High-energy nuclear data files: From ENDF-6 to NJOY to MCNP

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Abstract

The structure of a 150 MeV neutron datafile is outlined. Emphasis is put on the possibility to process a high-energy file with NJOY into an MCNP-library. Examples of parts of such a file are given as illustration. Furthermore, we have processed some existing data libraries and report on the encountered difficulties. Accordingly, we give recommendations for the format to be used for evaluated neutron datafiles for intermediate energy applications.

I. INTRODUCTION

Intermediate energy nuclear data evaluation is a rapidly increasing activity. Although a *formal* international platform does not yet exist, several laboratories have started extensive programs that should ensure decent nuclear reaction information for applied (transport) calculations for both neutrons and charged particles with energies above 20 MeV.

Accelerator-based transmutation of waste, energy amplification, light-ion production and medical isotope production as well as some new fusion concepts are the main projects that drive the work on nuclear data above 20 MeV. Accordingly, relevant nuclear reaction measurements have been performed and are being organised, more theory and model code developers are focussing on the $E > 20$ MeV region, and the database for experimental nuclear reactions, EXFOR, has substantially been extended.

In parallel, the actual provision of nuclear data to the users, which forms a justification of the aforementioned measurements and code developments that can hardly be overemphasized, is now also beyond the initial stage. As a complement to intranuclear cascade codes, the creation of an Evaluated Nuclear Data File (ENDF) is required that gives a more precise representation of nuclear reactions below the pion threshold. A previous NEA document [1], gives an outline of the datafile aspects from the evaluators' point of view, *i.e.* it describes how high-energy cross sections and other physical quantities, coming from experiments and (mainly) nuclear model codes, should be categorized and stored in an evaluated data file. In this report we will perform such a study from the users' point of view. We address the technical aspects of the data format, but now in particular in relation with the processing of a datafile to a transport library. Arguably, the most important application of high energy data is in (high-energy) Monte Carlo particle transport calculations. The most widely used computer code in this field is MCNP4A [2]. Therefore, for this project the attention was only focussed on the requirements imposed by MCNP4A. Since this code can not (yet) handle charged particles, we restrict ourselves to neutron evaluations. The high-energy limit is 150 MeV.

In Section II we describe the structure of a high-energy data file. The starting point is Ref. [1], but it is complemented with the additional format aspects that have emerged since 1994. In each subsection, we give our recommendations for an ENDF, such that it can be processed by the code NJOY [3] into an MCNP-library. Next, we report about the processing of other existing libraries and we suggest which elements of these evaluations should be adopted and which should be rejected. Finally, we give the conclusions.

II. ENDF-6 PROCEDURES FOR HIGH ENERGY

For the construction of a high energy data file, procedures different from those used in the low energy files have to be employed, since it is no longer possible to store all reactions that describe different sequential particle emissions in separate MT-numbers. On the other hand, for accelerator-driven system calculations detailed low energy neutron data are as important as they are in normal reactor calculations. Therefore, as a general rule, the representation of cross sections below 20 MeV should be left untouched as much as possible. For energies above 20 MeV, the detailed information concerning each individual excited state of the target nucleus and each particular sequential reaction chain is less important than for low energy neutrons. Therefore, it is appropriate to lump almost all reaction information in MT5 (which comprises all non-elastic processes that are not explicitly considered in other MT-numbers). The particle and product yields and the outgoing energy-angle distributions can then be stored in MF6/MT5.

We will describe MF 1, 3, 4 and 6 in particular since they are usually subject to non-trivial changes following the inclusion of high-energy data. The other MF-numbers normally stay unchanged (apart from formally increasing the high-energy limit in each occurring MT-number). As basis we take the ^{56}Fe file from ENDF/B-VI which we extended to 150 MeV.

A. File 1 (General information)

This file contains the usual descriptive data and information text. Formal agreements still have to be made on library identifiers and version numbers. The various fission quantities are stored in MT452, 455, 456 and 458. As for documentation, it is appropriate to retain all information of the existing 20 MeV file and to let this precede by the new high-energy information.

Hence, the beginning of the datafile looks as follows:

```
 Intermediate energy neutron  evaluation for  56FE up to 150 MeV    99 0  0
 2.605600+4 5.545400+1          1          0          0         22631 1451
 0.000000+0 0.000000+0          0          0          0         62631 1451
 1.000000+0 0.000000+0          0          0         10         62631 1451
 0.000000+0 0.000000+0          0          0        360       1672631 1451
 26-FE- 56 ECN       EVAL-FEB97 A. Koning                          2631 1451
 ECN-XXX-XX-97       DIST-      REV1-                              2631 1451
----JEFF-3-HEL       Material 2631        REVISION 1               2631 1451
-----INCIDENT NEUTRON DATA                                         2631 1451
------ENDF-6 FORMAT                                                2631 1451
                                                                   2631 1451
       *************************************************           2631 1451
       *                                               *           2631 1451
       *          JEFF-3.0  HIGH ENERGY FILE           *           2631 1451
       *                                               *           2631 1451
       *************************************************           2631 1451
                                                                   2631 1451
             NEUTRON EVALUATION UP TO 150 MEV                      2631 1451

   Put new high-energy information here.


     ......
                                                                   2631 1451
            DESCRIPTION OF THE 1.E-11 - 20 MEV DATA                2631 1451
                                                                   2631 1451
   Retain existing low-energy information here.


     ......
```

B. File 3 (Cross sections)

In MT1 and MT2, the total and total elastic cross sections can be given. Below 20 MeV, a sufficiently fine energy grid is generally used for a proper description of the cross section. Above 20 MeV, progressively larger steps in energy can be taken, although it is generally easy to provide the total cross section by an optical model code at e.g. energy steps of 1 MeV. Below 20 MeV, sections MT16-21,38,51-90,91,102-107 can be used to represent the data. For energies above 20 MeV, the cross sections in these MT numbers are zero. Formally, this can be accomplished by setting the cross section equal to zero at the two energy points 20 and 150 MeV. In MT5, the situation is reversed: all partial non-elastic cross sections, with the exclusion of fission, can be lumped in MT5 for energies above 20 MeV and the data are zero below 20 MeV.

The transition from the 0-20 MeV part to the 20-150 MeV part in the total cross section (MT1) is as follows:

```
 1.700000+7 2.325000+0 1.750000+7 2.309000+0 1.800000+7 2.292900+02631 3  1
 1.850000+7 2.278400+0 1.900000+7 2.263900+0 1.950000+7 2.250900+02631 3  1
 2.000000+7 2.237900+0 2.000000+7 2.272900+0 2.100000+7 2.251600+02631 3  1
 2.200000+7 2.239400+0 2.300000+7 2.234900+0 2.400000+7 2.236600+02631 3  1
 2.500000+7 2.243500+0 2.600000+7 2.254400+0 2.700000+7 2.268500+02631 3  1
```

and at the end of the energy grid we have

```
 1.420000+8 1.369900+0 1.430000+8 1.362300+0 1.440000+8 1.354700+02631 3  1
 1.450000+8 1.347200+0 1.460000+8 1.339800+0 1.470000+8 1.332500+02631 3  1
 1.480000+8 1.325200+0 1.490000+8 1.318000+0 1.500000+8 1.310900+02631 3  1
                                                                  2631 3  0
```

The extension for the elastic (MT2) and non-elastic (MT3) cross section is analogous.

Note that a double point at 20 MeV is used. An alternative method is to use 20.00000 and 20.00001 MeV. However, we have found that this may lead to processing problems, as reported in the next section. The 20.00000 MeV will not appear in the MCNP library. The 20.00001 MeV point will appear, with an absorption cross section equal to zero. With an exact double point at 20 MeV, NJOY generates points at 19.99999 MeV and at 20.00001 MeV.

The beginning of the (n,x) cross section (MT5) is:

```
 2.605600+4 5.545400+1          0          0          0          02631 3  5
 0.000000+0 0.000000+0          0          0          1        1332631 3  5
        133          2                                            2631 3  5
 1.000000-5 0.000000+0 2.000000+7 0.000000+0 2.000000+7 1.304600+02631 3  5
 2.100000+7 1.284400+0 2.200000+7 1.264300+0 2.300000+7 1.244100+02631 3  5
 2.400000+7 1.223700+0 2.500000+7 1.203300+0 2.600000+7 1.183000+02631 3  5
```

From 20 MeV on, this section is simply filled with total reaction cross sections (stemming from model calculations) if other partial cross sections (such as fission) are not used.

All the other MT-numbers need to be formally extended to 150 MeV, e.g. the (n,2n) cross section becomes:

```
 2.605600+4 5.545400+1          0          0          0          02631 3 16
-1.120000+7-1.120000+7          0          0          1         282631 3 16
         28          2                                            2631 3 16
 1.140200+7 0.000000+0 1.150000+7 2.240400-3 1.160000+7 7.593600-32631 3 16
 1.170000+7 1.521000-2 1.180000+7 2.504200-2 1.190000+7 3.747700-22631 3 16
 1.200000+7 4.993300-2 1.220000+7 7.865300-2 1.240000+7 1.115900-12631 3 16
 1.260000+7 1.465600-1 1.280000+7 1.881600-1 1.300000+7 2.229300-12631 3 16
 1.350000+7 3.176800-1 1.400000+7 4.020400-1 1.450000+7 4.678000-12631 3 16
 1.500000+7 5.198900-1 1.550000+7 5.565200-1 1.600000+7 5.827100-12631 3 16
 1.650000+7 6.010700-1 1.700000+7 6.107800-1 1.750000+7 6.114500-12631 3 16
 1.800000+7 6.099200-1 1.850000+7 6.047900-1 1.900000+7 5.975900-12631 3 16
 1.950000+7 5.878800-1 2.000000+7 5.754000-1 2.000000+7 0.000000+02631 3 16
 1.500000+8 0.000000+0                                            2631 3 16
                                                                  2631 3  0
```

Fission cross section should be explicitly specified in MT18.

C. File 4 (Neutron angular distributions)

Neutron elastic angular distributions should be stored in MT2. The official ENDF6-limit for Legendre coefficients is 20 and this is insufficient for incident energies above 20 MeV. Currently, work is in progress [4] to extend this limit. It should be noted that if l_{max} is the number of partial waves in a direct reaction calculation, the number of required Legendre coefficients is equal to $2l_{max}$. The problem can easily be circumvented by representing the data in tabular form, which we recommend for the moment. However, this entails a small numerical change of the data below 20 MeV, which need to be transformed from the usual Legendre expansion into tabular format, since the ENDF6-format allows only one choice over the whole energy range.

The beginning of the elastic angular distribution section (MT2) is as follows:

```
 2.605600+4 5.545400+1          0          2          0          02631 4  2
 0.000000+0 5.545400+1          0          2          0          02631 4  2
 0.000000+0 0.000000+0          0          0          1        4152631 4  2
        415          2                                            2631 4  2
 0.000000+0 1.000000-5          0          0          1          22631 4  2
          2          2                                            2631 4  2
-1.000000+0 5.000000-1 1.000000+0 5.000000-1                      2631 4  2
 0.000000+0 4.581000+4          0          0          1         962631 4  2
         96          4                                            2631 4  2
-1.000000+0 4.769000-1-9.975641-1 4.765789-1-9.902681-1 4.756402-12631 4  2
-9.781476-1 4.741556-1-9.612617-1 4.722387-1-9.396926-1 4.700366-12631 4  2
-9.135455-1 4.677189-1-8.829476-1 4.654657-1-8.480481-1 4.634541-12631 4  2
-8.090170-1 4.618457-1-7.660444-1 4.607756-1-7.193398-1 4.603435-12631 4  2
```

and this should be given until the last energy point at 150 MeV:

```
 0.000000+0 1.500000+8          0          0          1      962631 4  2
        96          4                                          2631 4  2
-1.000000+0 1.130971-7-9.975641-1 1.296994-7-9.902681-1 1.732271-72631 4  2
-9.781476-1 2.193330-7-9.612617-1 2.292299-7-9.396926-1 1.869913-72631 4  2
-9.135455-1 1.339409-7-8.829476-1 1.189812-7-8.480481-1 1.122343-72631 4  2
-8.090170-1 8.162454-8-7.660444-1 9.556026-8-7.193398-1 1.188149-72631 4  2
-7.071068-1 1.096145-7-6.691306-1 6.406065-8-6.293204-1 6.792585-82631 4  2

        .....

 9.902681-1 3.834116+1 9.925462-1 4.932237+1 9.945219-1 6.103440+12631 4  2
 9.961947-1 7.283584+1 9.975641-1 8.398442+1 9.986295-1 9.369836+12631 4  2
 9.993908-1 1.012489+2 9.998477-1 1.060383+2 1.000000+0 1.076809+22631 4  2
                                                               2631 4  0
```

Only one incident energy point at 20 MeV, with its tabular set of data, should be used (no double point!).

The other MT-numbers can be formally extended to 150 MeV, such as the angular distribution to the first excited state (MT51):

```
 0.000000+0 2.000000+7          0          0         20          02631 4 51
 4.899400-1 2.216500-1 6.976500-2-1.878700-3-4.401100-2-5.915900-22631 4 51
-3.660500-2-2.251600-2-1.925800-2 1.583100-4 1.433900-2 7.685200-32631 4 51
 5.569200-3 2.582800-3 1.010600-3 3.551200-4 1.156300-4 3.536700-52631 4 51
 1.060300-5 3.229100-6                                         2631 4 51
 0.000000+0 1.500000+8          0          0         20          02631 4 51
 4.899400-1 2.216500-1 6.976500-2-1.878700-3-4.401100-2-5.915900-22631 4 51
-3.660500-2-2.251600-2-1.925800-2 1.583100-4 1.433900-2 7.685200-32631 4 51
 5.569200-3 2.582800-3 1.010600-3 3.551200-4 1.156300-4 3.536700-52631 4 51
 1.060300-5 3.229100-6                                         2631 4 51
                                                               2631 4  0
```

Copying the Legendre data of 20 MeV to 150 MeV, as above, is allowed (some values *have* to be given) since the corresponding 150 MeV cross section of MF3 is zero. Note that we have chosen to lump the high-energy discrete state data in the continuum (MF6) instead of extending each individual inelastic cross section to 150 MeV. In future evaluations, inelastic angular distributions for discrete states up to 150 MeV could be stored in MF4/MT51-90, giving more detailed reaction information. Sensitivity studies will have to reveal whether this is necessary.

Also, in MT18-21,38, the angular distributions for fission neutrons can be specified.

D. File 6 (Yields, energy-angle distributions)

Yields of important outgoing particles, photons and possible recoils are stored here. For formal reasons, an energy point at the highest incident energy, *i.e.* 150 MeV, must always be added to the 0-20 MeV sections. These formal additions have no further impact if the corresponding cross sections of MF3 are set equal to zero. Double points in MF6 may cause problems and should be avoided. In MF6/MT5, the yields and energy-angle distributions of the light particles, photons *and* the product nuclides or recoils (excluding fission products) for energies above 20 MeV can be stored. The data should be represented in the CM system for secondary energy and angle (LCT=2). First the yields and energy-angle distribution for neutrons, protons, deuterons, tritons, He-3 and alphas can be given. For the light-particle energy-angle distributions we recommend LAW=1 (continuum energy-angle distribution) with LANG=2 (Kalbach expansion) in the CM-system. For example, the subsection for neutron production starts with the yields, which is followed by the Kalbach energy-angle distribution:

```
 2.605600+4 5.545400+1          0          2          6          02631 6  5
 1.000000+0 1.000000+0          0          1          1        432631 6  5
         43          2                                            2631 6  5
 1.000000-5 1.505273+0 2.000000+7 1.505273+0 2.000001+7 1.505273+02631 6  5
 2.100000+7 1.495777+0 2.200000+7 1.488195+0 2.300000+7 1.486749+02631 6  5
                       ..........
 1.200000+8 3.889506+0 1.250000+8 3.960081+0 1.300000+8 4.034938+02631 6  5
 1.350000+8 4.108793+0 1.400000+8 4.185938+0 1.450000+8 4.235426+02631 6  5
 1.500000+8 4.300413+0                                            2631 6  5
 0.000000+0 0.000000+0          2          1          1        432631 6  5
         43          2                                            2631 6  5
 0.000000+0 1.000000-5          0          1          6         22631 6  5
 0.000000+0 1.000000+5 0.000000+0 1.000000-5 1.000000+5 0.000000+02631 6  5
 0.000000+0 2.000000+7          0          1          6         22631 6  5
 0.000000+0 5.000000-8 0.000000+0 2.000000+7 5.000000-8 0.000000+02631 6  5
 0.000000+0 2.000001+7          0          1        243        812631 6  5
 0.000000+0 1.789747-7 4.479000-2 3.683575+5 2.030147-7 5.891000-22631 6  5
 6.139291+5 2.360443-7 6.891000-2 8.595008+5 2.493974-7 7.974000-22631 6  5
 1.105072+6 2.436898-7 9.168000-2 1.350644+6 2.247146-7 1.055000-12631 6  5
```

This is followed by the yields and energy-angle distributions of the other light particles. Subsequently, MT5 can be filled with the recoils of the product nuclides. The possibility of adequate storage of recoils is rather new and will be reported elsewhere by the LANL group [5].

Finally, the photon yields can be given, usually with an isotropic energy-angle distribution. The MT-numbers that already existed in the 20 MeV file should be extended to the highest incident neutron energy, *i.e.* 150 MeV, as in MF4.

E. Other MF's

The resonance parameters from the existing low-energy neutron data file can be retained in MF2.

For fission-product yields it is probably most easy to keep the special yield and decay library, of course with addition of intermediate-energy data points. The advantage is that the current contents of this file can easily be extended.

For all other MF-numbers, it is again required to formally extend the file to 150 MeV, as in the following example for MF15/MT102:

```
 2.605600+4 5.545400+1          0          0          1          0263115102
 0.000000+0 0.000000+0          0          1          1          3263115102
          3          1                                           263115102
 1.000000-5 1.000000+0 2.000000+7 1.000000+0 1.500000+8 1.000000+0263115102
 0.000000+0 0.000000+0          0          0          1         25263115102
         25          1          0          0          0          0263115102
```

F. Proposed ENDF6-directory

In table I, a typical directory for an intermediate energy neutron datafile is given.

TABLES

TABLE I. Directory of neutron data file up to 150 MeV. The denoted energy regions contain non-zero cross sections.

MF	MT	Description	Remarks
1	451	General information	
1	452-458	Fission quantities	
2	151	Resonance parameters	
3	1	Total cross section	0-150 MeV
3	2	Elastic cross section	0-150 MeV
3	3	Non-elastic cross section	0-150 MeV
3	4	Inelastic cross section	0-20 MeV
3	5	(n,x) cross section	20-150 MeV
3	16	(n,2n) cross section	0-20 MeV
3	17	(n,3n) cross section	0-20 MeV
3	18	Total fission cross section	0-150 MeV
3	19-21,38	1st-4th chance fission cross section	0-150 MeV
3	51-90	(n,n') cross section for 1st-40th excited state	0-20 MeV
3	91	(n,n') cross section for continuum	0-20 MeV
3	102	(n,γ) cross section	0-20 MeV
4	2	Neutron elastic angular distribution	0-150 MeV
4	18	Total fission neutron angular distribution	0-150 MeV
4	19-21,38	1st-4th chance fission neutron angular distribution	0-150 MeV
6	5	(n,x) yields, energy-angle distribution	20-150 MeV
6	16	(n,2n) energy-angle distribution	0-20 MeV
6	17	(n,3n) energy-angle distribution	0-20 MeV
6	51-90	(n,n') angular distribution for 1st-40th excited state	0-20 MeV
6	91	(n,n') energy-angle distribution for continuum	0-20 MeV
6	102	(n,γ) energy-angle distribution	0-20 MeV
8	454	Independent fission-product yields	0-150 MeV
8	459	Cumulative fission-product yields	0-150 MeV

III. PROCESSING OF EXISTING LIBRARIES

The nuclear datafiles described and mentioned in this report were processed by the cross-section processing code system NJOY. Versions 91.128 (including ECN updates) and 94.035 were used.

Only few *released* high-energy neutron evaluated nuclear data files are currently in existence. Besides the datafile as described in the previous section, we have taken nuclear data from the 100 MeV Los Alamos evaluation (LANL) [6] and the 1 GeV BNL/JAERI evaluation (BNL) [7].

At present, an extensive 150 MeV data-library program is in progress at LANL [8] and this will replace the old 100 MeV-library. This new data library is not yet released and therefore does not enter this discussion. However, the technical aspects of this library are practically equal to those discussed in the previous section [9].

A. LANL

The 100 MeV LANL evaluation [6] contains data for ^{1}H, ^{9}Be, ^{12}C, ^{27}Al, ^{28}Si, $^{\mathrm{nat}}$Fe, $^{\mathrm{nat}}$W and ^{238}U. Cross-section data below 20 MeV were taken from ENDF/B-V and ENDF/B-VI evaluations. Data above 20 MeV result from nuclear model calculations. Data below 20 MeV are represented using standard MT-numbers. For the energy range above 20 MeV a simple representation is used: only MT2 (elastic cross section) and MT5 ((n, x) cross section), and sometimes MT102 (radiative capture cross section) are given.

The transition from the representation below 20 MeV to the high-energy representation should be made by using double-points in the evaluation. In some cases no use is made of double-points in this evaluation. Instead, a transition is made from 20.00000 MeV to 20.000001 MeV. This causes problems in the processing, as it gives rise to unphysical "jumps" in the cross section at 20 MeV. In particular, we found this for the ^{9}Be evaluation.

Furthermore, the use of MT102 above 20 MeV is conspicuous. Probably, this cross section should be zeroed at 20 MeV: radiative capture seems to be accounted for in MT5 (this seems to be solved in the new 150 MeV LANL library [9]).

The use of MT5 above 20 MeV simplifies the evaluation and gives no significant problem in MCNP. An exception is the fission cross section, which is included in MT5. Heating calculations at elevated energies in ^{238}U result in far too low heating values, due to the fact that fission is not correctly taken into account in MCNP when it is included in MT5. It is suggested, that the fission cross section is specified explicitly in MT18.

Problems occur, when rates of specific reactions should be calculated which are included in MT5. This is not possible in MCNP. This implies, that calculation of *e.g.* neutron absorption or α production is not possible. A solution for this problem would be the additional use of MT101 (neutron disappearance cross section) above 20 MeV.

Furthermore, the redundant production cross sections MT201 to MT207 could be used. Neutron producing reactions could be grouped in MT5. This would enable the user of MCNP to calculate absorption and production rates explicitly, whereas no reduction of information occurs compared to the original evaluation.

B. BNL

The 1 GeV BNL evaluation contains data for ^{56}Fe, ^{208}Pb and ^{209}Bi.

Cross-section data below 20 MeV were taken from the ENDF/B-VI evaluation. Data above 20 MeV result from nuclear model calculations. Data are represented using MT2, MT5 and MT18. Besides, the redundant production cross sections MT201 to MT207 are used.

No distributions for reaction products are given for MT5 (only the product yields are supplied). This conceptual flaw implies, that neutron and γ-production cannot be calculated in MCNP, although formal processing of the data file with NJOY is possible.

Minor modifications are needed in NJOY in order to process data at 1 GeV.

Furthermore, the MT456 section (*i.e.* ν_p) is missing, which implies that fission cannot be taken into account in NJOY.

C. ECN

The 150 MeV ECN/Bruyères-le-Châtel library currently contains data for ^{54}Fe ^{56}Fe, ^{58}Ni ^{60}Ni.

Cross-section data below 20 MeV were taken from the EFF-3.0 or ENDF/B-VI evaluations. Data above 20 MeV result from nuclear model calculations.

The data file, made according to the rules of the previous section, can be processed by NJOY (with a few minor modifications, see Appendix A) without problems.

Also for the ECN evaluation, it is suggested to use MT101 (neutron disappearance cross section) above 20 MeV and to make use of the redundant production cross sections MT201 to MT207. This would enable the user of MCNP to calculate absorption and production rates explicitly, whereas no reduction of information occurs compared to the original evaluation.

CONCLUSIONS

We conclude that it is rather straightforward to extend 20 MeV files to 150 MeV, and to retain the possibility of processing the file, by NJOY, into an MCNP-library. The LANL and ECN high-energy neutron data files can be processed by NJOY and used in MCNP4A. The BNL evaluation suffers from a conceptual problem, and cannot be used for production of libraries for MCNP, though the datafiles may be useful for other purposes.

In extending the datafile to 150 MeV, the part below 20 MeV is left untouched, apart from a transformation from a Legendre expansion into tabular format for the elastic angular distribution (MF4/MT2). We also stress that we have not studied recoils in this paper.

Recommendations resulting from this work are:

- Use double-points in the MF3 section of the evaluation for the transition of the low-energy range ($E_n < 20$ MeV) to the high-energy range ($E_n > 20$ MeV).

- Double-points should not be used in the MF6 section of the evaluation.

- Use a representation with MT2 (elastic cross section), MT5 ((n,x) cross section), MT18 (fission cross section) and possibly MT101 (neutron disappearance cross section) in the high-energy region.
- Use the redundant production cross sections MT201 to MT207 for the ease of the user: It enables the calculation of production rates in MCNP.
- Specify distributions of reaction products in the MF6 section of the evaluation for MT5.
- If MT18 is used, the MT456 section in MF1 is required.

If these recommendations are followed, high-energy neutron evaluated nuclear data files can be produced which can be used without problems in MCNP4A.

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[8] P.G. Young and M.B. Chadwick, contribution to the conference on Nuclear Data for Science and Technology, Trieste, May 19-24 1997.

[9] P.G. Young, private communication.

APPENDIX: APPENDIX A: NJOY UPDATES IN ACER AND HEATR

We present a few changes in NJOY94.035 that were necessary for high-energy file processing. We hope the used update-syntax is descriptive enough to make the changes.

The patches in ACER are:

```
*ident ecn_a41
*/ ----------------------------------------------
*/ module pttab
*/             prevent overflow
*/             23 oct 1996 ah/ajk/hlo
*/ ----------------------------------------------
*d acer.3229
      pjk=p(j)/p(k)
      if(pjk.eq.1.0)then
         write(nsyso,8)
         pjk=1.0000001
      endif
      b=log(pjk)/(amu(j)-amu(k))
      if(b.gt.600.0)then
         write(nsyso,9)
         b=600.0
      endif
      if(b.lt.-600.0)then
         write(nsyso,9)
```

```
         b=-600.0
      endif
*i acer.3327
    8 format(/39h ---note from pttab---log(1) error    .)
    9 format(/39h ---note from pttab---cosine error    .)
```

and

```
*ident ecn_a44
*/ -----------------------------------------------
*/ module
*/             25 oct 1996 ah/ajk/hlo
*/ -----------------------------------------------
*d acer.2306
      e2=1.e10
*d acer.3916
      if (enext.gt.1.e10) go to 150
```

The update in HEATR is:

```
*ident ecn_he8
*/ -----------------------------------------------
*/ module
*/             25 oct 1996 ah/ajk/hlo
*/ -----------------------------------------------
*d heatr.590
      if (enext.gt.9.e9) nen=-nen
*d heatr.596
      if (enext.lt.1.e10) go to 170
```