Storage of numerical neutron source spectra
(Action 25)

The averaged quantity data compiled in EXFOR have to be supplied by the numerical data of incident particle spectra.

Analysis of data, included in EXFOR, have shown that information about neutron spectra, is practically absent in EXFOR.

LEXFOR, Spectrum Average:
Cross sections averaged over a broad incident-projectile energy spectrum may be entered into EXFOR using the proper modifier to REACTION SF8.

Old:
The type of spectrum and its characteristic should be entered **in free text** under the information-identifier keyword **INC-SPECT**.

New:
The type of spectrum and its characteristic should be entered **in numeric data type** using **separate SUBENTRY** or **ENTRY** for neutron spectrum if the spectrum is commonly applied to measurements performed at the neutron source.

1. **Use** special form of **REACTION** to define the neutron source (see table below) with the proper modifier **SPD** to REACTION **SF8**.

2. **Use DATA** to enter **the numerical spectral data.**

Data, **that are averaged** over broad incident-projectile energy spectrum entered into the EXFOR System, should be labeled by the keyword **INC-SOURCE** with usage of all relevant keywords from the Inc-Source Dictionary **(#19)** and **the cross-reference** to the EXFOR entry/subentry with **these numerical spectral data.**
This cross-reference must be coded as an eight-digit integer.

The special form of REACTION to define the neutron spectrum

Name of neutron source	**In INC-SOURCE** (Dictionary #19)	**SF1-SF8 in REACTION in** Entry/SubEntry with spectrum
Alpha-Beryllium	**A-BE**	**4-BE-9(A, X)0-NN-1,,DE,,SPD**
Spont.fission of Californium-252	**CF252**	**98-CF-252(0,F) ,, NU/DE,,SPD**
Spont.fission of Curium-244	**CM244**	**96-CM-244(0,F),, NU/DE,,SPD**
Spont.fission of Curium-246	**CM246**	**96-CM-246(0,F),, NU/DE,,SPD**
Spont.fission of Curium-248	**CM248**	**96-CM-248(0,F),,NU/DE,,SPD**
Deuteron-Beryllium	**D-BE**	**4-BE-9(D,X)0-NN-1,,DE,,SPD**
Deuteron-Carbon 12	**D-C12**	**6-C-12(D,X)0-NN-1,,DE,,SPD**
Deuteron-Carbon 14	**D-C14**	**6-C-14(D,X)0-NN-1,,DE,,SPD**
Deuteron-Deuterium	**D-D**	**1-H-2(D,X)0-NN-1,,DE,,SPD**
Deuteron-Lithium	**D-LI**	**3-LI-0(D,X)0-NN-1,,DE,,SPD**
Deuteron-Lithium 7	**D-LI7**	**3-LI-7(D,X)0-NN-1,,DE,,SPD**
Deuteron-Nitrogen 14	**D-N14**	**7-N-14(D,X)0-NN-1,,DE,,SPD**

Deuteron-Nitrogen 15	**D-N15**	**7-N-15(D,X)0-NN-1,,DE,,SPD**
Deuteron-Tritium	**D-T**	**1-H-3(D,X)0-NN-1,,DE,,SPD**
Evaporation neutrons	**EVAP**	**13-Al-27(P,X)0-NN-1,,DE,,SPD** **74-W-0(P,X)0-NN-1,,DE,,SPD** **82-Pb-0(P,X) 0-NN-1,,DE,,SPD** **92-U-0(D,X) 0-NN-1,,DE,,SPD** **...**
Nuclear explosive device	**EXPLO**	**???**
Proton-Beryllium	**P-BE**	**4-BE-9(P, X)0-NN-1,,DE,,SPD**
Proton-Deuterium	**P-D**	**1-H-2(P,X)0-NN-1,,DE,N,SPD**
Photo-neutron	**PHOTO**	**1-H-2(G,X)0-NN-1,,DE,,SPD** **13-Al-27(G,X)0-NN-1,,DE,,SPD** **74-W-0(G,X)0-NN-1,,DE,,SPD** **92-U-0(G,X)0-NN-1,,DE,,SPD** **...**
Proton-Lithium 7	**P-LI7**	**3-LI-7(P,X)0-NN-1,,DE,,SPD**
Polarized neutron source	**POLNS**	**???**
Proton-Tritium	**P-T**	**1-H-3(P,X)0-NN-1,,DE,,SPD**
Spont.fission of Plutonium-240	**PU240**	**94-PU-240(0,F) ,, NU/DE,,SPD**
Spont.fission of Plutonium-242	**PU242**	**94-PU-242(0,F) ,, NU/DE,,SPD**
Reactor	**REAC**	**92-U-FUL(X,X) 0-NN-1,,DE,,SPD**
Thermal column	**THCOL**	**???**

The fields **SF8, SF9** in **REACTION** may be used to indicate, if this spectrum is given in relative (SPD/**REL**) values, and if it was obtained by calculation (**CALC**).

In **COMMENT** or in the line with **REACTION** we can write the additional information about given spectrum: codes and libraries used in calculations, components of the used neutron filter, etc.

To introduce separate special ENTRY/SUBENTRY for neutron spectrum allows:

1) To refrain from repetition of neutron spectrum information in Entries with data, obtained with the same neutron source spectrum.

2) To facilitate data search of neutron source spectrum.
It may be found using the modifier **SPD** in REACTION **SF8** through link with EXFOR Service.
It is important for experimenters, evaluators, compilers.

To demonstrate an example of such entries, let us consider subentry 32217003, where the filtered neutron spectrum was used in measurements of the averaged radiation cross section on Ta. There were two types of spectrum: calculated neutron spectrum and experimental one, obtained by differentiation of the instrumental proton recoil spectrum.

We can use **one new entry** (take for example **32**777) for the filtered neutron spectra. In subentry **32**777001, as usual, we describe general information using keywords TITLE, AUTHOR, INSTITUTE, ….

ENTRY	32777	20110408				32777	0	1
SUBENT	32777001	20110408				32777	1	1
BIB	7	10				32777	1	2
TITLE	Measurements of neutron capture cross-section for					32777	1	3
	tantalum at the neutron filtered beams					32777	1	4
AUTHOR	(O.Gritzay,V.Libman,A.V.Chyzh,V.F.Razbudey)					32777	1	5
INSTITUTE	(4UKRIJD)					32777	1	6
REFERENCE	(C,2008KYIV,,548,2008) Result on 59 keV was					32777	1	7
	presented at the NPAE-Kyiv2008,ID# 86-95.					32777	1	8
FACILITY	(REAC,4UKRIJD) Reactor WWR-M					32777	1	9
INC-SOURCE	(REAC) Neutron filters installed in horizontal channel					32777	1	10
	of the reactor.					32777	1	11
HISTORY	(20110408) UKRNDC					32777	1	12
ENDBIB	10	0				32777	1	13
NOCOMMON	0	0				32777	1	14
ENDSUBENT	13	0				32777	199999	

For calculated neutron spectrum we can use subentry **32**777**002**, for experimental one it may be used the subentry **32**777**003**. To note that this neutron spectrum was created from the reactor spectrum at the reactor used uranium fuel, we propose to write the fields **SF1-SF4** in **REACTION** as **92-U-FUL(X,X)0-NN-1**.

SUBENT	32777002	20110408				32777	2	1
BIB	2	8				32777	2	2
REACTION	(92-U-FUL(X,X)0-NN-1,,DE,,SPD/REL,CALC) Using JENDL-3.3					32777	2	3
					and CENDL-2	32777	2	4
COMMENT	Calculation was done by FILTER.5 using JENDL-3.3 for					32777	2	5
	Ni-58(83.15 g/cm2),V(24.44 g/cm2),Al(5.4 g/cm2),					32777	2	6
	B-10(0.5 g/cm2), and using CENDL-2 for S(147.78 g/cm2)					32777	2	7
	Calculated energy line is 58.9 keV, purity about 99%.					32777	2	8
	The limits of 95% response function for the 59 keV					32777	2	9
	filter spectrum were defined as 52.2 to 60.1 keV.					32777	2	10
ENDBIB	8	0				32777	2	11
NOCOMMON	0	0				32777	2	12
DATA	2	1543				32777	2	13
E	DATA					32777	2	14
EV	ARB-UNITS					32777	2	15
50000.15	7.05730E-11					32777	2	16
50019.84	7.85371E-11					32777	2	17
50039.52	8.42285E-11					32777	2	18
...								
63945.89	6.82944E-26					32777	2	1557
64018.24	7.01394E-26					32777	2	1558
ENDDATA	1545	0				32777	2	1559
ENDSUBENT	1558	0				32777	299999	
SUBENT	32777003	20110408				32777	3	1
BIB	2	6				32777	3	2
REACTION	(92-U-FUL(X,X)0-NN-1,,DE,,SPD/REL)					32777	3	3
COMMENT	Ni-58(83.15 g/cm2),V(24.44 g/cm2),Al(5.4 g/cm2),					32777	3	4
	B-10(0.5 g/cm2), and S(147.78 g/cm2) were used as					32777	3	5
	filter components.					32777	3	6
	Experimental shape was obtained by differentiation of					32777	3	7
	the instrumental proton recoil spectrum LND-281.					32777	3	8
ENDBIB	6	0				32777	3	9
NOCOMMON	0	0				32777	3	10
DATA	3	431				32777	3	11
E	DATA	DATA-ERR				32777	3	12
EV	ARB-UNITS	ARB-UNITS				32777	3	13
48793.33	0.487	0.008				32777	3	14
48831.11	0.698	0.013				32777	3	15

In COMMENT or in the line with REACTION we can write the additional information about given spectrum: libraries used in calculations, components of the used neutron filter, etc.

To give information about the used neutron spectrum in the subentry 32217003 with the measured average cross section data we can use the keyword

INC-SOURCE (**REAC, 32**777**002**) and (**REAC, 32**777**003**).

SUBENT	32217003 20110318 20110323 20110323 3148	32217	3	1
BIB	4 24	32217	3	2
REACTION	(73-TA-181(N,G)73-TA-182,,SIG,,SPA)	32217	3	3
INC-SOURCE	(REAC,32777002) Calculated neutron spectrum	32217	3	4
	(REAC,32777003) Experimental neutron spectrum	32217	3	5
ANALYSIS	For determination of sample activities, nine gamma	32217	3	6
	lines of W-182 were selected: 152, 179, 222, 229, 264,	32217	3	7
	1121, 1189, 1221 and 1231 keV.	32217	3	8
ERR-ANALYS	(ERR-T)Absolute uncertainty of cross section-	32217	3	9
	it includes the uncertainties of-	32217	3	10
	(ERR-1)Error in extrapolated cross-section - it	32217	3	11
	includes the uncertainties of-	32217	3	12
	error in determination of sample activities 1.3-4.4%	32217	3	13
	statistical error in peak area 1.2-21.1%	32217	3	14
	error in gamma-line efficiency 4.2%	32217	3	15
	error in quantum yield gamma-lines 0.05-0.48	32217	3	16
	(ERR-2)Error in determination of Ta sample mass 0.057%	32217	3	17
	(ERR-3)Error in determination of neutron flux - it	32217	3	18