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ASTM STANDARD RECOMMENDED GUIDE ON APPLICATION OF ENDF/A CROSS SECTION AND UNCERTAINTY FILE: ESTABLISHMENT OF THE FILE

E. P. Lippincott and W. N. McElroy

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ASTM STANDARD RECOMMENDED GUIDE ON APPLICATION OF ENDF/A CROSS SECTION AND UNCERTAINTY FILE: ESTABLISHMENT OF THE FILE

E. P. Lippincott and W. N. McElroy

Westinghouse Hanford Company
Hanford Engineering Development Laboratory
Richland, Washington 99352

A new ASTM Standard Recommended Guide on "Application of ENDF/A Cross Section and Uncertainty File" is in preparation by ASTM Committee E10 on Nuclear Technology and Applications. This ASTM Standard is being prepared in support of the standardization of physics-dosimetry procedures and data needed for Light Water Reactor (LWR) power plant pressure vessel and support structure materials surveillance and test reactor development programs. The main subject of this paper is the establishment of the "ENDF/A Cross Section and Uncertainty File".

The development of evaluated cross section files such as the "evaluated nuclear data file," ENDF/B, has occurred mainly to meet the needs of physics calculators. These files are tested by calculations of well-measured benchmark problems such as reactivity or critical mass measurements. Data in the files have then been re-evaluated where disagreements with the benchmark measurements indicate data to be deficient.

For cross sections of reactions used for dosimetry measurements it was found that a more specialized file was needed in order to contain the specific dosimetry reactions. For example, instead of an iron (n,p) cross section, the $^{54}Fe(n,p)^{54}Mn$ cross section is needed. Until the creation of the dosimetry file,[1] and later the ENDF gas production file,[2] the cross sections for many dosimetry reactions which are unimportant for neutron transport calculations, did not receive the proper attention by the evaluators.[3 5]

Furthermore, in neutron dosimetry and damage analysis work, standardized techniques and data must be established to characterize a diversity of irradiation environments.[6] The techniques must be implemented in such a manner that fuels and materials data from the different environments can be intercompared, and the environments are sufficiently characterized so that the fuels and materials data can be properly correlated, and then interpolated and extrapolated to different reactor design conditions. The need of such standardization is clear when the high cost of the replacement of fuels, materials, and components (including surveillance and irradiation tests) for light water reactor (LWR), fast breeder reactor (FBR), or magnetic fusion reactor (MFR) nuclear power systems are considered. Derived irradiation effects data, therefore, must have as much general applicability as possible to effect the highest benefit to cost ratio. For U. S. reactor programs key test irradiation facilities, adequately characterized and labeled as "benchmarks", are being utilized for the validation and calibration of dosimetry, damage analysis, and the associated reactor analysis procedures and data. A provisional list of such benchmarks is given in Reference 6 as well as a discussion of goal accuracies. More recent information for LWRs is given in References 7, 8, and 9.

The need for a standardized approach is accentuated by the variety of dosimetry monitors and techniques used for the various applications. Consistency from one set of measurement conditions to another must obviously start with a consistent cross section file. To meet this need for LWR pressure vessel surveillance dosimetry, an ENDF/A cross section and uncertainty file is being established together with an ASTM Standard recommended guide for application of the file.[10] The file will be issued as ENDF/A because it may contain cross sections inconsistent with those on ENDF/B. (ENDF/B files are evaluated files officially approved by the Cross Section Evaluation Working Group (CSEWG) after suitable review and testing.) In addition, the ASTM ENDF/A file will contain damage cross sections [e.g. displacements per atom (dpa)] for steel, graphite, silicon, sapphire, quartz, etc. for which reaction mechanisms are only known theoretically and differential cross section measurements do not exist.

Differences with the ENDF/B dosimetry file may be created by the need for a standardized, self-consistent cross section set. At present, evaluations and testing of many dosimetry reactions have reduced discrepancies between evaluations and integral data. Thus only a few cross sections may need significant adjustment from the ENDF/B file to achieve self-consistency with benchmark integral data. In general, these cross sections are ones for which present differential measurements are inadequate and theoretical calculations have only partly filled the gap. A prime example is the ^{58}Fe(n,γ) reaction. Table 1 shows the present status of cross sections measured in the ^{235}U thermal neutron induced fission spectrum compared with calculated values using the 620 point ENDF/B-V dosimetry file cross sections. It is seen that most reactions agree within about the quoted experimental error but discrepancies still exist with the reactions ^{47}Ti(n,p), ^{27}Al(n,p), ^{127}I(n,2n), and ^{55}Mn(n,2n).

Limiting the present ASTM ENDF/A file to LWR pressure vessel dosimetry and damage analysis applications may create an adjusted file not suitable for other applications. Thus caution must be observed when extending its use beyond the limits within which the file has been tested. This is caused by the fact that the adjustments may be caused by effects not explicitly considered. For example, in an environment containing thermal or low energy neutrons, the measured value for the ^{63}Cu(n,α)^{60}Co reaction may be affected by ^{59}Co impurity in the copper used as the dosimeter. As little as 1 ppm ^{59}Co may cause a 20% effect. Thus an effective copper cross section might contain a low energy part due to ^{59}Co(n,γ)^{60}Co that is specific for the source of the copper used. Other effects that could cause similar problems are photofission and burn-in, burn-out effects.[11]

An integral part of the ENDF/A file will be an uncertainty file which can be used by least squares adjustment codes such as FERRET[12] or STAYSL[13] to properly weight data used in neutron flux and spectrum determinations and provide a statistical evaluation of uncertainty in processed quantities such as fluence or dpa.[14,15] The use of a validated uncertainty file will provide the needed confidence to justify usage of the derived uncertainties for defining neutron induced materials property change exposure limits.[9]

In order to make the ENDF/A file easily usable by the adjustment codes,[14] it will be issued in a multigroup format with sufficient groups for most applications. Groups can be condensed for input to the codes. The uncertainties will be specified in the form of a covariance matrix and correlations between cross sections will be specified, either in the file or in the file documentation. Codes exist for collapsing or expanding covariance file data into any desired group structure.

It is expected that the use of the ENDF/A file will result in standardized analysis of LWR dosimetry and the subsequent derivation of exposure parameter values. It should therefore, find wide application to define uncertainties on a rigorous statistical basis, thereby enabling materials property exposure limits to be established in a consistent, scientifically justified manner. The use of such data files for international intercomparisons, such as REAL-80, can be expected to play an important part in meeting this goal.

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TABLE 1

COMPARISON OF MEASURED AND CALCULATED CROSS SECTIONS
IN THE U-235 FISSION NEUTRON SPECTRUM

Reaction	Effective Threshold (MeV)	Measured Value (mb)[a]	Quoted Error in Measured Value %[a]	Calculated Value (mb)[b]	Calculated/ Measured
$^{115}In(n,\gamma)^{116m}In$		134.5	4.5	124.7	0.93
$^{197}Au(n,\gamma)^{198}Au$		83.5	6.0	78.3	0.94
$^{63}Cu(n,\gamma)^{64}Cu$		9.30	15.1	9.87	1.06
$^{235}U(n,f)$		1203	2.5	1236	1.03
$^{239}Pu(n,f)$		1811	3.3	1791	0.99
$^{237}Np(n,f)$	0.6	1312	3.8	1347	1.03
$^{115}In(n,n')^{115m}In$	1.2	189	4.2	179	0.95
$^{232}Th(n,f)$	1.4	81	6.7	75.0	0.93
$^{238}U(n,f)$	1.5	305	3.3	305	1.00
$^{47}Ti(n,p)^{47}Sc$	2.2	19.0	7.4	22.5	1.18
$^{58}Ni(n,p)^{58}Co$	2.8	108.5	5.0	105.0	0.97
$^{32}S(n,p)^{32}P$	2.9	66.8	5.5	70.5	1.06
$^{54}Fe(n,p)^{54}Mn$	3.1	79.7	6.1	81.0	1.02
$^{46}Ti(n,p)^{46}Sc$	3.9	11.8	6.4	11.2	0.95
$^{27}Al(n,p)^{27}Mg$	4.4	3.86	6.5	4.26	1.10
$^{56}Fe(n,p)^{56}Mn$	6.0	1.035	7.2	1.036	1.00
$^{59}Co(n,\alpha)^{56}Mn$	6.8	0.143	7.0	0.150	1.05
$^{63}Cu(n,\alpha)^{60}Co$	6.8	0.500	11.2	0.558	1.12
$^{27}Al(n,\alpha)^{24}Na$	7.2	0.705	5.7	0.719	1.02
$^{48}Ti(n,p)^{48}Sc$	7.6	0.300	6.0	0.282	0.94
$^{127}I(n,2n)^{126}I$	10.5	1.05	6.2	1.21	1.15
$^{55}Mn(n,2n)^{54}Mn$	11.6	0.244	6.1	0.201	0.82

(a) Taken from Reference 2, 1σ values.
(b) Using ENDF/B-V dosimetry file 620 point cross sections and the ENDF/B-V Watt form for the ^{235}U fission spectrum.