Retroactive Covariance Matrix for ^{235}U in the Resolved-Resonance Region

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Abstract

The resonance-parameter covariance matrix (CM) in the resolved resonance region for ^{235}U was generated using the retroactive approach of the SAMMY code [1]. As reported elsewhere [2], a Reich-Moore resonance evaluation for ^{235}U has been performed in the energy range between 0 and 2.25 keV using SAMMY. A total of 3165 resonances, including the external levels, were identified. At the time the evaluation was performed, the complete resonance-parameter CM was not generated because of computer memory and storage limitation [13]. A procedure for retroactively creating resonance-parameter CMs has been developed and implemented in the code SAMMY. This procedure is used to generate the CM for the ^{235}U parameters of the ENDF/B-VI evaluation.

KEYWORDS: ^{235}U resonance parameter retroactive covariance matrix

1. INTRODUCTION

In the Reich-Moore formalism each resonance of ^{235}U is described by five parameters (resonance energy, gamma width, neutron width, and two channel fission widths), for a total of 15,825 parameters. [1] The large number of resonance parameters leads to two major issues when generating a resonance covariance matrix (CM): (a) the computer memory required to process the data and (b) the data storage for the resulting covariance file. [13]

The former has been addressed by using a computer with 32 GB of memory. To address the latter issue, a compact format has been used for reporting the CM. The full CM generated with SAMMY requires a file of approximately 2 GB. Conversion of the ^{235}U CM into the compact format produces a file of 2.4 MB. This reduction in storage space will permit the processing of SAMMY-generated covariance data and the subsequent generation of CMs for groupwise cross sections.[2] The retroactive procedure in SAMMY requires that the updated set of resonance parameters obtained after a single Bayes iteration be nearly identical to the original set. To fulfill this requirement the experimental data in each data set listed in Table 1 were replaced by the theoretical values obtained from a no-Bayes SAMMY run on the evaluated resonance parameters. We preserved the experimental uncertainty from the original experiment as the uncertainty on the theoretical values.

For each data set created in the above fashion, the components of the off-diagonal data CM are computed, including energy-dependent statistical uncertainties from the

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[2] We will address the effect of the compact CM format on the cross section uncertainties in the full paper.

measurement and systematic uncertainties in normalization and background. Simultaneous fitting of all the data sets is then accomplished via SAMMY, properly including the effect of the data CMs. The set of experimental data used in the retroactive covariance evaluation of ^{235}U is shown in Table 1.

2. RESULTS

Normalization and background corrections for the data listed in Table 1 were taken from Ref. 2. Statistical errors on the experimental data points, together with systematic errors on the normalization and background, were combined in SAMMY to produce a CM for the ^{235}U resonance parameters in the resolved-resonance region.

Average total, capture, and fission cross sections and their corresponding uncertainties have been calculated for the 30 low-energy groups in the 44-group structure of the SCALE system (Ref. 12), spanning the entire resolved-resonance energy region of ^{235}U. The calculations, including the effect of the resonance-parameter CM, were performed with SAMMY. Results are shown in Table 2.

For the first time, a CM for ^{235}U has been generated for the resolved-resonance region. The evaluated CM will be submitted for inclusion in the ENDF library.

Table 1: Experimental data sets used in the ^{235}U CM evaluation

Reference	Energy range (eV)	Data description Refs. 3-11
Harvey [3]	0.4 – 68.0	Transmission at 18 m 0.03269 atom/barn 77 K
Harvey [3]	4.0 – 2250.0	Transmission at 80 m 0.00233 atom/barn 77 K
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Schrack [4]	0.02 – 20.0	Fission at 18 m
de Saussure [5]	0.01 – 2250.0	Fission at 25.2 m
de Saussure [5]	0.01 – 2250.0	Capture at 25.2 m
Perez [6]	0.01 – 100.0	Fission at 39.7 m
Perez [6]	0.01 – 100.0	Capture at 39.7 m
Gwin [7]	0.01 – 20.0	Fission at 18.9 m
Spencer [8]	0.01 – 1.0	Transmission at 80 m 0.001468 atom/barn 77 K
Wagemans [9]	0.001 – 4.0	Fission at 18 m
Gwin [10]	0.01 – 4.0	Absorption
Gwin [10]	0.01 – 4.0	Fission
Weston [11]	14.0 – 2250.0	Fission at 18.9 m

Table 2: Average cross section (barn) and uncertainty calculated with SAMMY including resonance CM

Energy range (eV)		Total cross section	Uncertainty in total cross section	Fission cross section	Uncertainty in fission cross section	Capture cross section	Uncertainty in capture cross section
0.00001	0.003	4078	12.01	3414	9.289	647.8	10.2
0.003	0.008	1651	3.704	1377	2.572	258.9	3.21
0.008	0.010	1238	2.28	1032	1.724	190.8	1.875
0.010	0.025	874.1	1.322	730.3	1.098	128.6	0.9479
0.025	0.030	666.2	0.9501	557.8	0.8011	93.33	0.6511
0.030	0.040	584.5	0.8405	489.2	0.7181	80.29	0.5985
0.040	0.050	503.8	0.7479	420.7	0.6441	68.04	0.5587
0.050	0.070	425.7	0.6695	353.8	0.5575	56.99	0.4929
0.070	0.100	343	0.5936	282	0.4639	46.14	0.4062
0.100	0.150	271.7	0.5506	218.9	0.3967	38.08	0.339
0.150	0.200	231.8	0.5418	181.5	0.3685	35.85	0.3135
0.200	0.225	228.2	0.5676	175.1	0.4035	38.75	0.3752
0.225	0.250	238.1	0.5952	181.2	0.4468	42.61	0.4532
0.250	0.275	250.9	0.6448	190.7	0.4987	45.92	0.5209
0.275	0.325	245	0.6622	188	0.4783	42.69	0.4926
0.325	0.350	208.9	0.6046	162.2	0.4065	32.34	0.3819
0.350	0.375	181.5	0.523	141.6	0.3674	25.64	0.3083
0.375	0.400	158.9	0.4476	124.1	0.3325	20.61	0.2586
0.400	0.625	108	0.3128	82.94	0.2255	11.14	0.1863
0.625	1.000	78.79	0.2637	58.23	0.1815	7.169	0.1694
1.000	1.770	61.38	0.2508	38.12	0.1512	10.28	0.1789
1.770	3.000	33.35	0.2147	13.85	0.1085	7.318	0.1182
3.000	4.750	40.22	0.2293	18.57	0.1126	10.09	0.1402
4.750	6.000	43.47	0.2485	13.22	0.1218	19.31	0.1831
6.000	8.100	79.55	0.2953	24.47	0.1615	44.13	0.2656
8.100	10.000	171.7	0.4627	116.9	0.4115	43.02	0.4276
10.000	30.000	88.27	0.2436	43.62	0.1332	32.93	0.1663
30.000	100.000	To be submitted in the full paper by June 15, 2005					
100.000	550.000						
550.000	2250.000						

REFERENCES

1. N. M. Larson, *Updated Users' Guide for SAMMY: Multilevel R-Matrix Fits to Neutron Data Using Bayes' Equations*, ORNL/TM-9179/R6, Oak Ridge National Laboratory, Oak Ridge, TN, July 2003.

2. L. C. Leal, H. Derrien, N. M. Larson, and R. Q. Wright, *R-Matrix Analysis of ^{235}U Neutron Transmission and Cross Sections in the Energy Range 0 to 2.25 keV*, ORNL/TM-13516, Oak Ridge National Laboratory, Oak Ridge, TN, November 1997.

3. J. A. Harvey, N. W. Hill, F. G. Perey, G. L. Tweed, and L. C. Leal, "High Resolution Transmission Measurements on ^{235}U, ^{239}Pu, and ^{238}U," in *Proc. Int. Conf. on Nuclear Data for Science and Technology*, Mito, Japan, May 30-June 3, 1988, p. 115 (1988).

4. R. A. Schrack, "Measurement of the $^{235}U(n,f)$ Reaction from Thermal to 1 keV," in *Proc. Int. Conf. on Nuclear Data for Science and Technology*, Mito, Japan, May 30-June 3, 1988, p. 101 (1988).

5. G. de Saussure, R. Gwin, L. W. Weston, and R. W. Ingle, *Simultaneous Measurements of the Neutron Fission and Capture Cross Section for ^{235}U for Incident Neutron Energy from 0.4 eV to 3 keV*, ORNL/TM-1804, Oak Ridge National Laboratory, Oak Ridge, TN, 1967.

6. R. B. Perez, G. de Saussure, and E. G. Silver, *Nucl. Sci. Eng.* **52**, 46 (1973).

7. R. Gwin, R. R. Spencer, R. W. Ingle, J. H. Todd, and S. W. Scoles, *Nuc. Sci. Eng.* **88**, 37 (1984).

8. R. R. Spencer, J. A. Harvey, N. W. Hill, and L. Weston, *Nucl. Sci. Eng.* **96**, 318 (1987).

9. C. Wagemans, P. Schillebeeckx, A. J. Deruyter, and R. Barthelemy, "Subthermal fission Cross Section measurements for ^{233}U and ^{239}Pu," in *Proc. Int. Conf. on Nuclear Data for Science and Technology*, Mito, Japan, May 30-June 3, 1988, p. 91 (1988).

10. R. Gwin, unpublished.

11. L. W. Weston and J. H. Todd, *Nucl. Sci. Eng.* **88**, 567 (1984).

12. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, NUREG/CR-0200, Rev. 7 (ORNL/NUREG/CSD-2/R7), vols. 1-3, (May 2004) (draft). Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-725.

13. H. Derrien, L. C. Leal, and N. M. Larson, *Covariance Matrices for Use in Criticality Safety Predictability Studies*, ORNL/TM-13492, Oak Ridge National Laboratory, Oak Ridge, TN, September 1997.