Issues with ENDF/B-VII.1 Radioactive Decay Data

Jesse C. Holmes
Bettis Atomic Power Laboratory

Cross Section Evaluation Working Group (CSEWG) Meeting

BNL, November 2 - 4, 2015

Overview

- Sub-Library 4 (NSUB=4) contains Radioactive Decay Data.
- File 8 gives decay and fission product yields. MT=457 is Radioactive Decay Data.
- For spontaneous fission (SF), neutron emission energy spectra are given in File 5. Currently, the only spontaneous fission neutron emission spectrum available in ENDF/B-VII.1 is for Cf-252.
- JEFF-3.1.1 has spontaneous fission neutron emission spectra available for several nuclides produced through systematics.
- Experimentally determined Watt parameters are published for several nuclides from which the SF neutron spectra can be generated.
- File 8, MT=457 continuous gamma and neutron emission spectra are given for several nuclides in ENDF/B-VII.1 where the phenomena are discrete and discrete data is available from ENSDF or other databases.
- The corresponding JEFF-3.1.1 discrete spectra are consistent with discrete data in ENSDF and other referenced databases. In many cases, the ENDF/B-VII.1 continuous spectra are inconsistent with this information.

Spontaneous Fission

The neutron energy spectra from spontaneous fission is often represented using the Watt spectrum in neutron transport codes. The Watt spectrum is defined using the given analytical function:

E is the secondary neutron exit energy (MeV), and a and b are the Watt spectrum coefficients. Recommended $\bar{\nu}$, a, and b values for SF from several sources are available.

<u>Watt Spectrum</u>

$$p(E) = Ce^{-E/a}\sinh\left(\sqrt{bE}\right)$$

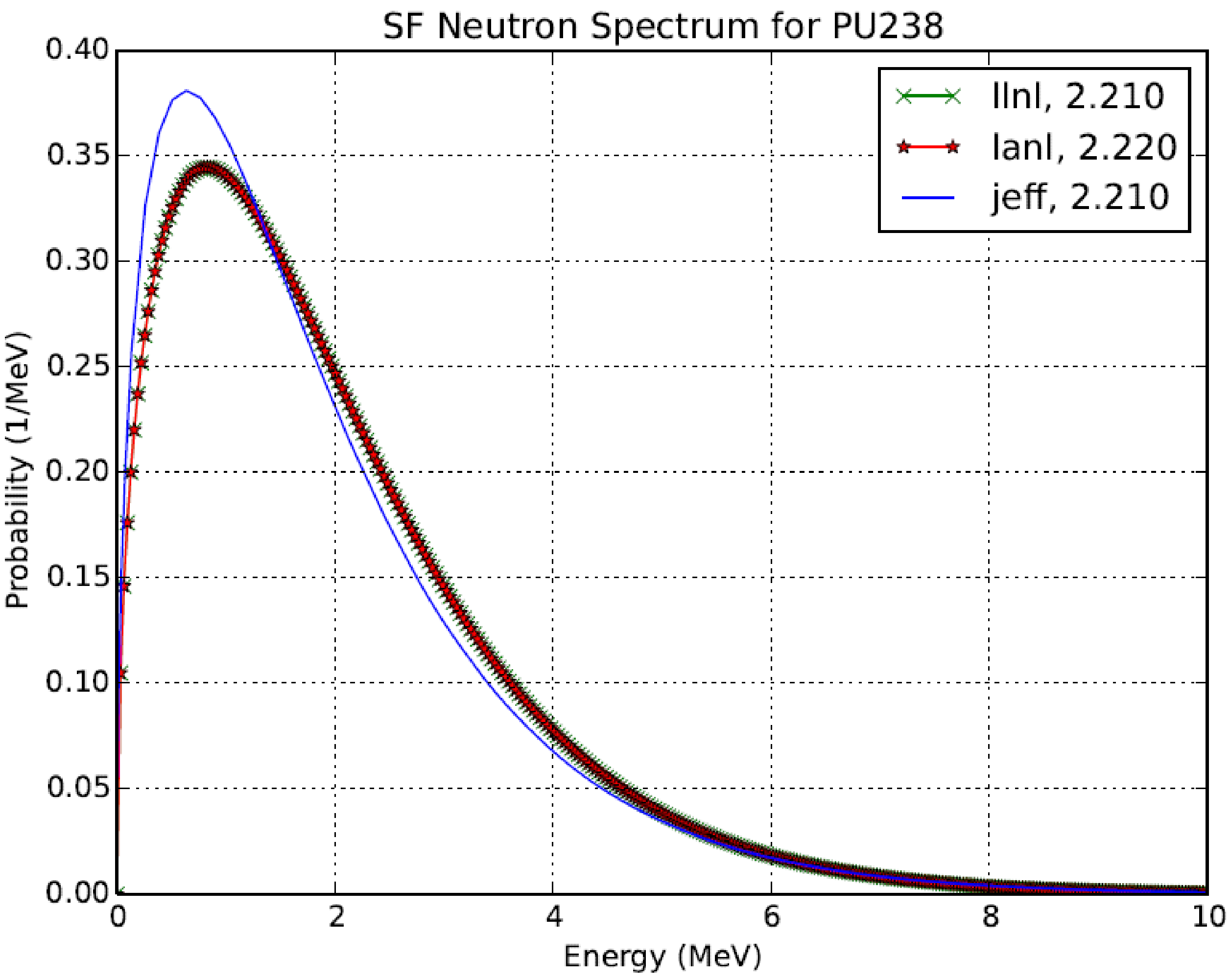
$$C = \frac{1}{2}\sqrt{\pi a^3 b}\, e^{\frac{ab}{4}}$$

1. J. M. Verbeke, C. Hagmann, and D. Wright, "Simulation of Neutron and Gamma Ray Emission from Fission and Photofission," Lawrence Livermore National Laboratory, UCRL-AR-228518, October 15, 2010.
2. E. F. Shores, "Data Updates for the SOURCES-4A Computer Code," Los Alamos National Laboratory, LA-UR-00-5016, October 18. 2000.
3. X-5 Monte Carlo Team, "MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, Volume I: Overview and Theory," Los Alamos National Laboratory, LA-UR-03-1987, April 24, 2003.

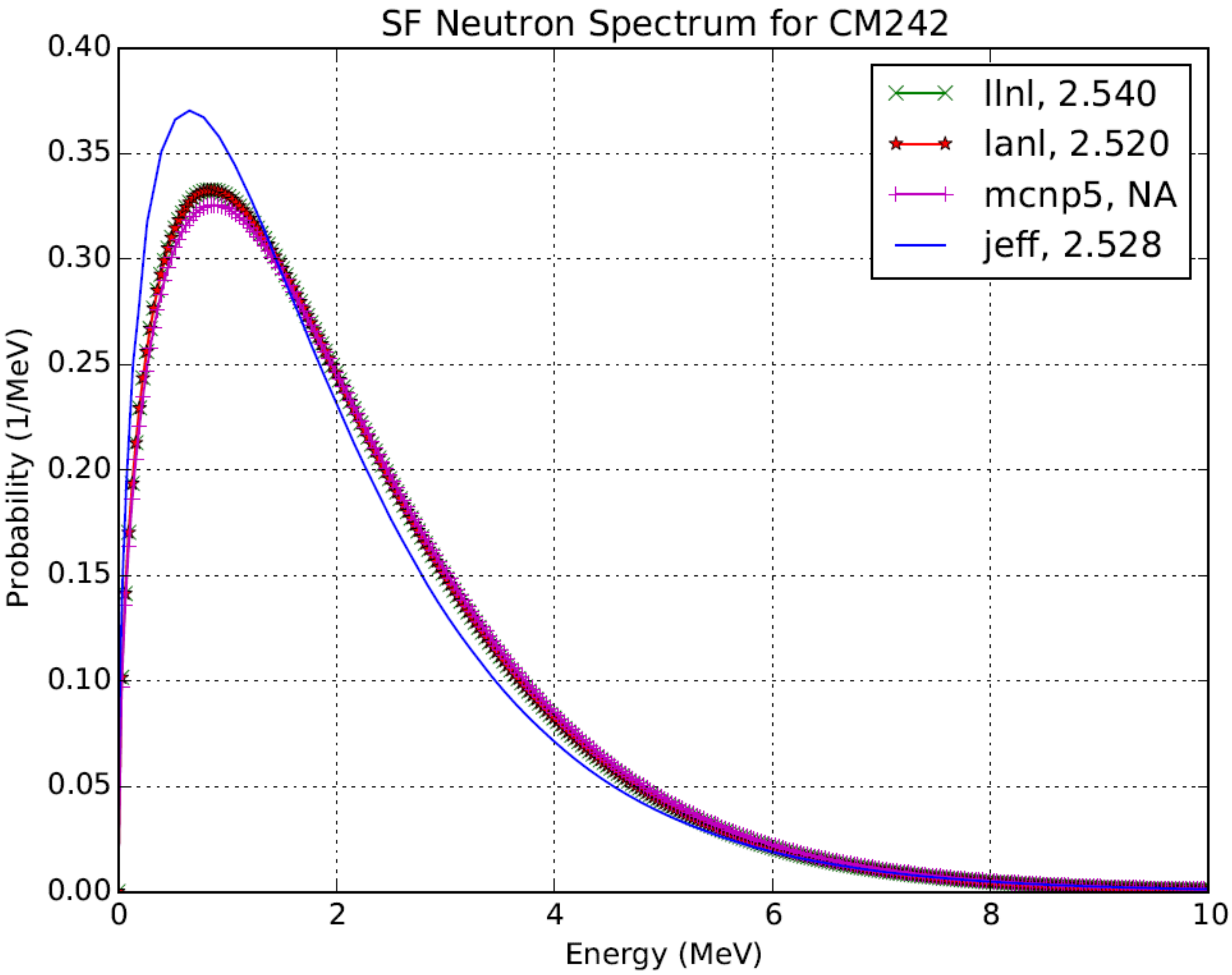
Watt Parameter Evaluations

Nuclide	Verbeke [1]			Shores [2]			MCNP5 [3]	
	$\bar{\nu}$	*a* (MeV)	*b* (1/MeV)	$\bar{\nu}$	*a* (MeV)	*b* (1/MeV)	*a* (MeV)	*b* (1/MeV)
^{232}Th	2.14	0.80	4.0	2.14	0.5934	8.030	-	-
^{232}U	1.71	0.89220	3.72278	1.71	0.8922	3.723	-	-
^{233}U	1.76	0.85480	4.03210	1.76	0.8548	4.032	-	-
^{234}U	1.81	0.77124	4.92449	1.81	0.7712	4.924	-	-
^{235}U	1.86	0.77471	4.85231	1.86	0.7747	4.852	-	-
^{236}U	1.91	0.73516	5.35746	1.91	0.7352	5.357	-	-
^{238}U	2.01	0.64832	6.81057	2.01	0.6483	6.811	-	-
^{237}Np	2.05	0.83344	4.24147	2.05	0.8334	4.241	-	-
^{236}Pu	-	-	-	2.13	0.9883	3.104	-	-
^{238}Pu	2.21	0.84783	4.16933	2.22	0.8478	4.169	-	-
^{239}Pu	2.16	0.88525	3.80269	2.16	0.8852	3.803	-	-
^{240}Pu	2.156	0.79493	4.68927	2.16	0.7949	4.689	0.799	4.903
^{241}Pu	2.25	0.84247	4.15150	2.25	0.8425	4.152	-	-
^{242}Pu	2.145	0.81915	4.36668	2.15	0.8192	4.367	0.833668	4.431658
^{244}Pu	-	-	-	2.30	0.6947	6.004	-	-
^{241}Am	3.22	0.93302	3.46195	2.27	0.9330	3.462	-	-
^{242m}Am	-	-	-	2.34	0.8990	3.708	-	-
^{243}Am	-	-	-	2.42	0.8643	3.990	-	-
^{240}Cm	-	-	-	2.39	1.0717	2.698	-	-
^{242}Cm	2.54	0.88735	3.89176	2.52	0.8874	3.892	0.891	4.046
^{243}Cm	-	-	-	2.68	0.9774	3.190	-	-
^{244}Cm	2.72	0.90252	3.72033	2.69	0.9025	3.720	0.906	3.848
^{245}Cm	-	-	-	2.87	0.9119	3.624	-	-
^{246}Cm	-	-	-	3.18	0.8782	3.886	-	-
^{248}Cm	-	-	-	3.11	0.8084	4.536	-	-
^{250}Cm	-	-	-	3.31	0.7345	5.436	-	-
^{249}Bk	3.40	0.89128	3.79405	3.60	0.8913	3.794	-	-
^{248}Cf	-	-	-	3.34	1.0277	2.932	-	-
^{252}Cf	3.757	1.18000	1.03419	3.765	1.025	2.926	1.025	2.926

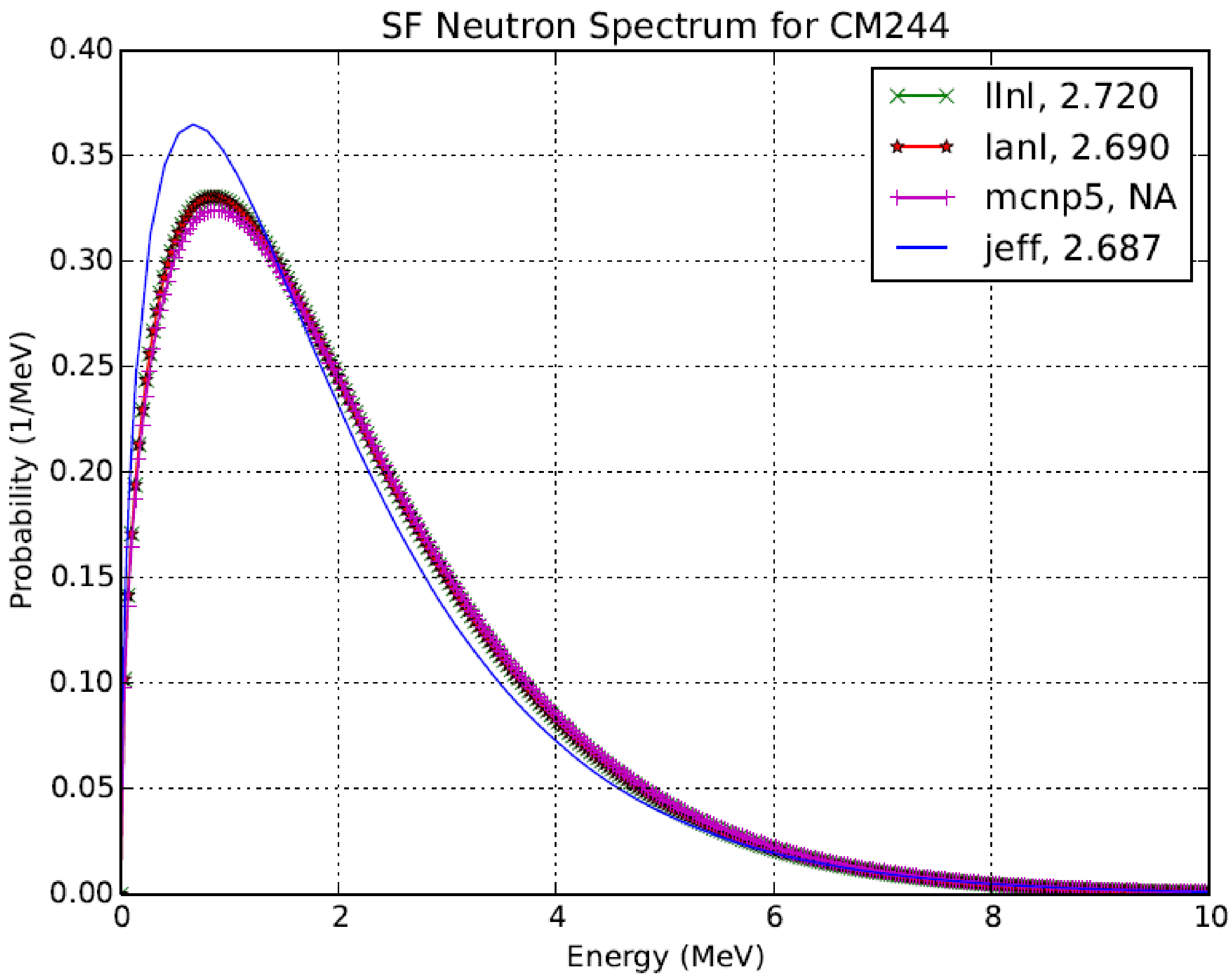
Pu-238 SF Neutron Spectra



Cm-242 SF Neutron Spectra



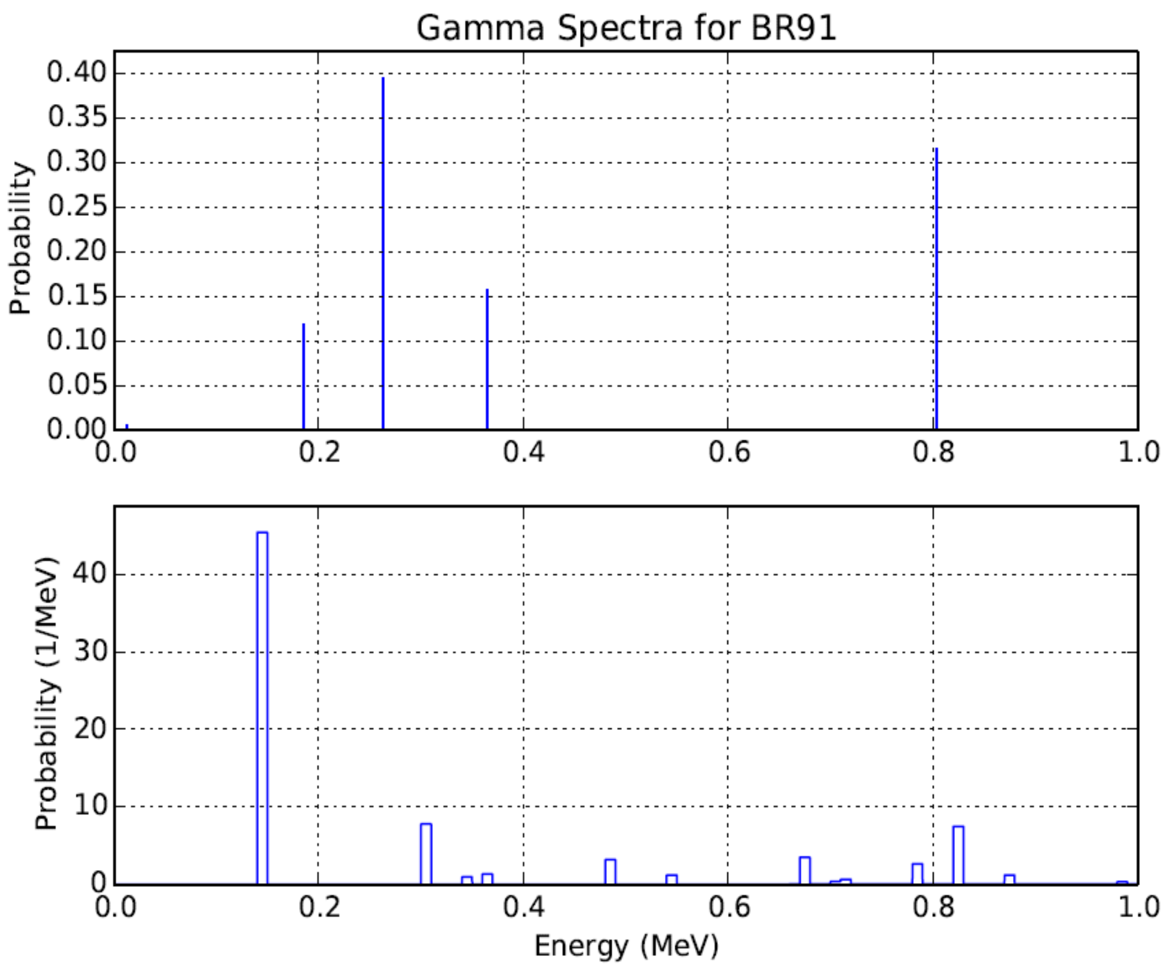
Cm-244 SF Neutron Spectra



Problems with File 8, MT=457 Continuous Neutron and Gamma Spectra

- 30 ENDF/B-VII.1 evaluations have continuous neutron and/or gamma spectra in File 8, MT=457 for which corresponding JEFF-3.1.1 evaluations have discrete spectra available.
  - Ag-123, As-85, Br-87, Br-88, Br-89, Br-90, Br-91, Ce-149, Ga-77, Gd-163, Ge-82, I-137, I-138, I-139, In-131, La-145, Pm-155, Rb-93, Rb-94, Rb-95, Rb-101, Rh-112, Rh-114, Rh-116, Sb-135, Tc-111, Y-97, Y-97M, Y-101, Zn-80
- The decay phenomena are discrete. The JEFF-3.1.1 evaluations are consistent with published discrete data from ENSDF and other databases.
- Inconsistencies between ENDF/B-VII.1 and JEFF-3.1.1 spectra can result in significant differences in average particle energies and productions.

Br-91 Gammas (JEFF-3.1.1 vs. ENDF/B-VII.1)



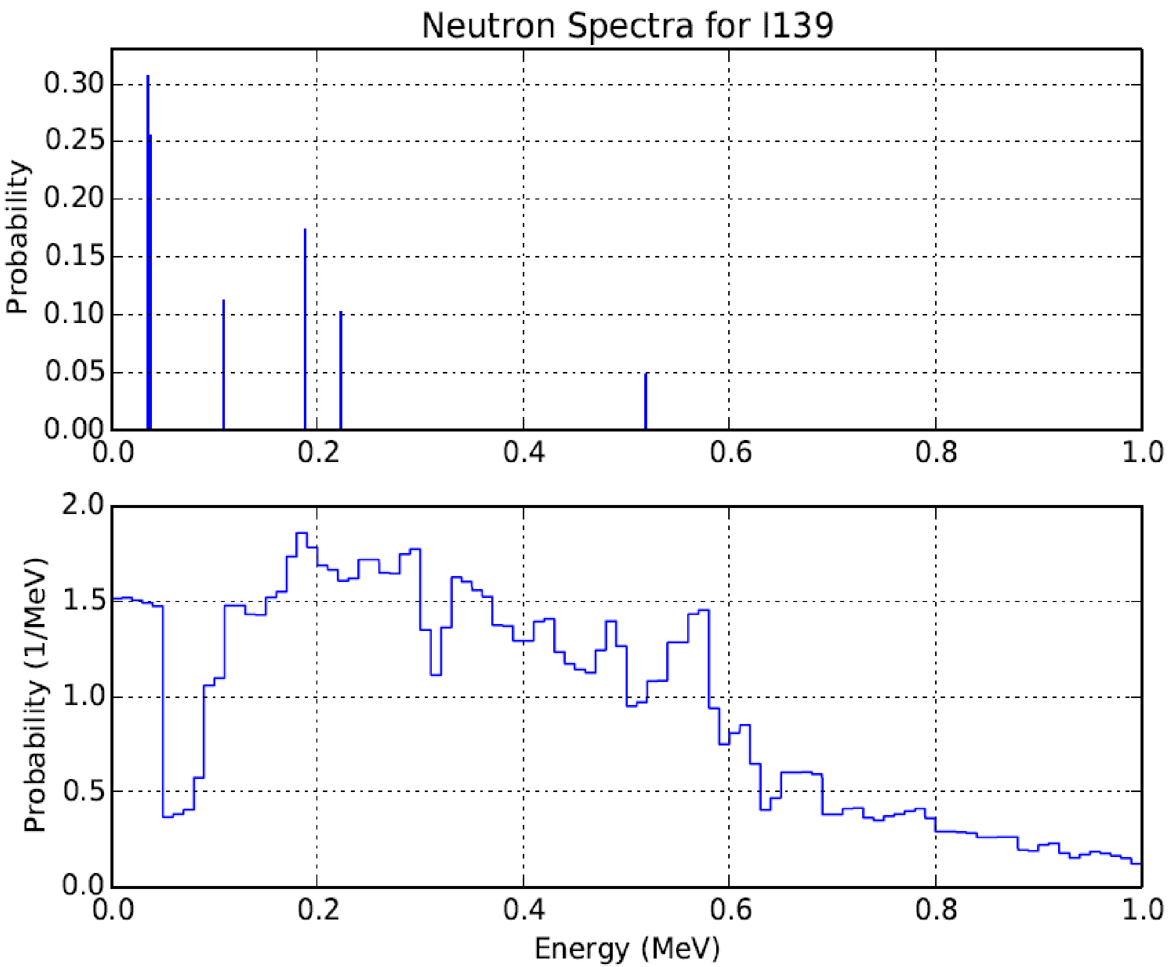
JEFF-3.1.1 (discrete)

Production / decay: 0.99
Ave. Particle Energy: 0.44 MeV

ENDF/B-VII.1 (continuous)

Production / decay: 0.55
Ave. Particle Energy: 0.73 MeV

I-139 Neutrons (JEFF-3.1.1 vs. ENDF-VII.1)



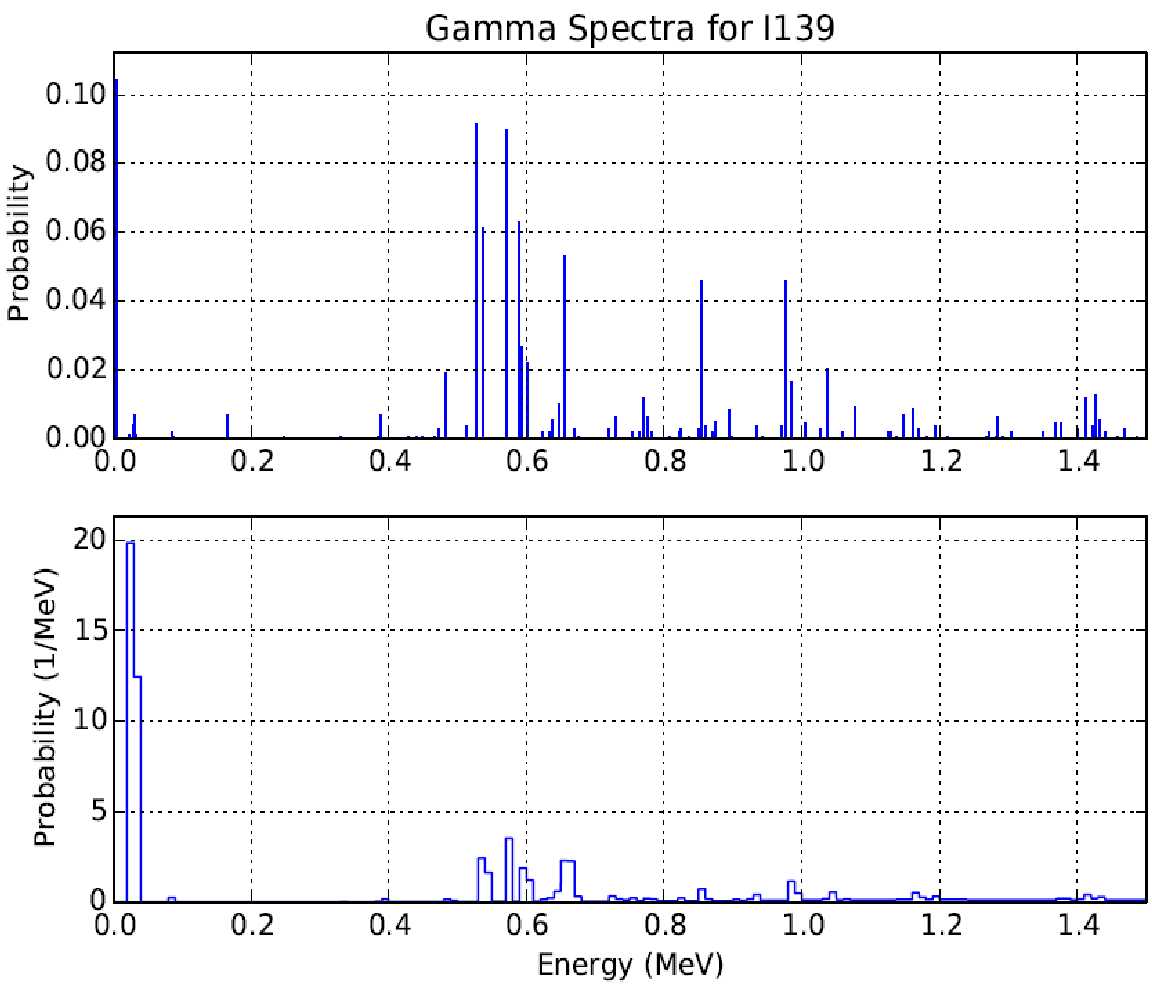
JEFF-3.1.1 (discrete)

Production / decay: 0.10
Ave. Particle Energy: 0.11 MeV

ENDF/B-VII.1 (continuous)

Production / decay: 0.10
Ave. Particle Energy: 0.41 MeV

I-139 Gammas (JEFF-3.1.1 vs. ENDF/B-VII.1)



JEFF-3.1.1 (discrete)

Production / decay: 0.88
Ave. Particle Energy: 0.75 MeV

ENDF/B-VII.1 (continuous)

Production / decay: 2.49
Ave. Particle Energy: 1.18 MeV

Conclusions

Accurate Radioactive Decay Data are important for many applications:

- Photon source term in irradiated nuclear fuel
- Decay heating
- Intrinsic neutron source
- Transuranic neutron source
- Radiation shielding

There is an interest in expanded spontaneous fission neutron emission spectrum data for use in spent fuel analyses (but currently only JEFF can meet user needs). This data should be evaluated and provided in ENDF. Options include using:

- Published Watt parameters (fitted to experimental measurements)
- Madland-Nix spectra (validated with experimental measurements)

Several ENDF Sub-Library 4 Radioactive Decay Data evaluations should have the gamma and neutron emission spectra updated. Continuous spectra are given for discrete phenomena when discrete data is available in JEFF, ENSDF, and other databases.

CSEWG should plan on reviewing Radioactive Decay Data with additional care for ENDF/B-VIII since this data is becoming more extensively used in applications.