Neutron source spectra format

History:

https://www-nds.iaea.org/publications/group_list.php?group=INDC-NDS INDC(NDS)-0590

EXFOR formats and rules:
present status and proposals how to store neutron source data

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Proposal for the source spectral information storage in EXFOR, suggested in this paper

2) To introduce into EXFOR **new special ENTRY/SUBENTRY** for neutron spectrum.

New: 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.

2.1 Use special form of **REACTION** to define the neutron source (see table below) with the proper modifier **SPD** to REACTION **SF8**.
2.2 Use **DATA** to enter **the numerical spectral data.**
2.3 **Data**, **that are averaged** over broad incident-projectile energy spectrum and entered into the EXFOR system, should be labelled by the keyword **INC-SOURCE** with use of all relevant keywords from the Inc-Source Dictionary (#19) and **the cross-reference** to the EXFOR entry/subentry with **these numerical spectral data.**
2.4 This cross-reference must be coded as an eight-digit integer.

Unfortunately, this proposal was not realized.

We continue to use the following rules.

LEXOR:

Spectrum Average

Cross sections averaged over a broad incident-projectile energy spectrum may be entered into EXFOR using the proper modifier to REACTION SF8[4]. **The type of spectrum and its characteristic should be entered in free text under the information-identifier keyword INC-SPECT.**

The following spectrum types are defined:

1. **Maxwellian Average**: Modifier MXW
   The spectrum temperature should be given, if known. For thermal Maxwellian spectrum averaged data, see **Thermal Neutron Energies**.
2. **Epithermal Spectrum Average**: Modifier EPI
   The energy quoted will be, typically the low energy cutoff.
3. **Fission-Neutron Spectrum Average**: Modifier FIS
   For details, see **Fission-Neutron Spectra**.
4. **Fast Reactor Spectrum Average**: Modifier FST
5. **Bremsstrahlung Spectrum Average**: Modifier BRA
   The energy quoted will be, typically, EN-MAX,or EN-MIN with EN-MAX.
6. **Average over "good resolution" Bremsstrahlung Spectrum**: Modifier BRS
   The energy quoted will be, typically, EN with EN-RSL.
7. **Spectrum Average (unspecified Spectrum)**: Modifier SPA
   Used for all other spectra, *e.g.*, thermal reactor spectra. Care should be taken to compile only those data that would be of value to the user of EXFOR.

EXFOR:

Chapter 7
INFORMATION-IDENTIFIER KEYWORDS AND CODING RULES

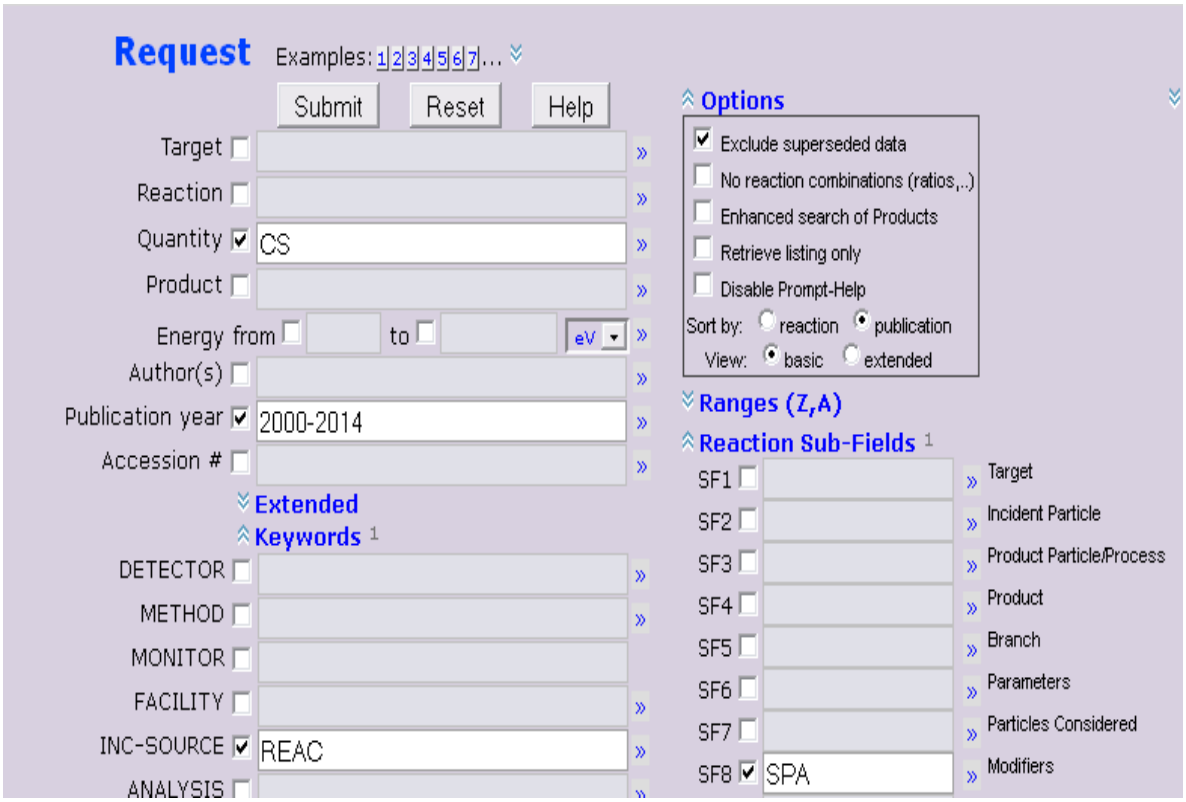
INC-SPECT. Provides information on the characteristics and resolution of the incident-projectile beam. See also **LEXFOR, Incident-Projectile Energy**.

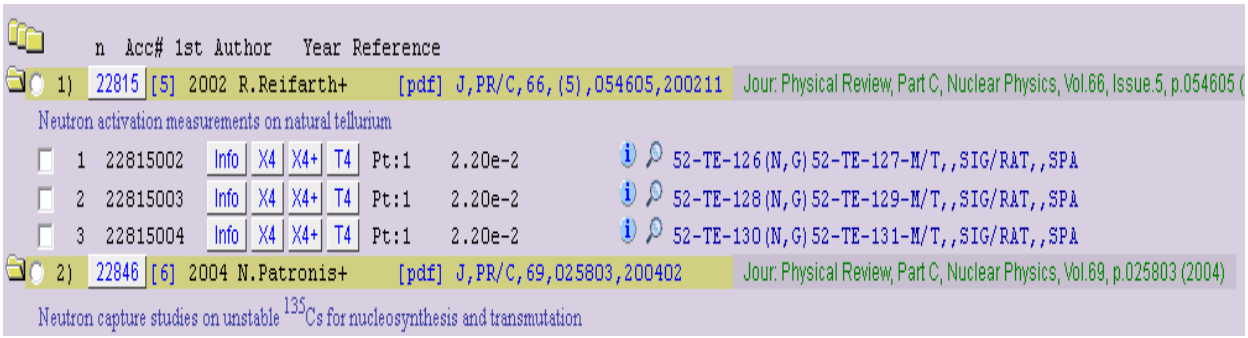
1. Must be present when a spectrum average modifier (*e.g.*, MXW, SPA, or FIS) is present in REACTION SF8. See also **LEXFOR, Spectrum Average**. Otherwise its use is optional. No coded information.

…

1. How can we find Entry with a spectrum presentation?

https://www-nds.iaea.org/exfor/exfor.htm

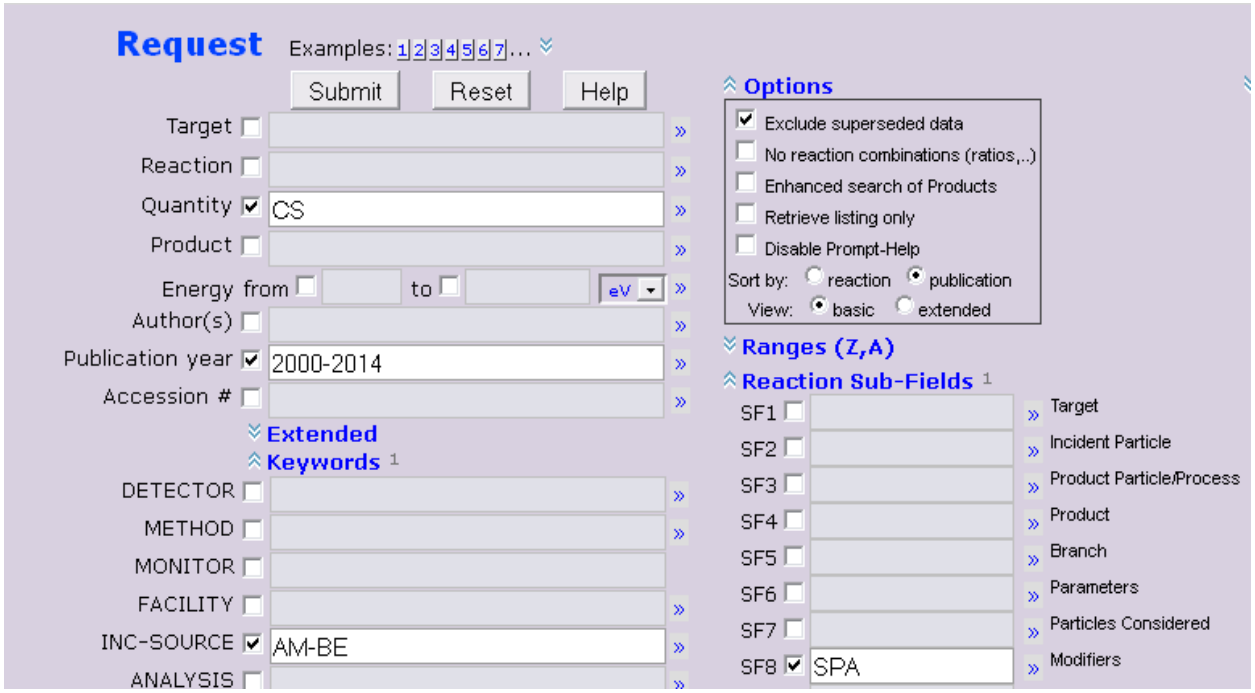


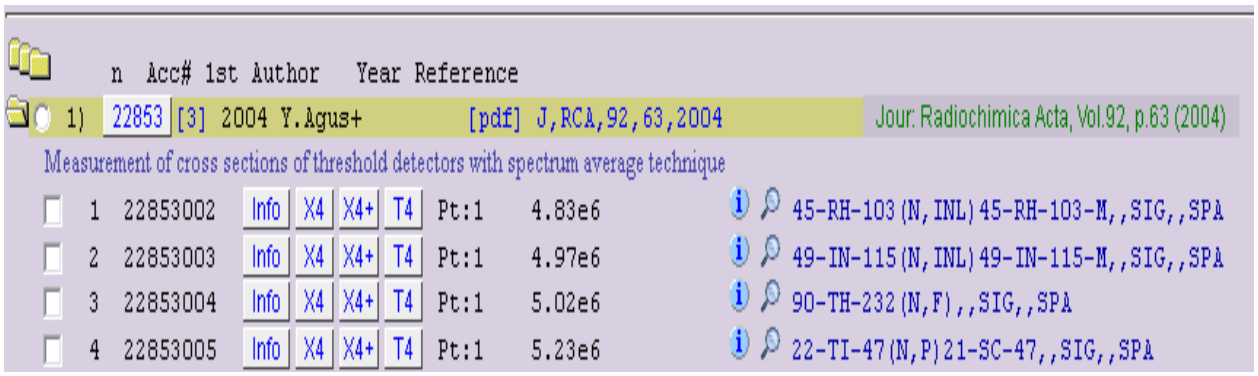


…

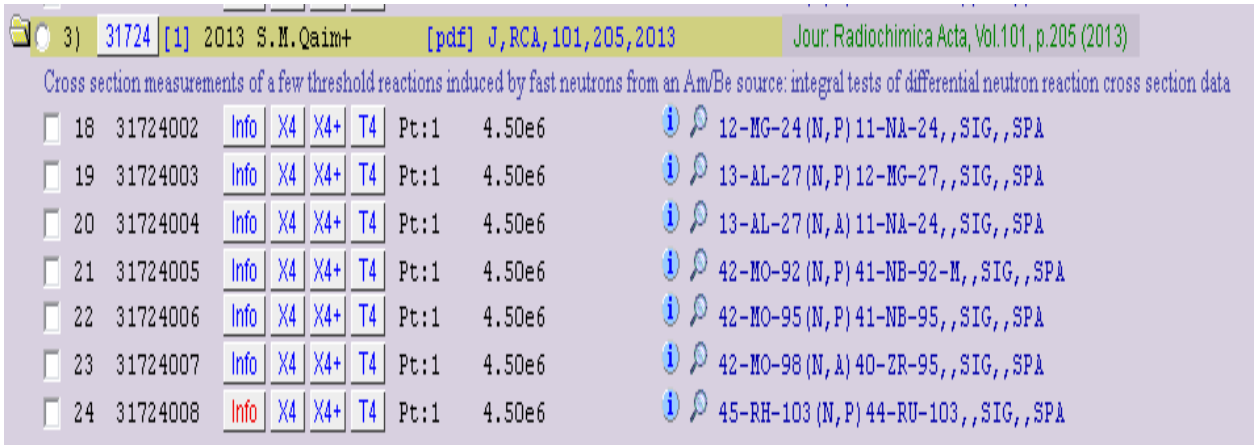
14 Entries. Only two entries (31733 and 23075) have spectrum presentation.

Without Publication year Results: Entries: **114**



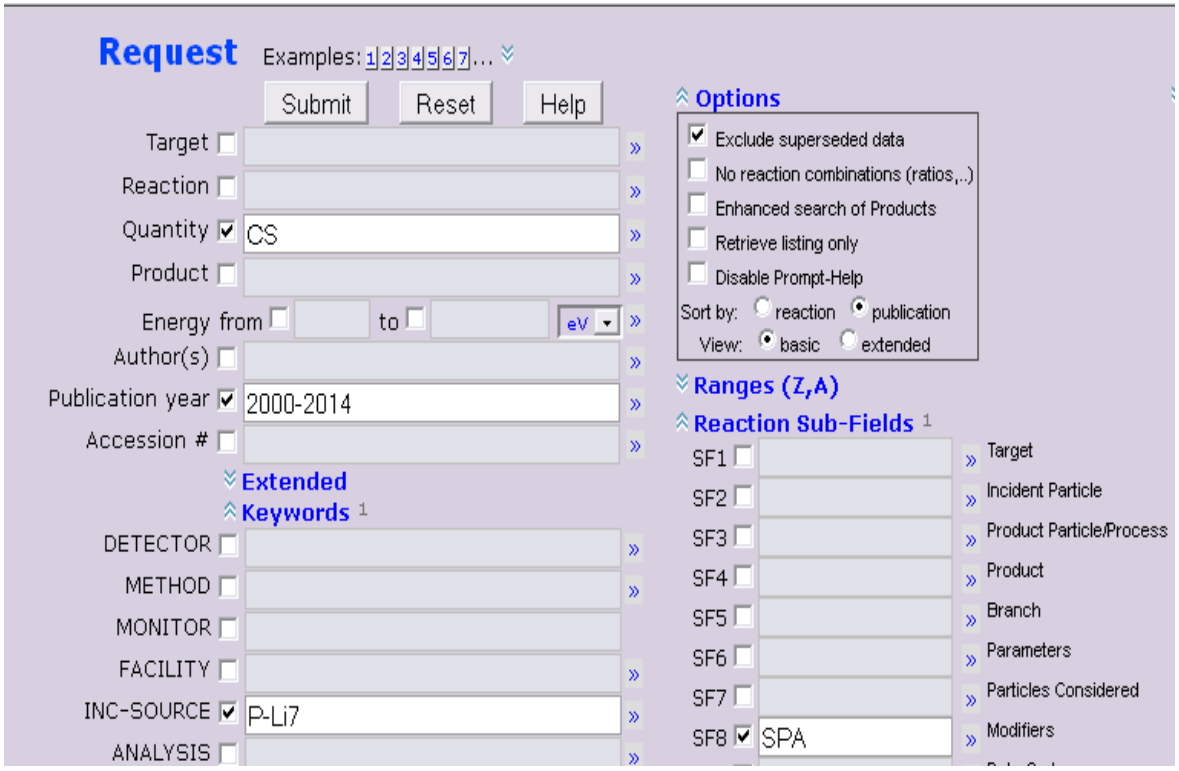


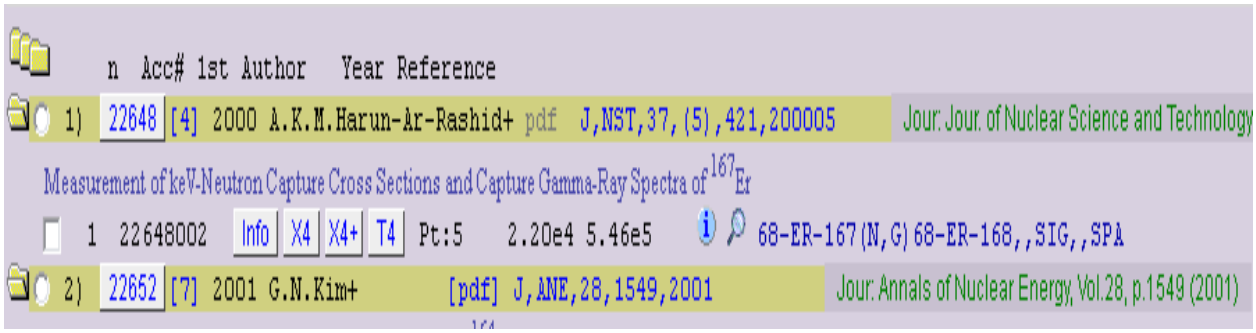
...



3 Entries. One entry (31724) has spectrum presentation.

Without Publication year Results: Entries: 4





...

24 Entries. One entry (22850) has spectrum presentation.
Without Publication year Results: Entries: 42

So, spectrum retrieval is not easy.
It takes a lot of time.

2. Can we plot this spectrum using

Plot: ☐ Quick plot ☑ Advanced plot ?

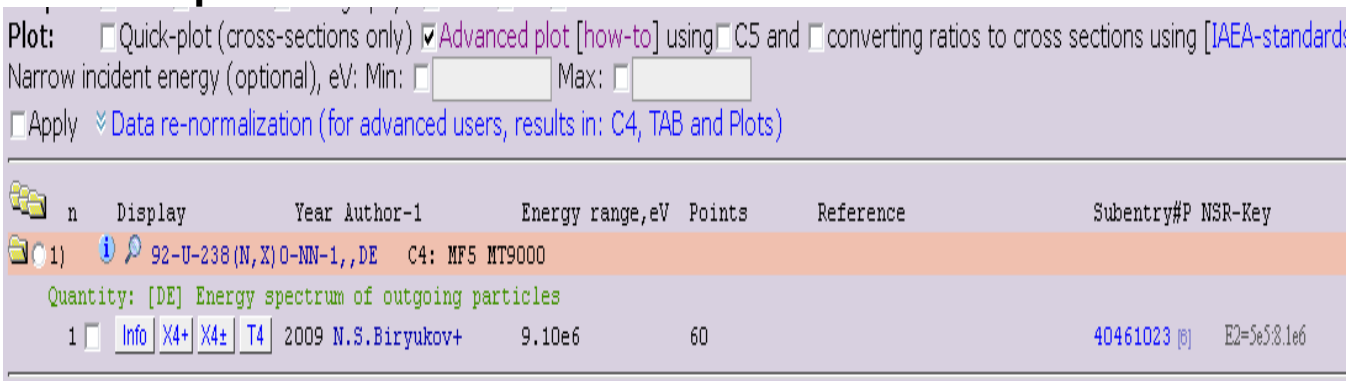
No, we can not plot.

If for spectrum presentation

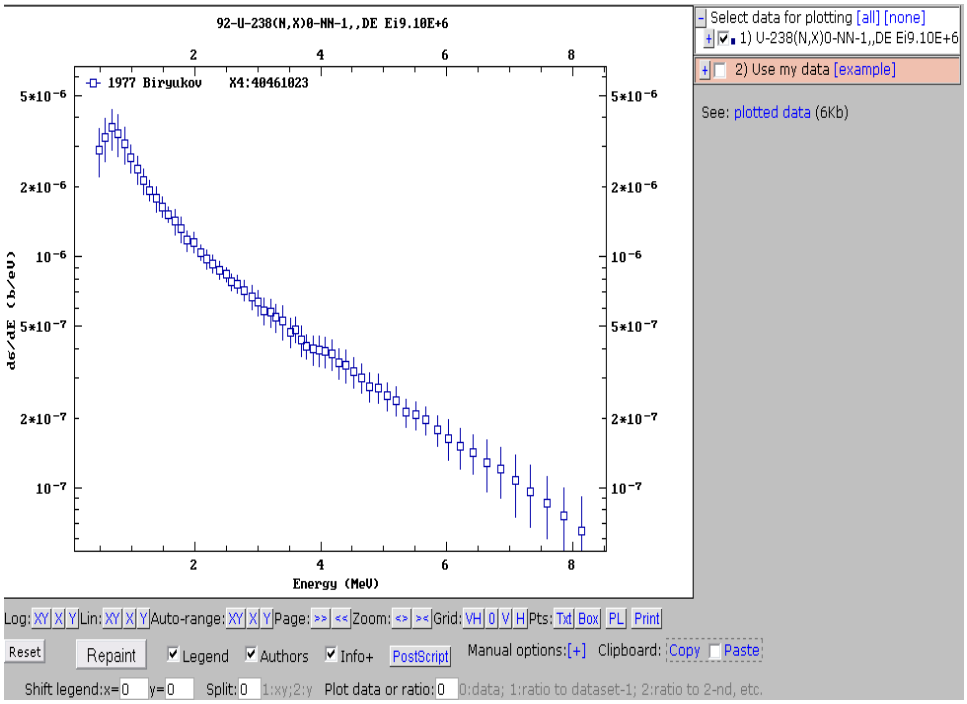
we use REACTION with SF6 ☑ DE

we can plot this spectrum.

For example:



REACTION (92-U-238(N,X)0-NN-1,,DE)



NB ! **…,,DE,,REL** and **…,,DE,,SPA/REL**

```
Sorry, "Advanced plotting" was not successful...
     Possible reasons:
      - plotting of the given data type is not yet implemented
```

3. Often this free text is too long.

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

For example,

in Subentry 22850001 it has 699 lines (from 769 lines);

in Subentry 23075001 it has 230 lines (from 326 lines).

If information about calculated and experimental spectra are available (e.g., for average cross sections measured by proton recoil counters using the NFBT), length of the text can be up to thousands of lines.

If several spectra are in this free text it is very difficult to find where is the end or the start of these spectra.

It is not convenient.

Proposal for the source spectral information storage in EXFOR

New for EXFOR, the same in INDC(NDS)-0590

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.

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

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

2.3 **Data**, **that are averaged** over broad incident-projectile energy spectrum and entered into the EXFOR system, should be labelled by the keyword **INC-SOURCE** with use of all relevant keywords from the Inc-Source Dictionary (#19) and **the cross-reference** to the EXFOR entry/subentry with **these numerical spectral data.**

2.4 This cross-reference must be coded as an eight-digit integer.

Now

LEXOR:

Spectrum Average

Cross sections averaged over a broad incident-projectile energy spectrum may be entered into EXFOR using the proper modifier to REACTION SF8[4]. The type of spectrum and its characteristic should be entered **in free text** under the information-identifier keyword INC-SPECT or

in numeric data type using separate special SUBENTRY or ENTRY.

Data, that are averaged over broad incident-projectile energy spectrum and entered into the EXFOR system, should be labelled with the code DEP under the keyword STATUS, if spectrum is entered into the EXFOR system in numeric data type.

In these special SUBENTRY/ ENTRY

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;**

Dependent Data

Data that are deduced by a trivial operation from other data sets entered into the EXFOR System or data, that are averaged over broad incident-projectile energy spectrum entered into the EXFOR system in numeric data type (using special SUBENTRY/ENTRY) should be labeled with the code DEP under the keyword STATUS. Free text under STATUS and/or ANALYSIS should give information as to how the data were deduced. Cross-reference to the EXFOR entries from which the data were deduced or to the EXFOR entries with spectrum in numeric data type must be coded as an eight-digit integer following the code.

We leave Exfor rule for INC-SPECT and INC-SOURCE (no changes)

The special form of REACTION to define the neutron source

REACTION: It is a reaction by which neutrons appear.
SF1 – target, SF2 – incident particle (SF2=0 for spontaneous fission),
SF3 – outgoing particle/product X (SF3=F for spontaneous fission),
SF4 – outgoing neutron 0-NN-1,
SF5 – empty,
SF6 – Differential with energy of outgoing particle (neutron),
SF7 – empty,
SF8 – SF8=SPD – SPectrum Description, SPD/REL Relative data (ARB-UNITS)
SF9 – CALC/EXP Calculated/Experimental data

Spont. Fission FIS (not SPA)

Name of neutron source	In INC-SPECT	SF1-SF8 in REACTION in SubEntry with SF8= SPD
Alpha-Beryllium	A-BE	4-BE-9(A,X)0-NN-1,, DE,,SPD
Americium-Beryllium ns	AM-BE	4-BE-9(A,X)0-NN-1,, DE,,SPD ?
Spont.fission of Californium-252	CF252	98-CF-252(0,F)0-NN-1,, DE,,SPD
Spont. fission of Curium-244	CM244	96-CM-244(0,F) 0-NN-1,, DE,,SPD
Spont. fission of Curium-246	CM246	96-CM-246(0,F) 0-NN-1,, DE,,SPD
Spont. fission of Curium-248	CM248	96-CM-248(0,F) 0-NN-1,, 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-0(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	Entries with SPA are absent in EXFOR. Time-of-flight method was used
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,,SPD
Photo-neutron	PHOTO	1-H-2(G,X) 0-NN-1,, DE,,SPD 13-Al-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	Entries with SPA are absent in EXFOR.
Proton-Tritium	P-T	1-H-3(P,X) 0-NN-1,, DE,,SPD
Plutonium-Beryllium ns	AM-BE	4-BE-9(A,X) 0-NN-1,, DE,,SPD ?
Spont. fission of Plutonium-240	PU240	94-PU-240(0,F) 0-NN-1,, DE,,SPD
Spont. fission of Plutonium-242	PU242	94-PU-242(0,F) 0-NN-1,, DE,,SPD
Reactor	REAC	92-U-235(N,X) 0-NN-1,, DE,,SPD or 92-U-0(N,X) 0-NN-1,, DE,,SPD
Thermal column	THCOL	92-U-235(N,X) 0-NN-1,, DE,,SPD
Thorium-Beryllium ns	TH-BE	4-BE-9(A,X) 0-NN-1,, DE,,SPD ?

```
ENTRY            32238   20140507                                    32238  0     1
SUBENT        32238001   20140507                                    32238  1     1
BIB                12         35                                    32238  1     2
INSTITUTE  (4UKRIJD,4UKRUKR)4UKRUKR - State Scientific-Engineering32238  1     3
           Center for Control and Emergency Response, Kyiv         32238  1     4
REFERENCE  (C,2012KYIV,,430,2013)                                  32238  1     5 Include!
           #NSR 2013GRZY                                           32238  1     6
AUTHOR     (O.O.Gritzay, A.K.Grymalo, V.V.Kolotyi,                 32238  1     7
           V.A.Pshenychnyi, V.P.Shakhov, V.M.Venedyktov)           32238  1     8
TITLE      Determination of total neutron cross section of 52Cr    32238  1     9
           with using average energy shift method for filtered     32238  1    10
           neutron beam                                            32238  1    11
FACILITY   (REAC,4UKRIJD) Reactor WWR-M                            32238  1    12
INC-SOURCE (REAC) Neutron filter installed in horizontal channel   32238  1    13
           of the reactor. Filter components: S-116.53 g/cm2,      32238  1    14
           58Ni-81.42g/cm2,V-24.44g/cm2,Al-5.4g/cm2,10B-0.5g/cm2.  32238  1    15
           The calculated energy and width of the neutron line     32238  1    16
            (95% response function) after this filter are:         32238  1    17
           59 (+1.2, -6.7) keV.                                    32238  1    18
           Calculated and experimental spectra after this filter   32238  1    19
           are presented in 32238002 and 32238003.                 32238  1    20
INC-SPECT  The initial neutron line with the average energy 59 keV32238  1    21
           produces at the scattering angles 15, 20 and 25 ADEG    32238  1    22
           the scattered neutron lines with the average energies   32238  1    23
           55, 52, 48.4 keV  and 58.6, 58.3, 58 keV if neutrons    32238  1    24
           are scattered by hydrogen and carbon, respectively.     32238  1    25
           Thickness of the used scattering-samples C and CH2 was  32238  1    26
           10.02+-0.01 and 4.52+-0.01 mm.                          32238  1    27
METHOD     (FNB,TRN) Transmission measurement of the scattered     32238  1    28
           filtered neutrons.                                      32238  1    29
DETECTOR   (PROPC) The proton recoil detector LND-281 (gas filling32238  1    30
           H+CH4+N2, diameter-38.1 mm, length-254.0 mm,            32238  1    31
           gas pressure-3240 torr)                                 32238  1    32
SAMPLE     52Cr sample was made of a metal powder, loaded into     32238  1    33
           aluminum container. A thickness of the 52Cr sample was  32238  1    34
           0.0173+-0.0002 nucl/barn.                               32238  1    35
STATUS     (TABLE) From text                                       32238  1    36
HISTORY    (20140425) UKRNDC                                       32238  1    37
ENDBIB           35          0                                     32238  1    38
NOCOMMON          0          0                                     32238  1    39
ENDSUBENT        38          0                                     32238  199999
SUBENT        32238002   20140507                                    32238  2     1
BIB               1          8                                     32238  2     2
REACTION   (92-U-235(N,X)0-NN-1,,DE,,SPA/REL,CALC)                 32238  2     3
           Calculated was done by FILTER-7 using JENDL-3.3         32238  2     4
            (58Ni-81.42 g/cm2, 10B-0.5 g/cm2, 11B-0.088 g/cm2,     32238  2     5
           27Al-5.4 g/cm2, Vnat-24.44 g/cm2),                      32238  2     6
           CENDL-2(Snat-116.53 g/cm2). Calculated energy line is   32238  2     7
           58.98 keV, purity about 95%. The limits of 95% responce32238  2     8
           function for the 59 keV filter spectrum were defined as32238  2     9
           52.111 to 60.319 keV.                                   32238  2    10
ENDBIB            8          0                                     32238  2    11
COMMON            2          3                                     32238  2    12
EN-MIN     EN-MAX                                                  32238  2    13
EV         MEV                                                     32238  2    14
1.e-5      20.                                                     32238  2    15
ENDCOMMON         3          0                                     32238  2    16
DATA              2       1201                                     32238  2    17
E          DATA                                                    32238  2    18
KEV        ARB-UNITS                                               32238  2    19
  40.5499  .318100E-01                                             32238  2    20
  40.6795  .370900E-01                                             32238  2    21
...
  61.5046  .689100E-01                                             32238  2  1219
  61.5322  .474500E-01                                             32238  2  1220
ENDDATA        1203          0                                     32238  2  1221
ENDSUBENT      1220          0                                     32238  299999
```

```
SUBENT          32238003    20140507                                    32238   3     1
BIB                    1           1                                    32238   3     2
REACTION    (92-U-235(N,X)0-NN-1,,DE,,SPA/REL,EXP)                      32238   3     3
ENDBIB                 1           0                                    32238   3     4
COMMON                 2           3                                    32238   3     5
EN-MIN      EN-MAX                                                      32238   3     6
EV          MEV                                                         32238   3     7
1.e-5       20.                                                         32238   3     8
ENDCOMMON              3           0                                    32238   3     9
DATA                   3         240                                    32238   3    10
E           DATA        DATA-ERR                                       32238   3    11
KEV         NO-DIM      NO-DIM                                         32238   3    12
39.9696     2.8626      0.0320                                         32238   3    13
40.0953     2.7958      0.0301                                         32238   3    14
...
69.8826     1.0589      0.1184                                         32238   3   251
70.0083     0.9470      0.0985                                         32238   3   252
ENDDATA              242           0                                    32238   3   253
ENDSUBENT            252           0                                    32238 399999
SUBENT          32238004    20140507                                    32238   4     1
BIB                    5          24                                    32238   4     2
REACTION    (24-CR-52(N,TOT),,SIG,,SPA)                                 32238   4     3
INC-SOURCE  (REAC)                                                      32238   4     4
STATUS      (DEP,32238002)Calculated neutron spectrum after filter      32238   4     5
            (DEP,32238003)Experimental neutron spectrum                 32238   4     6
            after filter                                                32238   4     7
            (DEP,32238005) Scattered neutron spectrum at the angle      32238   4     8
            15 ADEG (55 keV) calculated by the MCNP 4C code             32238   4     9
            (DEP,32238006) Scattered neutron spectrum at the angle      32238   4    10
            20 ADEG (52 keV) calculated by the MCNP 4C code             32238   4    11
            (DEP,32238007)Scattered neutron spectrum at the angle       32238   4    12
            25 ADEG (48.8 keV) calculated by the MCNP 4C code           32238   4    13
COMMENT     Correction connected with existence in the neutron          32238   4    14
            scattered spectrum on CH2 neutrons scattered on carbon      32238   4    15
            was done. A correction on the self-shielded effect was      32238   4    16
            not be done for the measured values of the total cross      32238   4    17
            sections of 52Cr. The rough estimation, done by the         32238   4    18
            MCNP 4C code calculations, shown that the difference        32238   4    19
            between the total cross sections at these energies and      32238   4    20
            the observed self-shielded cross sections may rich 2-6%     32238   4    21
ERR-ANALYS  Total error includes:                                       32238   4    22
            statistical error of the transmission measurements;         32238   4    23
            measurement errors of dimension and weight of the           32238   4    24
            sample; error due to presence of impurities in the          32238   4    25
            sample.                                                     32238   4    26
ENDBIB                24           0                                    32238   4    27
NOCOMMON               0           0                                    32238   4    28
DATA                   3           3                                    32238   4    29
EN          DATA         ERR-T                                          32238   4    30
KEV         B            B                                              32238   4    31
48.8        5.94         0.49                                           32238   4    32
52.         18.29        0.89                                           32238   4    33
55.         9.27         0.51                                           32238   4    34
ENDDATA                5           0                                    32238   4    35
ENDSUBENT             34           0                                    32238 499999
SUBENT          32238005    20140507                                    32238   5     1
BIB                    1           3                                    32238   5     2
REACTION    (92-U-235(N,X)0-NN-1,,DE,,SPA/REL,CALC)                     32238   5     3 <-SPD
            Scattered neutron spectrum at the angle                     32238   5     4
            15 ADEG (55 keV) calculated by the MCNP 4C code             32238   5     5
ENDBIB                 3           0                                    32238   5     6
COMMON                 2           3                                    32238   5     7
EN-MIN      EN-MAX                                                      32238   5     8
EV          MEV                                                         32238   5     9
1.e-5       20.                                                         32238   5    10
ENDCOMMON              3           0                                    32238   5    11
DATA                   2         248                                    32238   5    12
E           DATA                                                        32238   5    13
KEV         ARB-UNITS                                                   32238   5    14
40.006      0.108                                                       32238   5    15
```

```
40.087      0.095                                                    32238  5   16
…
59.969      0.065                                                    32238  5  261
60.050      0.063                                                    32238  5  262
ENDDATA              250          0                                  32238  5  263
ENDSUBENT            262          0                                  32238  599999
SUBENT         32238006    20140507                                  32238  6    1
BIB                    1          3                                  32238  6    2
REACTION    (92-U-235(N,X)0-NN-1,,DE,,SPA/REL,CALC)                  32238  6    3 <-SPD
            Scattered neutron spectrum at the angle                  32238  6    4
            20 ADEG (52 keV) calculated by the MCNP 4C code          32238  6    5
ENDBIB                 3          0                                  32238  6    6
COMMON                 2          3                                  32238  6    7
EN-MIN      EN-MAX                                                   32238  6    8
EV          MEV                                                      32238  6    9
1.e-5       20.                                                      32238  6   10
ENDCOMMON              3          0                                  32238  6   11
DATA                   2        248                                  32238  6   12
E           DATA                                                     32238  6   13
KEV         ARB-UNITS                                                32238  6   14
40.006      0.116                                                    32238  6   15
40.087      0.156                                                    32238  6   16
…
59.969      0.038                                                    32238  6  261
60.050      0.010                                                    32238  6  262
ENDDATA              250          0                                  32238  6  263
ENDSUBENT            262          0                                  32238  699999
SUBENT         32238007    20140507                                  32238  7    1
BIB                    1          3                                  32238  7    2
REACTION    (92-U-235(N,X)0-NN-1,,DE,,SPA/REL,CALC)                  32238  7    3 <-SPD
            Scattered neutron spectrum at the angle                  32238  7    4
            25 ADEG (48.8 keV) calculated by the MCNP 4C code        32238  7    5
ENDBIB                 3          0                                  32238  7    6
COMMON                 2          3                                  32238  7    7
EN-MIN      EN-MAX                                                   32238  7    8
EV          MEV                                                      32238  7    9
1.e-5       20.                                                      32238  7   10
ENDCOMMON              3          0                                  32238  7   11
DATA                   2        309                                  32238  7   12
E           DATA                                                     32238  7   13
EV          ARB-UNITS                                                32238  7   14
35.056      0.110                                                    32238  7   15
35.137      0.115                                                    32238  7   16
…
59.969      0.031                                                    32238  7  322
60.050      0.014                                                    32238  7  323
ENDDATA              311          0                                  32238  7  324
ENDSUBENT            323          0                                  32238  799999
ENDENTRY               7          0                                  3223899999999
```

```
ZCHEX (Version 2012-03-21) run on 01-Oct-2014
 Input file: E:\!!!!!!AfterMay2014\EXFOREditor\After_May2014\32238s.EXF
 ENTRY 32238
   First pass completed with no fatal errors
1
 - Second pass checking -
 ENTRY 32238
  **  Illegal code field                                            32238001
       REFERENCE  (C,2012KYIV,,430,2013)                           32238001    5
                  ^^^^^^^^
```

Statistics

ERRORS: 0 WARNINGS: 1

Error Type	SUBENTRY	Error Message	Details	Line
WARNING	32238001	[REFERENCE] : Bad Conference reference coding, unknown Conference code	2012KYIV	8