Dunn, Michael E.

From: christopher.dean@serco.com
Sent: Thursday, November 01, 2007 1:55 PM
To: jean-christophe.sublet@cea.fr; rugama@nea.fr
Cc: cullen1@llnl.gov; soppera@nea.fr
Subject: Re: [Fwd: Re: Be-9 from JEFF3.1]

Yolanda

The Be-9 (n,2n) reaction is most important as a neutron multiplier in Fusion.
It is also important is some thermal reactors and elsewhere.

JEFF3.1 has adopted the EFF file where there are cross sections in MF=3, MT=875 to 891 and a detailed description of neutron emission spectra in MF=6, MT= 875 to 891.

The fusion experts believe that a neutron emission spectra cannot be designed to associate with MT=16. The data must be input in MT= 875 to 891.

The ENDF format rules state that if a cross section for MT=16 is present there must be associated emission spectra either in MF=5 or MF=6.

I would like this restriction removed.

This would allow cross sections for MT=16 to be present in a file. They would have to be the sum of MT=875 to 891. (present rule) but there need not be emission data associated with MT=16.

The present JEFF3.1 file contains no MT=16 data and the total cross section has been made too small because these are not present.

Without MT=16 present we are finding people producing Be-9 processed files without ANY (n,2n)!

I note that most files contain MT=4 cross sections without secondary energy data. MT=4 cross sections are the sum of inelastic levels and the inelastic continuum. The secondary data are associated with the levels and continuum not MT=4.

I am suggesting the MT=16 data are similarly represented IF there are MT= 875 to 891 data present.

This would allow a future JEFF3.2 file to have a correct total cross section.

Chris

The reaction MT=16 (n,2n) cross section is

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5. FILE 5, ENERGY DISTRIBUTION OF SECONDARY PARTICLES

5.1. General Description

File 5 is used to describe the energy distributions of secondary particles expressed as normalized probability distributions. File 5 is for incident neutron reactions and spontaneous fission only, and should not be used for any other incident particle. Data will be given in File 5 for all reaction types that produce secondary neutrons, unless the secondary neutron energy distributions can be implicitly determined from data given in File 3 and/or File 4. No data will be given in File 5 for elastic scattering (MT=2), since the secondary energy distributions can be obtained from the angular distributions in File 4. No data will be given for neutrons that result from excitation of discrete inelastic levels when data for these reactions are given in both File 3 and File 4 (MT=51, 52, ..., 90).

Data should be given in File 5 for MT=91 (inelastic scattering to a continuum of levels), MT=18 (fission), MT=16 (n,2n), MT=17 (n,3n), MT=455 (delayed neutrons from fission), and certain other nonelastic reactions that produce secondary neutrons. The energy distribution for spontaneous fission is given in File 5 (in sub-library 4).

File 5 may also contain energy distributions of secondary charged particle for continuum reactions where only a single outgoing charged particle is possible (MT=649, 699, *etc.*). Continuum photon distributions should be described in File 15.

The use of File 6 to describe all particle energy distributions is preferred when several charged particles are emitted or the particle energy and angular distribution are strongly correlated. In these cases Files 5 and 15 should not be used.

Each section of the file gives the data for a particular reaction type (MT number). The sections are then ordered by increasing MT number. The energy distributions $p(E \rightarrow E')$, are normalized so that

$$\int_0^{E'_{max}} p(E \rightarrow E')dE' = 1 \qquad (5.1)$$

where E'_{max} is the maximum possible secondary particle energy and its value depends on the incoming particle energy E and the analytic representation of $p(E \rightarrow E')$. The secondary particle energy E' is always expressed in the *laboratory system*.

The differential cross section is obtained from

$$\frac{d\sigma(E \rightarrow E')}{dE'} = m\sigma(E)p(E \rightarrow E') \qquad (5.2)$$

where $\sigma(E)$ is the cross section as given in File 3 for the same reaction type number (MT) and m is the neutron multiplicity for this reaction (m is implicit; *e.g.*, m=2 for n,2n reactions).

The energy distributions $p(E \rightarrow E')$ can be broken down into partial energy distributions, $f_k(E \rightarrow E')$, where each of the partial distributions can be described by different analytic representations;

$$p(E \rightarrow E') = \sum_{k=1}^{NK} p_k(E) f_k(E \rightarrow E') \qquad (5.3)$$

and at a particular incident neutron energy E,

$$\sum_{k=1}^{NK} p_k(E) = 1$$

where $p_k(E)$ is the fractional probability that the distribution $f_k(E \rightarrow E')$ can be used at E.

Dunn, Michael E.

From: Christopher Dean [christopher.dean@serco.com]
Sent: Thursday, May 31, 2007 9:26 AM
To: Dunn, Michael E.
Cc: B Thom; Jean-Christophe Sublet; Ray Perry
Subject: ENDF Format and the JEFF Be-9 file

Mike

On page 5.1 of the ENDF-102 it states File 5 data must be present for MT16 (n,2n).
The ENDF checking codes comment if it is not present.

I would like this requirement removed because:- In the JEFF3.1 file for Be-9 the secondary energy distribution is described in MF=6, MT=875 .... 890.
The evaluators have found it impossible to describe a single secondary distribution to associate with MF=6 MT=16 that reproduces the physics adequately.
Hence there is no MF=6, MT=16.
Hence (because of the rule I would like removed) there can be no MF=3, MT=16.
Hence the total cross section has been reduced by the (n,2n) (making it wrong but consistent).

This has to be fixed by special processing of this nuclide.

I would like to suggest the (n,2n) data are treated in the same way as total inelastic if any of the levels (875 - 889) or continuum 890 are present.

Could you please discuss this issue at the appropriate WPEC/CSEWG committee.

Chris

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