the

Table 1. Integral Sensitivities of ZPR-6/7 Performance Parameters to Various Cross Section Reaction Types[1]

k Calculated .9844 Calculation/Experiment .9844		$^{28}c/^{49}f$ (Central) Calculated 0.1529 Calculation/Experiment 1.092		$^{25}f/^{49}f$ (Central) Calculated 0.0[illegible]67 Calculation/Experiment .971		$^{28}f/^{49}f$ (Central) Calculated 1.1099 Calculation/Experiment 1.033	
Reaction	Integral Sensitivity	Reaction	Integral Sensitivity	Reaction	Integral Sensitivity	Reaction	Integral Sensitivity
^{239}Pu($\bar{\nu}$):	+0.518	^{239}Pu(n,f):	-1.073	^{235}U(n,f):	+0.960	^{239}Pu(n,f):	-1.063
^{239}Pu(n,f):	+0.591	^{238}U(n,γ):	+0.896	^{239}Pu(n,f):	-0.761	^{238}U(n,f):	+0.994
^{238}U(n,γ):	-0.239	O(n,n):	+0.109	^{238}U(n,γ):	+0.271	O(n,n):	+0.109
^{238}U($\bar{\nu}$):	+0.126	^{238}U(n,n'):	+0.068	^{238}U(n,n'):	-0.248	^{238}U(n,γ):	-0.103
^{238}U(n,f):	+0.079	Na(n,n):	+0.025	Fe(n,n'):	-0.199	Na(n,n):	+0.053
^{239}Pu(n,γ):	-0.0[illegible]	Fe(n,n'):	+0.025	O(n,n):	-0.089	^{238}U(n,n'):	+0.050
^{238}U(n,n'):	-0.042	Fe(n,n):	+0.024	^{239}Pu(n,γ):	+0.079	^{239}Pu(n,γ):	-0.036
Fe(n,n'):	-0.024	^{239}Pu(n,γ):	-0.020	Na(n,n'):	-0.07[illegible]	Fe(n,n):	+0.030
^{240}Pu($\bar{\nu}$):	+0.023			Na(n,n):	-0.048		
Fe(n,γ):	-0.020			Fe(n,n):	-0.044		
				Fe(n,γ):	+0.024		

[1]Calculated and C/E values listed are for the microscopic reaction rates in the heterogeneous system, including appropriate corrections. The sensitivity profiles are based on a homogeneous model. Only those integral sensitivities are listed whose absolute magnitude exceeds .02.

reader an overall sense of whether the integral response parameter is currently well calculated (see for example the ∿ 9% calculated overprediction of the central ^{238}U capture/^{239}Pu fission ratio).

Table 1 shows the expected high sensitivities to the ^{239}Pu(n,f) and ^{238}U(n,γ) cross sections. The sensitivity to a cross section that appears in the definition of a particular ratio response is also high (e.g., the ^{238}U(n,f) cross section in the (^{28}f/^{49}f) ratio). This is a result of the large direct-effect contribution associated with a cross section which appears in a ratio response definition. Note should also be made of the small but non-negligible sensitivities (∿ 0.1) for iron, oxygen, and sodium. These sensitivity coefficients are the result of indirect flux effect terms only, and are therefore smaller than the direct-effect components; nevertheless, they are important.

More detailed information can be obtained about the sensitivity of the performance parameters to cross section data by looking at the energy-dependent sensitivities for each nuclide. Figure 1 presents the relative

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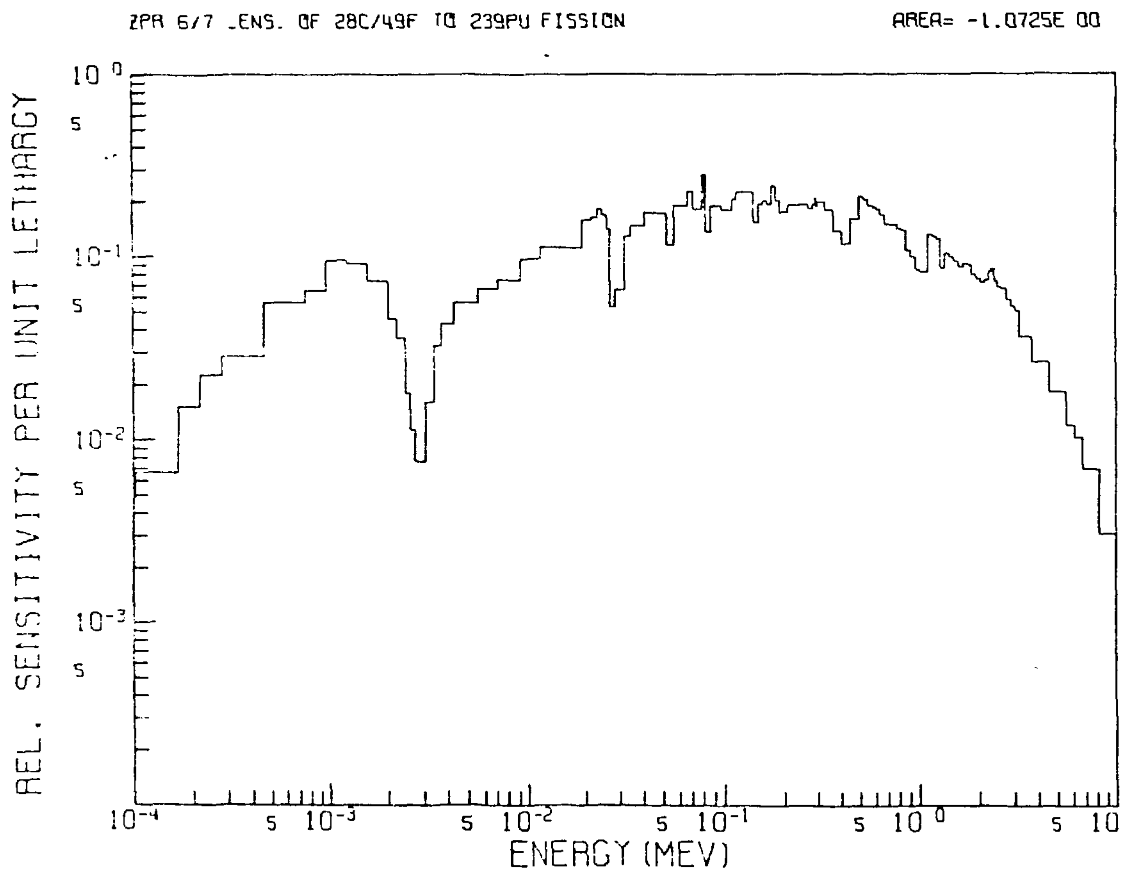

Figure 1. Sensitivity of ^{238}U Capture/^{239}Pu Fission Central Reaction Ratio in Assembly ZPR-6/7 to the ^{239}Pu Fission Cross Section.

sensitivity per unit lethargy of the central ^{238}U capture/^{239}Pu fission ratio in ZPR-6/7 with respect to the ^{239}Pu fission cross section. The library of fast reactor energy dependent sensitivity coefficients described in Reference 12 is available in a documented computer retrievable format[1] through the Radiation Shielding Information Center at ORNL and the National Nuclear Data Center at BNL.

II. THERMAL REACTOR SENSITIVITY COEFFICIENTS

Cross section sensitivities are currently available for TRX-1 and 2,[15] MIT-1,[15] and UL-212[18] (the latter is currently being considered for benchmark status). The TRX criticals are rod lattices of 1.3% (^{235}U) enriched uranium metal rods in water, MIT-1 is a natural uranium metal rod lattice in heavy water, and UL-212 is a mixed oxide lattice in water.

The commonly measured integral parameters are (for a U235-U238 system):

$^{28}\rho$ = ratio of epithermal-to-thermal U238 captures
$^{25}\delta$ = ratio of epithermal-to-thermal U235 fissions
$^{28}\delta$ = ratio of U238 fissions to U235 fissions
C^* = ratio of U238 captures to U235 fissions (modified conversion ratio)
k_{eff} = effective multiplication factor
(Thermal cut energy = .625 eV)

In addition to the detailed perturbation theory approach,[5,19] several studies[4,8,9] in which specific data variations were made provide valuable information for physics characterization of these assemblies. This information describes the sensitivities of lattice performance parameters to ^{238}U, ^{235}U, ^{239}Pu, ^{240}Pu, H, O, and Al cross sections and resonance parameters. The work of Hardy[4] demonstrated clearly the large sensitivities for the ^{235}U capture and ^{235}U fission integrals, particularly the unshielded components of the capture.

Rothenstein,[9] Finch[8] and Tomlinson[5] have computed sensitivities to resonance parameters. As an example, Finch[8] has obtained the sensitivity of integral parameters in TRX-2 to the variation of Γ_n and Γ_γ for the first seven resonances of ^{238}U. These are listed below in Table 2.

Finally, an extensive set of energy dependent coefficients have been computed for TRX-2[5] and UL-212[19] respectively. In TRX-2, for example, sensitivities were obtained in 131 energy groups for k, $^{28}\rho$, $^{25}\delta$, $^{28}\delta$ and C^* with respect to the following nuclear parameters: σ_f, σ_c, σ_e, and ν of ^{235}U; σ_s, σ_c, σ_f, $\bar{\nu}$, σ_{in} (by level), and individual resonance parameters of ^{238}U; σ_s and σ_c for H, σ_s and σ_c for ^{16}O, and σ_s and σ_c for Al. Table 3 lists group sensitivities (collapsed to four groups for ease of reading) for $^{28}\rho$ as computed by Tomlinson[5].

Table 2. TRX-2 Sensitivity Coefficients $\frac{dP/P}{d\sigma/\sigma}$

^{238}U Resonance Energy (ev)	Resonance Parameter Varied	k Calculated .9894 Calculation/ Experiment .9894	$^{28}\rho$ Calculated 0.867 Calculation/ Experiment 1.036	$^{25}\delta$ Calculated 0.060 Calculation/ Experiment 0.980	$^{28}\delta$ Calculated .069 Calculation/ Experiment 0.990	C* Calculated 0.648 Calculation/ Experiment 1.002
6.67	Γ_n	-1.7×10^{-2}	1.18×10^{-1}	8.3×10^{-3}	3.49×10^{-2}	5.33×10^{-2}
	Γ_γ	-1.6×10^{-2}	1.17×10^{-1}	7.3×10^{-3}	3.49×10^{-2}	5.26×10^{-2}
[illegible]	Γ_n	[illegible]	5.01×10^{-2}	[illegible]	2.9×10^{-2}	2.2×10^{-2}
	Γ_γ	[illegible]	4.87×10^{-2}	5.9×10^{-3}	2.9×10^{-2}	2.2×10^{-2}
36.8	Γ_n	-6.2×10^{-3}	[illegible]	4.9×10^{-3}	2.7×10^{-2}	1.4×10^{-2}
	Γ_γ	-7.2×10^{-3}	4.03×10^{-2}	4.9×10^{-3}	2.8×10^{-2}	1.8×10^{-2}
66.15	Γ_n	-2.2×10^{-3}	2.0×10^{-3}	1.0×10^{-2}	2.8×10^{-2}	3×10^{-4}
	Γ_γ	-3.0×10^{-3}	8.6×10^{-3}	9.0×10^{-3}	2.9×10^{-2}	3.1×10^{-3}
80.74	Γ_n	-3.0×10^{-4}	-9.8×10^{-3}	8.7×10^{-3}	2.7×10^{-2}	-5.0×10^{-3}
	Γ_γ	-6.0×10^{-4}	-1.1×10^{-2}	8.7×10^{-3}	2.7×10^{-2}	-9.1×10^{-3}
102.5	Γ_n	-1.6×10^{-3}	-1.2×10^{-3}	8.7×10^{-3}	2.8×10^{-2}	-1.0×10^{-3}
	Γ_γ	-3.0×10^{-3}	8.9×10^{-3}	8.0×10^{-3}	2.9×10^{-2}	3.4×10^{-3}
116.8	Γ_n	-1.0×10^{-3}	-8.4×10^{-3}	8.7×10^{-3}	2.7×10^{-2}	-4.4×10^{-3}
	Γ_γ	-1.8×10^{-3}	-2.0×10^{-3}	8.0×10^{-3}	2.8×10^{-2}	-2.0×10^{-3}

III. MANIPULATION AND HANDLING TECHNIQUES FOR SENSITIVITY COEFFICIENTS

SENDIN[13] and SENTINEL[13] are two codes written to help assess the effects of nuclear data changes. SENDIN is used to convert unformatted (binary) sensitivity files in SENPRO format[1,13] to card image form and vice versa. This is particularly useful for transferring sensitivity files from one installation to another. SENTINEL is used to calculate the percent change in a specified response due to given percent changes in specified reaction cross sections over selected energy regions. An edited list of the most significant individual contributions to the response change is also provided. COVERT[14] and CAVALIER[14] are two additional codes written for estimating uncertainties in reactor performance parameters using SENPRO sensivitity and COVERX covariance files. All codes are available from the

Table 1. Sensitivities for [illegible] in the
TRX-2 Thermal Lattice

		$dR/R/d\Sigma/\Sigma$			
Nuclide	Item	Group 1[a]	Group 2[b]	Group 3[c]	Group 4[d]
H	σ_s	-0.084	-0.115	-0.845	0.010
^{235}U	σ_f	0.00002	0.00002	0.009	0.540
H	σ_c	0.0	0.0	0.005	0.094
^{238}U	σ_c	0.0	0.0	0.002	0.177
Moderator	DB^2	0.001	0.001	0.010	0.017
^{238}U	σ_c	0.102	0.114	0.563	-0.802
O	σ_s	-0.002	-0.001	-0.008	0.0009
Al	σ_c	0.0	0.0	0.0002	0.008
^{238}U	σ_s	-0.002	-0.00007	0.005	0.002
Fuel	DB^2	[illegible]	0.00006	0.0005	0.001
Clad	DB^2	[illegible]	[illegible]	[illegible]	[illegible]
Void	DB^2	0.00001	0.00002	0.0002	0.0004
Al	σ_s	0.0	-0.00003	-0.0007	0.0002
^{235}U	ν	0.0	0.0	-0.00001	-0.0003
^{238}U	σ_f	-0.0001	0.0	0.0	0.0
O	σ_c	0.0	0.0	0.0	0.0005
^{235}U	σ_s	-0.00001	0.0	-0.00004	0.00004
^{238}U	$\bar{\nu}$	-0.00002	0.0	0.0	0.0

[a] 10 MeV - 67.37 keV.
[b] 67.37 keV - 3.35 keV.
[c] 3.35 keV - 0.625 eV.
[d] 0.625 eV - 10^{-5} eV.

Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory.

For those cases in which sensitivities are available with respect to specific resonance parameters (see for example Table 2), assessment of proposed changes to the parameters is a relatively simple multiplication process which, to date, has been done by hand. (Modifications to the SENPRO format to include sensitivities to resonance parameters are under consideration.) However, in general, sensitivities are available in a large number of groups and for many reaction types, integral parameters and assemblies. Thus, it was important to generate computer programs to facilitate the handling of this data.

IV. ANTICIPATED NEUTRON CROSS SECTION MODIFICATIONS FOR URANIUM AND PLUTONIUM SYSTEMS

The cross section modifications for all fast and thermal reactor materials are too numerous to present here. Instead, proposed modifications for specific reactions in ^{238}U and ^{239}Pu will be discussed. Table 4 lists proposed changes [(Version 5 - Version 4)/Version 4] in percent based upon processed cross sections from ENDF/B-IV[20] and from an early version (pre-preliminary) of Version 5 data.

Table 4. Possible Cross Section Modifications from ENDF/B-4 to ENDF/B-5

GRP	UPPER ENERGY	U-238(N,G)	PU-239(N,F)	NU BAR 239	U-235(N,G)	U-235(N,F)	NU BAR 235
1	1.73E 07	27.12	-1.76	0.33	1.98	1.35	-0.86
2	1.00E 07	14.80	-0.18	0.95	2.24	2.97	-1.17
3	6.07E 06	-3.06	1.47	1.33	0.73	0.52	-1.16
4	3.68E 06	-17.63	0.36	0.73	-0.60	-0.58	-1.39
5	2.23E 06	-16.14	1.58	0.16	0.90	0.03	-1.53
6	1.35E 06	-1.66	2.43	-0.10	-1.96	-2.09	-1.77
7	8.21E 05	0.63	-0.66	-0.20	-3.11	-3.14	-0.58
8	4.98E 05	-0.77	-1.22	-0.25	-2.77	-2.76	-0.27
9	3.02E 05	-5.46	0.47	-0.28	-3.86	-3.65	-0.04
10	1.83E 05	-2.48	0.52	-0.30	-2.31	-2.37	0.14
11	1.11E 05	5.73	-1.18	-0.31	-2.32	-2.32	0.12
12	6.74E 04	1.26	0.94	-0.26	-2.33	-2.33	0.16
13	4.09E 04	4.64	0.42	-0.23	-2.27	-2.31	0.14
14	2.48E 04	-0.06	1.46	-0.21	1.03	-0.49	0.16
15	1.50E 04	1.99	1.74	-0.19	-1.12	-0.99	0.16

Finally, 10 - 20% reductions to the ^{232}Th capture cross section in the fast reactor energy range are being proposed.

For thermal reactor systems, a most important proposed change is the reduction of the capture widths of the first three levels of ^{238}U (by about 15%) and increased neutron widths for the 20.9 and 36 eV levels ($\sim$ 10%). Furthermore, the ^{240}Pu(n,γ) resonance at 1 eV is being reduced, and the ^{239}Pu($\bar{\nu}$) representation is being changed from energy independent to .1 - .5% increases below .3 eV and .2% decrease above.

V. IMPACT OF VERSION V DATA ON CSEWG FAST REACTOR BENCHMARKS

Projected modifications to ENDF/B-IV (see Table 4) have been folded with sensitivity coefficients to estimate the impact on CSEWG fast reactor benchmark performance parameters. Some of this work has, in fact, been done earlier[19,3] to provide initial guidance and feedback to the evaluators. Estimates were verified by recalculation of the problem with the new cross section set.

Reevaluation of ^{235}U($\bar{\nu}$) experimental data coupled with renormalization of ^{252}Cf($\bar{\nu}$) for ENDF/B-V leads to a reduction in ^{235}U($\bar{\nu}$) of the order of

1-2% above 1 MeV, and .3-1% between 300 keV and 1 MeV. This factor combined with proposed 2-3% reductions to the fast $^{235}U(n,f)$ cross section, and ignoring other changes, has significant consequences for uranium fueled fast assemblies as shown in Table 5.

Table 5. The Impact of Proposed Changes to ^{235}U Fission and $\bar{\nu}$ is Substantial for Fast Reactor Calculations.

INTEGRAL PARAMETER	CALCULATION / EXPERIMENT	PERCENT CHANGE IN PERFORMANCE DUE TO	
		$^{235}U(n,f)$	$^{235}U(\bar{\nu})$
GODIVA			
k	1.0040	- .97	- .98
$^{28}f/^{25}f$	1.033	.92	- .06
$^{49}f/^{25}f$	.972	1.36	-
$^{28}c/^{25}f$	.942	1.01	.08
ZPR-6/6A			
k	.985	-1.08	- .93
$^{28}c/^{49}f$	1.023	2.09	-
$^{28}f/^{25}f$	.923	1.21	-
ZPR-9/31			
k	.9583	- .02	-
$^{25}f/^{49}f$	.953	-2.00	-

Criticality predictions for GODIVA and ZPR-6/6A (fast and LMFBR uranium assemblies) are significantly lowered resulting in discrepancies with integral data. The direct impact on plutonium-fueled assemblies (e.g. ZPR-9/31) is clearly less glaring. However, the reductions to $^{235}U(n,f)$ impact directly the measured reaction rate ratios relative to $^{235}U(n,f)$ alleviating discrepancies in some cases, and further aggravating them in others. The $^{25}f/^{49}f$ ratio in ZPR-9/31, already $\sim$ 4% low, becomes 6% low under these conditions, some 3-5 standard deviations away in terms of the integral experiment uncertainty.

The result of proposed changes to $^{238}U(n,\gamma)$ and $^{239}Pu(n,f)$ are presented in Table 6. The long-standing discrepancy for the overprediction of $^{28}c/^{49}f$ (see Table 1) is not helped by currently proposed changes to Version 5 data. The upward evaluation for the $^{239}Pu(n,f)/^{235}U(n,f)$ ratio improves criticality predictions for Pu fueled fast assemblies.

Table 6. The $^{238}U(n,\gamma)$ and $^{239}Pu(n,f)$ Cross Section Changes Will Also Impact Fast Reactor Calculations.

INTEGRAL PARAMETERS	CALCULATION / EXPERIMENT	PERCENT CHANGE IN PERFORMANCE DUE TO	
		$^{238}U(n,\gamma)$	$^{239}Pu(n,f)$
GODIVA			
k	1.0040	-	
$^{49}f/^{25}f$	1.033	-	.93
$^{28}c/^{25}f$	.942	-3.35	
JEZEBEL			
k	.9920		.71
[illegible]	.905		.09
[illegible]	.933		.07
ZPR-6/7			
k	.9[illegible]4	-.10	.30
$^{28}c/^{49}f$	1.092	.41	-.5[illegible]
[illegible]	.971	.19	-.39

The ^{238}U total inelastic scattering cross section for Version 5 is significantly larger than that of Version 4 over wide energy ranges. However, the newer evaluation actually leads to smaller energy transfers due primarily to reductions in the continuum scattering cross section[21] (cross sections for lower lying levels were increased). Assessment of such changes via sensitivity coefficients was not undertaken since the evaluation had changed in form (different number of levels, etc.). The shape of the fission spectrum was also changed from a Maxwellian to an energy-dependent Watt spectrum resulting in more neutrons at intermediate energies and fewer neutrons at low and high energies. Kujawski[22] has determined the sensitivity of k in ZPR-6/7 to the change in the fission spectrum shape to be ~ .3%.

Finally, McKnight[23] has shown that the changes to the $^{232}Th(n,\gamma)$ cross section are such as to increase the eigenvalue for ^{233}U-^{232}Th fast systems (no ^{238}U) by ~ 4.8% and to reduce the predicted breeding ratio by ~ 11%. These results are considerably alleviated by the necessary reduction of the ^{233}U fissile inventory which is required to keep the reactor critical.

VI. IMPACT OF VERSION V DATA ON CSEWG THERMAL REACTOR BENCHMARKS

The impact of the reduction of the capture widths of the first three s-wave levels of ^{238}U, and the increased neutron widths of the levels at

20.9 eV and 36.8 eV are given below in Table 7. These changes by themselves would appear to remove much of the existing discrepancy for $^{28}\rho$. However, it is important to recall that final projections for the integral quantity must include other changes (other resonances, fission spectrum, changes to the inelastic cross sections).

Table 7. The Reduction of the Capture Widths of the First Three Levels of ^{238}U (~ 15%) and Increased Neutron Widths for the 20.9 and 36.8 eV Levels (~ 10%) Are Important for Thermal Benchmarking.

^{238}U Resonance Energy (eV)	Resonance Parameter Varied	k Calculated .9894 Calculation/ Experiment .9894	$^{28}\rho$ Calculated 0.867 Calculation/ Experiment 1.036	$^{25}\delta$ Calculated 0.060 Calculation/ Experiment 0.989	$^{28}\delta$ Calculated .069 Calculation/ Experiment 0.990	C* Calculated 0.648 Calculation/ Experiment 1.002
6.67	Γ_n	-0.0121	0.0998	0.0027	0.0087	0.0462
	Γ_γ	0.215	-1.7?92	[illegible]	-0.2239	-0.8160
20.9	Γ_n	-0.1290	1.0209	-0.0105	0.1350	0.4650
	Γ_γ	0.1112	-0.9174	0.0097	-0.1251	-0.4309
36.8	Γ_n	-0.0542	0.4972	-0.0131	0.0633	0.2350
	Γ_γ	0.0708	-0.6490	0.0236	-0.0826	-0.3068
Total Change		.22	-1.75	- .05	- .25	- .32

Introduction of proposed energy dependence for ^{239}Pu ($\bar{\nu}$) increases the eigenvalue in UL-212 by ~ .1%. In addition, changes to the 1 eV resonance capture in ^{240}Pu reduces the $^{40}c/^{25}f$ ratio in UL-212 by ~ 2%. It currently appears that the shape of the cross section between .0253 and .5 eV does not agree with the Version 5 resonance parameters; the cross section should actually be higher. Since this inconsistency could have important implications for thermal reactor fuel cycle analysis a proposed revision is being suggested for consideration by the Data Testing Subcommittee.

CONCLUSIONS

The extensive effort made to generate, compile, and document sensitivity coefficients for fast and thermal reactor CSEWG benchmarks has been extremely valuable in providing rapid feedback to the data testing process. The computer software and the library developed in conjunction with this effort are available from RSIC and NNDC.

Based upon an early version of ENDF/B-5 (pre-preliminary) we find that proposed reductions to $^{235}U(\bar{\nu})$ and $^{235}U(n,f)$ in the fast energy range have significant import for uranium fueled fast critical assemblies. The long-standing LMFBR $^{28}c/^{49}f$ calculated overprediction is not resolved by proposed Version 5 cross section modifications for $^{238}U(n,\gamma)$ and $^{239}Pu(n,f)$. The upward evaluation for the $^{239}Pu(n,f)/^{235}U(n,f)$ ratio improves criticality predictions for Pu fueled fast assemblies. Changes to the $^{232}Th(n,\gamma)$ cross section are such as to increase the eigenvalue for ^{233}U-^{232}Th fast systems (no ^{238}U) by $\sim$ 4.8% and to reduce the predicted breeding ratio by $\sim$ 11%. These changes are considerably diminished by the necessary reduction of the ^{233}U fissile inventory which is required to keep the reactor critical.

For thermal reactors, changes to the ^{238}U resonance parameters significantly reduce the long-standing $^{28}\rho$ discrepancy. The shape of the ^{240}Pu capture cross section between .0253 and .5 eV does not agree with the newly evaluated resonance parameters; the cross section should actually be higher. Since such an inconsistency could have significant implications for fuel cycle studies, a proposed revision is being suggested for consideration by the Data Testing Subcommittee.

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