Presentation of Neutron Standards Data in the ENDF-6 Formatted Files

V.G. Pronyaev

IAEA/NAPC contractor

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Results of the international evaluation of the neutron cross section standards are presented in the paper [1] in the table format. But the use of standards in the measurements and the evaluations requires its presentation in the computerized formats. The ENDF-6 [2] formatted standards files allow to use the data processing codes [3] for interpolation of the standard cross sections and averaging on the energy groups for normalization of the results of measurements or for inter-comparisons of results of measurements with the standards values [4].

Standards are evaluated in a limited number of common (for all cross sections) energy points (nodes). This allows to use in the evaluation of standards as the results of direct beam measurements with fixed neutron energy and usually medium energy resolution, as well as the results of pulsed beam high resolution time of flight measurements. The interpolation of the standard cross sections between the nodes should follow, in the best case, to the law used for data reduction to the common nodes, as it was defined by W. Poenitz, the author of the GMA code [5, 6]. The log-log interpolation scheme below 30 keV neutron energy for all cross sections having energy dependence close to 1/v and lin-lin interpolation for all other cross sections should be used [4]. For all cross sections above 30 keV the lin-lin interpolation should be used.

The standard capture and fission cross sections are evaluated at thermal point (0.0253 eV) and above 0.15 keV. For ^{235}U(n,f), the additional integral of the cross section between 7.8 and 11.0 eV, often used for cross section normalization in the resonance range, is evaluated. The evaluated covariance matrices include the cross-energy correlations for uncertainties at thermal point, integral value in the resonance range (or average cross section for the energy range) and at the energy nodes for higher energies.

All evaluated data (cross sections and covariances) can be presented in the ENDF-6 formatted files as they are obtained in the GMA fit with additional smoothing where is important. Because some files in this case will be incomplete (cross sections in the resonance energy range are absent), the linearized fission and capture cross sections obtained with LINEAR code [3] for neutron energy below 150 eV will be meaningless for exclusion of the thermal point value. The format don’t allow keep simultaneously the values at discrete points (e.g. thermal value) and continuous presentation of the cross sections given in energy points with the interpolation law between them (e.g. above 150 eV). But nevertheless, the files of 2006 standards cross sections and covariances [7,8] were presented in the ENDF-6 format as they were obtained in the GMA evaluation at thermal energy, energy range of 7.8 – 11 eV and energy nodes above, with all drawbacks of this presentation.

For 2017 standards, decision is taken to present in MF3 cross section only point-wise values, where interpolation of cross sections between the nodes can be applied, but keep uncertainties for all MF33 relative covariances, including covariances in and between between thermal value, integral value (if it was evaluated), and point-wise values at higher energy. The evaluated thermal and integral values are given in the free text of MF451 section. This allows to avoid wrong presentation of the results of standards evaluation after cross section linearization and simultaneously to keep full relative covariance matrix in the MF33 computerized format.

The covariances between different reactions are also obtained in the combined fit of standards cross sections. Most important and relatively high covariances due to correlations between uncertainties of ^{235}U(n,f), ^{238}U(n,f) and ^{239}Pu(n,f) reaction cross sections. But the uncertainties (variances) of 2017 standards were increased by an additional component, which accounts the contribution from Unrecognized Sources of Uncertainties (USU). The simple one-group estimation of this uncertainty was done [1]. This additional diagonal component of uncertainty substantially diminishes all off-diagonal correlations and reduces the influence of the account of covariances between different reactions in the calculations of uncertainties of different integral quantities (k_{eff}, breeding coefficient, …). Because of this, it is decided not to introduce blocks of covariance matrices describing correlations between uncertainties of different reactions in the files. These blocks of correlations can be included in the files of general purpose evaluated libraries. More detailed discussion of USU is given in [9].

The specific features of ENDF-6 formatted files for Standards (2017) reactions are the following:

^{1}H(n,n): File has more data points which provide better accuracy in the case of data interpolation between the points.

^{6}Li(n,t): File gives the data in the wider energy (up to 4 MeV) than recommended for use as standard. The file has more data points which provide better accuracy in the case of data interpolation between the points.

^{10}B(n,α) and ^{10}B(n,$\alpha_1\gamma$): Covariances in the file are slightly different from given in the Tables [1] because the different group boundaries for the covariances in the file.

^{nat}C(n,n): Integral elastic scattering cross section for natural Carbon was obtained by MIXER, using R-matrix evaluations done for ^{12}C and ^{13}C. Angular distributions for elastic scattering cross section and covariance matrix of uncertainties for integral elastic scattering cross section for ^{nat}C was taken as for ^{12}C evaluation (contribution of ^{13}C is small).

^{197}Au(n,γ): Cross section in file MF3 is given between 2.5 keV and 2.8 MeV. Thermal cross section standard value is give in the free text part of MT451 section. Covariances in MF33 file includes the cross correlations between thermal value and higher point-values data.

^{238}U(n,γ): This is not standard cross section. It is obtained as byproduct in the combined fit of all cross sections. USU is given as fully correlated components with uncertainty 1.7% for 0.0001< En<1 MeV, and 2.4% above.

^{238}U(n,f): Cross section is given between 0.5 and 200 MeV with detailed energy dependence between 1 and 2 MeV. Cross section is recommended for use as standards in the energy range 2 – 32 MeV. USU is given as fully correlated component for the all energy range.

^{235}U(n,f): Cross section in file MF3 is given between 150 eV and 200 MeV. Thermal cross section value and integral cross section between 7.8 – 11 eV, used as standard, is given in the free text part of MT451 section. Covariances in MF33 file includes the cross correlations between thermal value, integral value and higher point-values data. USU is given as fully correlated component for the energy range above 150 eV.

^{239}Pu(n,f): This is not standard cross section. It is obtained as byproduct in the combined fit of all cross sections. Cross section in file MF3 is given between 150 eV and 200 MeV. Thermal cross section value, used as standard, is given in the free text part of MT451 section. Covariances in MF33 file includes the cross correlations between thermal value and higher point-values data. USU is given as fully correlated component for the energy range above 150 eV.

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