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FENDL/MG-2.0 and FENDL/MC-2.0

The processed cross-section libraries for neutron photon transport calculations

Version 1, March 1997

Summary documentation

by

H. Wienke and M. Herman

Abstract: Evaluated neutron reaction data and photon-atom interaction cross sections for materials contained in the general purpose Fusion Evaluated Nuclear Data Library (FENDL/E-2.0) have been processed with the NJOY code system into VITAMIN-J multigroup structure, for use in discrete-ordinates transport codes, and into continuous energy ACE format, for use in the Monte Carlo transport code MCNP. This document summarizes the resulting data libraries FENDL/MG-2.0 version 1 and FENDL/MC-2.0 version 1. The data are available costfree from the IAEA Nuclear Data Section online or on magnetic tape.

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FENDL/MG-2.0 AND FENDL/MC-2.0

The processed cross-section libraries for
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Version 1 of February 1998

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Summary

The data libraries FENDL/MG-2.0 and FENDL/MC-2.0 contain cross section data in multigroup and continuous-energy ACE format respectively, derived from FENDL/E-2.0, a library of selected evaluated neutron-nucleus and photon-atom interaction cross-sections for nuclides of importance for neutron-photon transport calculations for fusion reactor design, in the energy range from 10^{-5} eV up to 20 MeV [Ref. 1]. The library FENDL/E-2.0 has been developed from the previous version FENDL/E-1.1 [Ref. 2] by including thirteen replacing evaluations and six additional materials. The new evaluations were selected from candidate evaluations submitted by five national projects viz. JENDL-FF (Japan), BROND (Russia), EFF (European Union), ENDF/B-VI (USA) and CENDL (China). The multigroup and continuous-energy cross-sections have been produced with the nuclear data processing system NJOY [Ref. 3]. For the evaluations taken over of FENDL/E-1 the processed cross section data, produced for FENDL/MG-1.0 and FENDL/MC-1.0 by R.E. MacFarlane [Refs. 4,5,6]), were included. The processed data for the new evaluations were provided by the contributing projects. Whenever necessary, these data were supplemented at IAEA/NDS using NJOY94.105.

A. Multigroup cross section data library FENDL/MG-2.0

The subdirectory [FENDL2.TRANSPORT.PROCESSED.FENDLMG] contains the following files:

- GENDF formatted files (filename extension .G), output of the NJOY module GROUPR (neutron-interaction and photon-production cross section data only).
- MATXS formatted files (filename extension .M), for use in the coupled neutron-photon transport calculations with discrete-ordinate codes such as ANISN, ONEDANT, etc. These MATXS files were generated by post-processing the .G files mentioned above, using the MATXSR module of the NJOY code system, along with the GENDF files of photon-atom interaction data. The latter, produced with the NJOY module GAMINR, are not included in the present subdirectory.
- A file named AAAREADME.WWW which contains the summary of information presented in this report.

The multigroup data library FENDL/MG-2.0 has a total size of 332.1 Megabytes (uncompressed) and contains 57 data files in GENDF format (198.4 Megabytes) and the same number of files in MATXS format (133.7 Megabytes). Each data file corresponds to one material. The list of available GENDF and MATXS formatted data files is given in Appendix A. Appendix C describes the MATXS format.

Specifications:

The specifications for processing the FENDL/E-2.0 library into the multigroup coupled neutron-photon library are:

- Neutron groups: 175, in Vitamin-J structure
- Gamma groups: 42, in Vitamin-J structure
- Neutron weight function: VITAMIN-E (IWT=11 in NJOY)
- Gamma weight function: 1/E with roll-offs (IWT=3 in NJOY)
- Legendre order for neutrons and photons:
  P-6 for transport calculations correction to P-5
- Temperatures:
  300 Kelvin (plus 600, 900, 1500 Kelvin for data from JENDL-FF)
- Dilution factors: see Table 1
- Reconstruction, linearization and thinning tolerances used in RECONR: 0
- Accuracy for resonance reconstruction up to 7 digits
- Reactions included:
  - All reactions contained in the evaluations
  - Energy balance heating (MT-301)
  - Kinematic heating (MT=443)
  - Damage (MT=444)
  - Thermal data only for H-2, Be-9, C-12, N-14, O-16, Al-27, V-51, Fe-56, Zr-nat, Ga-nat, Nb-93, Mo-nat, Sn-nat, W-nat, Au-197
  - Gas production data: see Table 2

B. Library of continuous-energy data FENDL/MC-2.0

The subdirectory [FENDL2.TRANSPORT.PROCESSED.FENDLMC] contains neutron and coupled neutron-photon pointwise cross section data files in the ACE (ASCII) format, with filename extension ACE, intended for use in the Monte-Carlo code MCNP. The data have been derived with the ACER module of the NJOY system. Files with extension .XSDIR consist of one line with the correct "file" and "route" entries. These .XSDIR files have also been included in the index file XSDIR-FENDL used by the MCNP code.

The specifications used in processing with NJOY/ACER are:

- Temperature: 300 Kelvin = 2.585E-8
- No thinning
- Reactions included:
  - All reactions present in evaluations
  - Damage (MT = 444, only for Al-27, Fe-56, Ta-181 and Au-197)
  - No thermal data
  - Gas production data (see Table 2)

The library has a total size of 149.5 Megabytes (uncompressed). It contains 57 data files. Each data file corresponds to one material. The list of available ACE formatted data files is given in Appendix B.

C. File name conventions in FENDL/MG-2.0 and FENDL/MC-2.0:

In both sublibraries (FENDL/MG-2.0 and FENDL/MC-2.0) the file name convention is the same as the one in FENDL/E-2.0 eg:

"H002BR2.M,-.G,-.ACE,-.XSDIR" H-2 data from BROND-2;
"SI030E6.M,-.G,-.ACE,-.XSDIR " Si-30 data from ENDF/B-VI;
"MO000JFF.M,-.G,-.ACE,-.XSDIR " Mo-nat data from JENDL-FF;
"FE056EFF3.M,-.G,-.ACE,-.XSDIR " Fe-56 data from EFF-3;
"TA181J3.M,-.G,-.ACE,-.XSDIR " Ta-181 data from JENDL-3.1.

The atomic number is always written with three digits, starting with '0' when smaller than 100 and '00' when smaller than 10.

The inputs for NJOY, as used by R.E. MacFarlane, the contributors of the new evaluations and by the IAEA/NDS, for generating the multigroup and ACE data files, are given in Appendix D. These should help the users to check, repeat or extend the work.

Table 1: Dilution factors (in barns) in the multigroup data files

Nuclide	10^{10}	10^{5}	10^{4}	10^{3}	300.	100.	30.	10.	3.	1.	0.3	0.1	.001
H-1	X												
H-2	X		X	X		X		X		X			
H-3	X												
He-3	X												
He-4	X												
Li-6	X												
Li-7	X												
Be-9	X		X	X	X	X	X	X	X	X		X	X
B-10	X												
B-11	X												
C-12	X		X	X	X	X	X	X	X	X		X	X
N-14	X		X	X	X	X	X	X	X	X		X	X
N-15	X												
O-16	X		X	X	X	X	X	X	X	X		X	X
F-19	X												
Na-23	X												
Mg-nat	X												
Al-27	X												
Si-28	X					X		X		X			
Si-29	X					X		X		X			
Si-30	X					X		X		X			
P-31	X												
S-nat	X			X	X	X	X	X					
Cl-nat	X			X	X	X	X	X					
K-nat	X			X	X	X	X	X					
Ca-nat	X			X	X	X	X	X	X	X			
Ti-nat	X			X	X	X	X	X	X	X			
V-51	X		X	X	X	X	X	X	X	X		X	X
Cr-50	X			X	X	X	X	X	X	X			
Cr-52	X			X	X	X	X	X	X	X			
Cr-53	X			X	X	X	X	X	X	X			
Cr-54	X			X	X	X	X	X	X	X			
Mn-55	X	X		X		X		X		X			
Fe-54	X	X	X	X		X		X	X	X	X	X	
Fe-56	X	X	X	X		X		X					
Fe-57	X	X	X	X		X		X					
Fe-58	X	X	X	X		X		X					
Co-59	X	X	X	X		X		X					
Ni-58	X			X	X	X	X	X	X	X			
Ni-60	X			X	X	X	X	X	X	X			
Ni-61	X			X	X	X	X	X	X	X			
Ni-62	X			X	X	X	X	X	X	X			
Ni-64	X			X	X	X	X	X	X	X			
Cu-63	X		X		X	X	X	X					
Cu-65	X		X		X	X	X	X					
Ga-nat	X		X	X	X	X	X	X	X	X		X	X
Zr-nat	X		X	X	X	X	X	X	X	X		X	X
Nb-93	X		X	X	X	X	X	X	X	X		X	X
Mo-nat	X		X	X	X	X	X	X	X	X		X	X
Mo-nat	X		X	X	X	X	X	X	X	X		X	X
Sn-nat	X		X	X		X		X		X			
Ta-191	X		X	X		X		X					
W-nat	X		X	X	X	X	X	X	X	X		X	X
Au-197	X		X	X	X	X	X	X		X			
Pb-206	X			X		X		X		X			
Pb-207	X			X		X		X		X			
Pb-208	X			X		X		X		X			
Bi-209	X		X	X		X		X					

Table 2: Status of gas production cross sections present in the multigroup and MCNP formatted libraries

Nuclide	MT203 H-1	MT204 H-2	MT205 H-3	MT206 He-3	MT207 He-4
H-2	X	X			
Be-9					X
C-12					X
N-14	X	X	X		X
N-15	X	X	X		X
O-16	X	X			X
Al-27	X				X
Si-28	X	X			X
Si-29	X				X
Si-30	X				X
V-51	X	X	X		X
Fe-56	X	X	X	X	X
Ga-nat	X	X	X	X	X
Zr-nat	X	X	X	X	X
Nb-93	X	X			X
Mo-nat	X	X	X	X	X
Sn-nat	X	X	X	X	X
W-nat	X	X			X
Ta-181[*)]	X				X
Au-197	X				X

References

1. A.B. Pashchenko, and H. Wienke, "FENDL/E-2.0, Evaluated nuclear data library of neutron nuclear interaction cross-sections and photon production cross-sections and photon-atom interaction cross-sections for fusion applications, version 1 of March 1997", report IAEA-NDS-175 Rev. 0 (International Atomic Energy Agency, March 1997).

2. A.B. Pashchenko, H. Wienke, S. Ganesan and P.K. McLaughlin, "FENDL/E, Evaluated Nuclear Data Library of Neutron Interaction Cross Sections, Photon Production Cross Sections and Photon-Atom Interaction Cross Sections for Fusion Applications, version 1.1 of November 1994", IAEA(NDS)-128, Rev. 3 (February 1996).

3. R.E. MacFarlane, "The NJOY Nuclear Data Processing System, Version 91", Los Alamos National Laboratory report LA-12740-M (1994). Code and manual distributed as package PSR-171 by the Radiation Shielding Information Center (RSIC), Oak Ridge National Laboratory, Oak Ridge, USA.

4. R.E. MacFarlane, "FENDL/MG, library of multigroup cross-sections in GENDF and MATXS format for neutron-photon transport calculations, version 1.1 of March 1995". Summary documentation by A.B. Pashchenko, H. Wienke and S. Ganesan, report IAEA-NDS-129 Rev. 3 (International Atomic Energy Agency, February 1996).

[*)] Not present in :FENDL/MG-2.0

5. R.E. MacFarlane, "Processing of ENDF/B-VI and FENDL for Multigroup and Monte Carlo Applications", paper presented at the IAEA Advisory Group Meeting "Review of Uncertainty Files and Improved Multigroup Cross Section Files for FENDL", Tokai, Japan, 8-12 November 1993 [Ref. 7].

6. R.E. MacFarlane, "FENDL/MC, Library of cross-sections in continuous-energy ACE format for neutron-photon transport calculations with the Monte Carlo N-particle Transport Code system MCNP 4A, version 1.1 of March 1995" summary documentation by A.B. Pashchenko, H. Wienke and S. Ganesan, report IAEA-NDS-169 Rev. 3 (International Atomic Energy Agency, November 1995).

7. S. Ganesan, Ed., "Improved Evaluations and Integral Data Testing for FENDL", summary report of the IAEA Advisory Group Meeting organized by the International Atomic Energy Agency in cooperation with the Max-Planck Institute für Plasmaphysik, Garching, Germany, 12-16 September 1994, report INDC(NDS)-312, Nuclear Data Section, International Atomic Energy Agency, Vienna, Austria (Dec 1994).

8. Judith F. Briesmeister, Ed, "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A", Los Alamos National Laboratory report LA-12625-M (1993).

9. R.E. MacFarlane, "TRANSX 2: A Code for Interfacing MATXS Cross Section Libraries to Nuclear Transport Codes," LA-12312-MS (1992).

Appendix A: List of MATXS and GENDF formatted files

Ftp data set: UD7:[FENDL2.TRANSPORT.PROCESSED.FENDLMG]*.*

Nuclide	Filename	Original library	No. of blocks (1 blk=512 bytes)
^{1}H	H001E6.G	FENDL/MG-1.1	3367
	H001E6.M		2584
^{2}H	H002BR2.G	BROND-2	7831
	H002BR2.M		6286
^{3}H	H003E6.G	FENDL/MG-1.1	1683
	H003E6.M		1151
^{3}He	HE003E6.G	FENDL/MG-1.1	1356
	HE003E6.M		742
^{4}He	HE004E6.G	FENDL/MG-1.1	1077
	HE004E6.M		607
^{6}Li	LI006E6.G	FENDL/MG-1.1	6311
	LI006E6.M		4860
^{7}Li	LI007E6.G	FENDL/MG-1.1	5688
	LI007E6.M		4385
^{9}Be	BE009JFF.G	JENDL-FF	15302
	BE009JFF.M		11331
^{10}B	B010E6.M	FENDL/MG-1.1	3142
	B010E6.G		4723
^{11}B	B011E6.G	FENDL/MG-1.1	3422
	B011E6.M		2426
^{12}C	C012JFF.G	JENDL-FF	14173
	C012JFF.M		10381
^{14}N	N014JFF.G	JENDL-FF	15911
	N014JFF.M		11708
^{15}N	N015BR2.G	BROND-2.1	3746
	N015BR2.M		2460
^{16}O	O016JFF.G	JENDL-FF	13693
	O016JFF.M		9944
^{19}F	F019E6.G	FENDL/MG-1.1	7989
	F019E6.M		5998
^{23}Na	NA023E6.G	FENDL/MG-1.1	4300
	NA023E6.M		2902
Mg-nat	MG000E6.G	FENDL/MG-1.1	5161
	MG000E6.M		3385
^{27}Al	AL027EFF3.G	EFF-2.4	7005
	AL027EFF3.M		3877
^{28}Si	SI028E6.G	ENDF/B-6.1	5412
	SI028E6.M		3524
^{29}Si	SI029E6.G	ENDF/B-6.1	6144
	SI029E6.M		4018
S-nat	S000BR2.G	BROND-2.1	5537
	S000BR2.M		3483

Nuclide	Filename	Original library	No. of blocks (1 blk=512 bytes)
Cl-nat	CL000E6.G	FENDL/MG-1.1	2976
	CL000E6.M		1783
K-nat	K000E6.G	FENDL/MG-1.1	3045
	K000E6.M		1848
Ca-nat	CA000J3.G	FENDL/MG-1.1	6702
	CA000J3.M		4399
Ti-nat	TI000J3.G	FENDL/MG-1.1	6846
	TI000J3.M		4400
^{51}**V**	V051JFF.G	JENDL-FF	11128
	V051JFF.M		7232
^{50}**Cr**	CR050E6.G	FENDL/MG-1.1	5900
	CR050E6.M		4199
^{52}**Cr**	CR052E6.G	FENDL/MG-1.1	5989
	CR052E6.M		4201
^{53}**Cr**	CR053E6.G	FENDL/MG-1.1	6919
	CR053E6.M		4829
^{54}**Cr**	CR054E6.G	FENDL/MG-1.1	4696
	CR054E6.M		3207
^{55}**Mn**	MN055E6.G	FENDL/MG-1.1	8241
	MN055E6.M		5574
^{54}**Fe**	FE054E6.G	FENDL/MG-1.1	5435
	FE054E6.M		3916
^{56}**Fe**	FE056EFF3.G	EFF-3	11278
	FE056EFF3.M		5788
^{57}**Fe**	FE057E6.G	FENDL/MG-1.1	6557
	FE057E6.M		4755
^{58}**Fe**	FE058E6.G	FENDL/MG-1.1	5065
	FE058E6.M		3721
^{59}**Co**	CO59E6.G	FENDL/MG-1.1	3985
	CO59E6.M		2359
^{58}**Ni**	NI058E6.G	FENDL/MG-1.1	6393
	NI058E6.M		4585
^{59}**Ni**	NI059E6.G	FENDL/MG-1.1	233
	NI059E6.M		126
^{60}**Ni**	NI060E6.G	FENDL/MG-1.1	6545
	NI060E6.M		4616
^{61}**Ni**	NI061E6.G	FENDL/MG-1.1	6916
	NI061E6.M		4858
^{62}**Ni**	NI062E6.G	FENDL/MG-1.1	5569
	NI062E6.M		3966
^{64}**Ni**	NI064E6.G	FENDL/MG-1.1	5034
	NI064E6.M		3603
^{63}**Cu**	CU063E6.G	FENDL/MG-1.1	8452
	CU063E6.M		5862
^{65}**Cu**	CU065E6.G	FENDL/MG-1.1	7000
	CU065E6.M		4864
Ga-nat	GA000J3.G	JENDL-3.2	12504
	GA000J3.M		7430

Nuclide	Filename	Original library	No. of blocks (1 blk=512 bytes)
Zr-nat	ZR000JFF.G	JENDL-FF	12843
	ZR000JFF.M		8022
^{93}Nb	NB093JFF.G	JENDL-FF	11836
	NBB093JFF.M		7298
Mo-nat	MO000JFF.G	JENDL-FF	13677
	MO000JFF.M		8539
Sn-nat	SN000BR2.G	BROND-2	7314
	SN000BR2.M		4732
^{181}Ta	TA181J3.G	FENDL/MG-1.1	4974
	TA181J3.M		2943
W-nat	W000JFF.G	JENDL-FF	14093
	W000JFF.M		8539
^{207}Pb	PB207E6.G	FENDL/MG-1.1	6852
	PB207E6.M		4324
^{208}Pb	PB208E6.G	FENDL/MG-1.1	4305
	PB208E6.M		2878
^{209}Bi	BI209E6.G	FENDL/MG-1.1	3101
	BI209E6.M		1918

Appendix B: List of NJOY/ACER output files

Nuclide	Filename	Original library	No. of blocks (1 blk=512 bytes)
^{1}H	H001E6.ACE	FENDL/MG-1.1	142
^{2}H	H002BR2.ACE	BROND-2	1924
^{3}H	H003E6.ACE	FENDL/MG-1.1	136
^{3}He	HE003E6.ACE	FENDL/MG-1.1	116
^{4}He	HE004E6.ACE	FENDL/MG-1.1	121
^{6}Li	LI006E6.ACE	FENDL/MG-1.1	498
^{7}Li	LI007E6.ACE	FENDL/MG-1.1	585
^{9}Be	BE009JFF.ACE	JENDL-FF	3319
^{10}B	B010E6.ACE	FENDL/MG-1.1	1122
^{11}B	B011E6.ACE	FENDL/MG-1.1	4340
^{12}C	C012JFF.ACE	JENDL-FF	1072
^{14}N	N014JFF.ACE	JENDL-FF	2483
^{15}N	N015BR2.ACE	BROND-2.1	1890
^{16}O	O016JFF.ACE	JENDL-FF	1736
^{19}F	F019E6.ACE	FENDL/MG-1.1	3759
^{23}Na	NA023E6.ACE	FENDL/MG-1.1	1776
Mg	MG000E6.ACE	FENDL/MG-1.1	1808
^{27}Al	AL027EFF3.ACE	EFF-2.4	2939
^{28}Si	SI028E6.ACE	ENDF/B-6.1	5631
^{29}Si	SI029E6.ACE	ENDF/B-6.1	4651
^{30}Si	SI030E6.ACE	ENDF/B-6.1	3341
^{31}P	P031E6.ACE	FENDL/MG-1.1	271
S-nat	S000BR2.ACE	BROND-2.1	4354
Cl-nat	CL000E6.ACE	FENDL/MG-1.1	967
K-nat	K000E6.ACE	FENDL/MG-1.1	982
Ca-nat	CA000J3.ACE	FENDL/MG-1.1	4326
Ti-nat	TI000J3.ACE	FENDL/MG-1.1	2699
^{51}V	V051JFF.ACE	JENDL-FF	1854
^{50}Cr	CR050E6.ACE	FENDL/MG-1.1	7316
^{52}Cr	CR052E6.ACE	FENDL/MG-1.1	6868
^{53}Cr	CR053E6.ACE	FENDL/MG-1.1	4666
^{54}Cr	CR054E6.ACE	FENDL/MG-1.1	4025
^{55}Mn	MN055E6.ACE	FENDL/MG-1.1	10415
^{54}Fe	FE054E6.ACE	FENDL/MG-1.1	6919
^{56}Fe	FE056EFF3.ACE	EFF-3	38827

Nuclide	Filename	Original library	No. of blocks (1 blk=512 bytes)
^{57}Fe	FE057E6.ACE	FENDL/MG-1.1	6730
^{58}Fe	FE058E6.ACE	FENDL/MG-1.1	4404
^{59}Co	CO59E6.ACE	FENDL/MG-1.1	8707
^{58}Ni	NI058E6.ACE	FENDL/MG-1.1	12256
^{60}Ni	NI060E6.ACE	FENDL/MG-1.1	7347
^{61}Ni	NI061E6.ACE	FENDL/MG-1.1	3815
^{62}Ni	NI062E6.ACE	FENDL/MG-1.1	3339
^{64}Ni	NI064E6.ACE	FENDL/MG-1.1	2721
^{63}Cu	CU063E6.ACE	FENDL/MG-1.1	7810
^{65}Cu	CU065E6.ACE	FENDL/MG-1.1	6394
Ga-nat	GA000J3.ACE	JENDL-3.2	2381
Zr-nat	ZR000JFF.ACE	JENDL-FF	6353
^{93}Nb	NB093JFF.ACE	JENDL-FF	7839
Mo-nat	MO000JFF.ACE	JENDL-FF	8587
Sn-nat	SN000BR2.ACE	BROND-2	11411
^{181}Ta	TA181J3.ACE	FENDL/MG-1.1	13072
W-nat	W000JFF.ACE	JENDL-FF	9447
^{197}Au	AU197E6.ACE	ENDF/B-6.1 Modl.	11132
^{206}Pb	PB206E6.ACE	FENDL/MG-1.1	10284
^{207}Pb	PB207E6.ACE	FENDL/MG-1.1	4462
^{208}Pb	PB208E6.ACE	FENDL/MG-1.1	4489
^{209}Bi	BI209E6.ACE	FENDL/MG-1.1	3069

Appendix C: Brief description of "MATXS" Format

The MATXS files are produced by MATXSR module of NJOY. The MATXSR module of NJOY reformats multigroup constants from the GENDF tape (i.e the output file of GROUPR for neutron interaction cross sections and photon production cross sections or the output file of GAMINR for photon atomic interaction cross sections) into the MATXS interface format. The MATXS file has a very general organization to hold arbitrary vectors and matrices. The file is first divided into "data types" such as neutron scattering, photon production, gamma scattering, and neutron thermal data. Each data type is assigned a name (NSCAT, NG, GSCAT, NTHERM). Data types are distinguished by the choice of incident and secondary group structures. Each data type is divided into materials specified by nuclide, temperature, and background cross section. Each material is further subdivided into "vector partials" and "matrix partials". These reaction partials are labeled with Hollerith names so there is no limit on the quantities that can be stored. MATXSR reads cross sections from the GENDF tape, assigns the Hollerith names, and packs the cross sections into MATXS format. The code TRANSX 2.0 [Ref. 9], for instance, serves to interface MATXS cross section libraries to nuclear transport codes such as ANISN, ONEDANT etc. TRANSX reads nuclear data from a library in MATXS format and produces transport tables compatible with many discrete-ordinates (S_N) and diffusion codes. The FENDL multigroup library may be post-processed using TRANSX to produce tables for neutron, photon or coupled transport for specific application calculations.

The MATXS format is briefly described below [priv. comm. R.E. McFarlane]:

Material cross section file:

This file contains cross section vectors and matrices for all particles, materials, and reactions; delayed neutron spectra by time group; and decay heat and photon spectra. Formats given are for file exchange only.

File structure:

```
Record type                                              Present if

File identification                                      Always
File control                                             Always
Set hollerith identification                             Always
File data                                                Always

      ***********     (Repeat for all particles)
         Group structures                                Always
      ***********

      ***********(Repeat for all materials)
      *  material control                                Always
      *
      * *********(Repeat for all submaterials)
      * *vector control                                  N1DB.GT.0

Record type                                              Present if

      * *
      * * *******     (Repeat for all vector blocks)
      * * *     Vector block                             N1DB.GT.0
      * * *******
      * *
      * * *******     (Repeat for all matrix blocks)
```

```
* * *     Matrix control                                N2D.GT.0
* * *
* * * *****     (Repeat for all sub-blocks)
* * * *   Matrix sub-block                              N2D.GT.0
* * * *****
* * *
* * *     Constant sub-block                            JCONST.GT.0
* * *
***********
```

File identification

HNAME,(HUSE(I),I=1,2),IVERS

1+3*MULT

FORMAT(4H OV ,A8,1H*,2A8,1H*,I6)

HNAME	Hollerith file name - MATXS -	(A8)
HUSE	Hollerith user identifiation	(A8)
IVERS	File version number	
MULT	Double precision parameter	

1- A8 word is single word
2- A8 word is double precision word

File control:

NPART,NTYPE,NHOLL,NMAT,MAXW,LENGTH

FORMAT(4H 1D ,4I6)

NPART	Number of particles for which group structures are given
NTYPE	Number of data types present in set
NHOLL	Number of words in set hollerith identification record
NMAT	Number of materials on file
MAXW	Maximum record size for sub-blocking
LENGTH	Length of file

Set hollerith identification:

(HSETID(I),I=1,NHOLL)

NHOLL*MULT

FORMAT(4H 2D ,8A8/(9A8))

HSETID Hollerith identification of set (A8)
(To be edited out 72 characters per line)

File data:

(HPRT(J),J=1,NPART),(HTYPE(K),K=1,NTYPE),(HMATN(I),I=1,NMAT),
(NGRP(J),J=1,NPART),(JINP(K),K=1,NTYPE,(JOUTP(K),K=1,NTYPE),
(NSUBM(I)I=1,NMAT),(LOCM(I),I=1,NMAT)

(NPART+NTYPE+NMAT)*MULT+2*NTYPE+NPART+2*NMAT

FORMAT(4H 3D ,8A8/(9A8))	HPRT,HTYPE,HMATN
FORMAT(12I6)	NGRP,JINP,JOUTP,NSUBM,LOCM

```
HPRT(J)    Hollerith identification for particle J
             N       Neutron
             G       Gamma
             P       Proton
             D       Deuteron
             T       Triton
             H       He-3 Nucleus
             A       Alpha (He-4 nucleus)
             B       Beta
             R       Residual or recoil
                     (Heavier than alpha)

HTYPE(K)     Hollerith identification for data type K
             NSCAT   Neutron scattering
             NG      Neutron induced gamma production
             GSCAT   Gamma scattering
             PN      Proton induced neutron production
             DKN     Delayed neutron data
             DKHG    Decay heat and gamma data
             DKB     Decay beta data

HMATN(I)     Hollerith identification for material I
NGRP(J)      Number of energy groups for particle J
JINP(K)      Type of incident particle associated with data type K.
             For dk data types, JINP is 0.
JOUTP(K)     Type of outgoing particle associated with data type K
NSUBM(I)     Number of submaterials for material I
LOCM(I)      Location of material I
```

Group structure:

```
(GPB(I),I=1,NGR),EMIN

NGR=NGRP(J)

NGRP(J)+1

FORMAT(4H 4D ,1P5E12.5/(6E12.5))
GPB(I)       Maximum energy bound for group I for particle J
EMIN         Minimum energy bound for particle J
```

Material control:

```
HMAT,AMASS,(TEMP(I),SIGZ(I),ITYPE(I),N1D(I),N2D(I),LOCS(I),
I=1,NSUBM)MULT+1+6*NSUBM

FORMAT(4H 6D ,A8,1H*,1P2E12.5/(2E12.5,5I6))

HMAT         Hollerith material identifier
AMASS        Atomic weight ratio
TEMP         Ambient temperature or other parameters for submaterial
I.
SIGZ         Dilution factor or other parameters for submaterial I
ITYPE        Data type for submaterial I
N1D          Number of vectors for submaterial I
N2D          Number of matrix blocks for submaterial I
LOCS         Location of submaterial I
```

Vector control:

```
(HVPS(I),I=1,N1D),(NFG(I),I=1,N1D),(NLG(I),I=1,N1D)
(MULT+2)*N1D
FORMAT(4H 7D ,8A8/(9A8))          HVPS
FORMAT(12I6)                      IBLK,NFG,NLG
```

HVPS(I)   Hollerith identifier of vector

NELAS	Neutron elastic scattering
N2N	(n,2n)
NNF	Second chance fission
GABS	Gamma absorption
P2N	Protons in, 2 neutrons out
.	.
.	.
.	.

NFG(I)   Number of first group in band for vector I
NLG(I)   Number of last group in band for vector I

Vector block

(VPS(I),I=1,KMAX)

KMAX=Sum over group band for each vector in block J

KMAX

FORMAT(4H 8D,1P5E12.5/(6E12.5))

VPS(I)   Data for group bands for vectors in block J. The block size is determined by taking all the group bands that have a total length less than or equal to MAXW.

Scattering matrix control:

```
HMTX,LORD,JCONST,
1(JBAND(L),L=1,NOUTG(K)),(IJJ(L),L=1,NOUTG(K))

MULT+2+2*NOUTG(K)

FORMAT(4H 9D ,A8/(12I6))          HMTX,LORD,JCONST,
                                  JBAND,IJJ
```

HMTX	Hollerith identification of block
LORD	Number of orders present
JCONST	Number of groups with constant spectrum
JBAND(L)	Bandwidth for group L
IJJ(L)	Lowest group in band for group L

Scattering sub-block:

(SCAT(K),K=1,KMAX)

MAX=LORD times the sum over all JBAND in the group range of this sub-block
FORMAT(5H 10D ,1P5E12.5/(6E12.5))

KMAX

SCAT(K) : Matrix data given as bands of elements for initial groups that lead to each final group. The order of the elements is as follows: Band for P0 of of group I, band for P1 of group I, ... , band for P0 of group I+1, band for P1 of group I+1, etc. The groups in each band are given in descending order. The size of each sub-block is determined by the total length of a group of bands that is less than or equal to MAXW. If JCONST.GT.0, the contributions from the JCONST low-energy groups are given separately.

Constant sub-block:

(SPEC(L),L=1,NOUTG(K)),(PROD(L),L=L1,NING(K))

L1=NING(K)-JCONST+1

NOUTG(K)+JCONST

FORMAT(4H11D ,1P5E12.5/(6E12.5))

SPEC Normalized spectrum of final particles for initial particles in groups L1 To NING(K)
PROD Production cross section (e.g., NU*SIGF) for initial groups L1 through NING(K)

This option is normally used for the energy-independent neutron and photon spectra from fission and radiative capture usually seen at low energies.

Appendix D: NJOY inputs for generating neutron interaction and photon production cross sections in GENDF, MATXS and ACE format

a) NJOY Input file used by R.E. MacFarlane for generating FENDL/MG-1.0

```
0
5
moder
20-21
reconr
21 22
*pendf tape for jendl3.1 al-27*/
3131 7 0 /
.002 0. 7 /
*13-al-27 from jendl3.1 */
*processed with the njoy nuclear data
processing system*/
*see original endf/b6 tape for details of evaluation*/
*the following reaction types are added*/
*mt301   heating*/
*mt443   kinematic kerma*/
*mt444   damage energy production*/
0/
broadr
22 23
3131 9/
.002/
300 400 600 800 1200 1600 2000 3000 4000 /
0/
heatr
21 23 24
3131 6 0 1 0 2/
302 303 304 402 403 443/
heatr
21 23 24
3131 2/
443 444/
stop

0
6
moder
20 21
groupr
21 22 0 23
3131 17 10 11 6 1 1 1
*al27 jendl3.1 175x42*/
300
1e10
 3/
 3 251 *mubar*/
 3 252 *xi*/
 3 253 *gamma*/
 3 259 *1/v*/
 6/
 16/
0/
0/
matxsr
23 26 25/
15 *t2lanl njoy*/
2 3 1 1/
*al27 jendl3.1 njoy91.91 28nov93*/
n g/
175 42/
nscat ng gscat/
1 1 2/
1 2 2/
al27 3131 1300/
stop
```

b) NJOY input files used for processing data from EFF

NJOY input file for producing multigroup data

```
0
6
'moder'
20 -21
'reconr'
-21 -22
'Pendf tape for Al-27 from FENDL-2'/
1325 2/ mat ncards ngrids
.002 0. 7 .007/ err tempr(0.) ndigit(7) errmax
'Al-27 from FENDL-2 '/
'Processed at IAEA/NDS with NJOY94 '/
0/
'broadr'
-21 -22 -23
1325 1/ mat ntemp istart(0) itrap(0) temp(0.)
.002 2.e6 .01/ err emaxbr errmax(20*err) errin(.0001*err)
300./ temp
0/
'unresr'
-21 -23 -24/ endf ipendf opendf
1325 1 1 1/ mat ntem nsig ipr(0)
300./
1.e10/
0/
'thermr'
0 -24 -25
0 1325 8 1 1 0 1 221 2
300.
0.001 1.0
'heatr'
-21 -25 -26 /in1 in2 out
1325 2 0 0 0 2 / mat npk(0) nqa(0) ntem(0) loc(0) ipr(1)
443 444/  kermatot damage
'gaspr'
-21 -26 -27
'groupr'
-21 -27 0 28/in1 in2 gin out
1325 17 10 11 6 1 1 2/
'al-27 groupr n.ng data for fendl-2'/
300./
1.e10/
3/
3 251 'mubar'/
3 252 'xi'/
3 253 'gamma'/
3 259 '1/v'/
6/
16/
21 /
22 /
23 /
24 /
25 /
0/
0/
'moder'
 30 -31/ tape30 = fendlep.dat
'reconr'
-31 -32/
'pendf tape for Al from endf/b-6 gamma-int'/
1300 2/
.002 /err tempr(0.) ndigit(7) errmax
' Al from endf/b-6'/
' processed at IAEA/NDS with njoy94.105'/
0/
'gaminr'
-31 -32 0 33/
1300 10 3 6 1/
' 42 group Al photon interaction gaminr data endf6'/
'neutrons x's, g-prod. for al-27 from FENDL-2 '/
-1/
0/
'matxsr'
28 33 34/
1 'matxs al27'/
2 3 2 1/npart ntype nholl nmat
'neutrons x's, g-prod. for al-27 from FENDL-2 '/
'processed by IAEA/NDS with NJOY94.105 '/
'n' 'g'/hpart

175 42/ngrp(part)
'nscat' 'ng' 'gscat' 'ntherm'/htype
1 1 2 1/jinp
1 2 2 1/joutp
'al27' 1325 1300/hmat matno matgg
'stop'
```

NJOY input file for producing ACE data,provided by Dr. Andrej Trkov, Institute "Jozef Stefan", Ljubliana, Slovenia

```
0
6
'moder'
20 -21
'reconr'
-21 -22
'Pendf tape for 13-Al-27 from Bologna-97' /
1325 2 /
.002 0. 7 0.007 /
'13-Al-27 from Bologna-97 ' /
'Processed by the NJOY94 nucl. data processing system' /
0 /
'moder'   / Convert to 'coded' for verification purposes
-22 42
'broadr'
-22 -23
1325 1 0 0 0. / No restart, no bootstrap
.002 2.e6 0.01 / 0.2% thinning (1.0%max)
300.
0 /
'unresr'
-21 -23 -30
1325  1 1 1
300.
1.0e10
0 /
'thermr'
0 -30 -31
0 1325 8 1 1 0 1 221 2
300.
0.001 1.0
'heatr'
-21 -31 -32/
1325 2 0 0 0 2 /
443 444 /
'gaspr'
-21 -32 -24
'acer' / Generate the Ace formatted library
-21 -24  0  26  27 / use Expanded Photon Prod.
1 1 1 .00 0 /
'13-Al-27 from Bologna-97 '/
1325 300. /
0.01 1 /
-2 20000
/
'stop'


'groupr' / Generate old photon production matrices
-21 -24 0 -25
1325 17 2 9 0 1 1 1 /
'13-Al-27 from Bologna-97 '/
300.
1.0e10
16 /
0 /
0 /
'stop'
```

c) NJOY input files used for processing data from JENDL-FF provided by Dr. Fujio Maekawa, JAERI, Japan

Input file for producing multi-group data

```
0
6
'moder'
20 -21
'moder'
30 -31
'reconr'
-21 -22
'pendf tape for zr-0 from jendl fusion file' /
4000 3 /
.005 /
'zr-0 from jendl fusion file' /
'processed by the njoy nuclear data processing system' /
'see original jff tape for details of evaluation' /
'broadr'
-22 -23
4000 4 0 1 0 /
.005 /
300. 600. 900. 1500.
0 /
'unresr'
-21 -23 -25
4000 4 10  0 /
300. 600. 900. 1500.
1.0e10 1.0e4 1000. 300. 100. 30. 10. 1. 0.1 1.0e-3
0 /
'heatr'
-21 -25 -24 0 /
4000 2 /
443 444
'thermr'
0 -24 -23
0 4000 8 4 1 0 1 221 0
300. 600. 900. 1500.
.005 4.6
'groupr'
-21 -23 0 91
4000 17 10 11 6 4 10 0 /
'zr-0 from jendl fusion file' /
300. 600. 900. 1500.
1.0e10 1.0e4 1000. 300. 100. 30. 10. 1. 0.1 1.0e-3
3 /
3 221 'free thermal scattering'/
6 /
6 221 'free thermal scattering'/
16 /
0 /
3 1 'total'/
3 2 'elastic'/
3 102 'capture'/
3 221 'free thermal scattering'/
6 2 'elastic'/
6 221 'free thermal scattering'/
0 /
3 1 'total'/
3 2 'elastic'/
3 102 'capture'/
3 221 'free thermal scattering'/
6 2 'elastic'/
6 221 'free thermal scattering'/
0 /
3 1 'total'/
3 2 'elastic'/
3 102 'capture'/
3 221 'free thermal scattering'/
6 2 'elastic'/
6 221 'free thermal scattering'/
0 /
0 /
'reconr'
-31 -23
'pendf tape for zr from fendl-1:ep'/
4000 1 /
.005 /
'zr from fendl-1:ep' /
0 /
'gaminr'
-31 -23 0 -24
4000 10 4 8 0
'vitamin-j photon interaction library'/
-1 0 /
0 /
'matxsr'
91 -24 90 /
0 'saei&fns njoy'/
2 4 1 1
'jff vitamin-j multigroup library'/
'n' 'g'
175 42
'nscat' 'ng' 'gscat' 'ntherm'
1 1 2 1
1 2 2 1
'zr-0' 4000 4000
'stop'
```

Input file for producing data in ACE format

```
0
6
'moder'
20 -21
'reconr'
-21 -22
'pendf tape for be-9 from jendl fusion file' /
425 3 /
.005 /
'be-9 from jendl fusion file' /
'processed by the njoy nuclear data processing system' /
'see original jff tape for details of evaluation' /
0 /
'broadr'
-22 -23
425 1 0 0 0 /
.005 /
300.
0 /
'heatr'
-21 -23 -24 0 /
425 /
'thermr'
0 -24 -23
0 425 8 1 1 0 1 221 0
300.
.005 4.6
'groupr'
-21 -23 0 -25
425 3 2 3 0 1 1 1 /
'be-9 from jendl fusion file' /
300.
1.0e10
16 /
0 /
0 /
'acer'
-21 -23 -25 26 27 /
1 1 1 .40 /
'be-9 jendl fusion file (njoy94)'/
425 300. /
0.01 1 /
/
'stop'
```

d) NJOY input file used for processing data from BROND-2

Input file for producing multigroup data

```
0
6
'moder'
20 -21
'reconr'
-21 -22
'Pendf tape for Sn-nat from FENDL-2'/
5000 2/ mat ncards ngrids
.002 0. 7 .007/ err tempr(0.) ndigit(7) errmax
'Sn-nat from FENDL-2'/
'processed at IAEA/NDS with NJOY94.105'/
0/
'broadr'
-21 -22 -23/
5000 1/ mat ntemp istart(0) itrap(0) temp(0.)
.002 2.e6 .01/ err emaxbr errmax(20*err) errin(.0001*err)
300./ temp
0/
'unresr'
-21 -23 -24
5000 1 6/
300./
1.e10 1.e4 1.e3 100. 10. 1./
0/
'thermr'
0 -24 -25
0 5000 8 1 1 0 1 221/
300.
0.001 1.0
'heatr'
-21 -25 -26 /in1 in2 out
5000 2 0 0 0 2 / mat npk(0) nqa(0) ntem(0) loc(0) ipr(1)
443 444/  kermatot damage
'gaspr'
-21 -26 -27
'groupr'
-21 -27 0 28/in1 in2 gin out
5000 17 10 11 6 1 6 2/
'Sn-nat groupr n.ng data for fendl-2'/
300./
1.e10 1.e4 1000. 100. 10. 1./
3/
3 221/
3 251 'mubar'/
3 252 'xi'/
3 253 'gamma'/
3 259 '1/v'/
6/
6 221/
16/
0/
0/
'moder'
32 -33/ tape32 = fendlep.dat
'reconr'
-33 -34/
'pendf tape for Sn from ENDF/B-6 gamma-int'/
5000 2/
.002 /err tempr(0.) ndigit(7) errmax
' Sn from endf/b-6'/
' processed with njoy94.105'/
0/
'gaminr'
-33 -34 0 35/
5000 10 3 6 1/
' 42 group Sn photon interaction gaminr data endf6'/
-1/
0/
'matxsr'
28 35 36/
1 'matxs Snnat'/
2 3 2 1/npart ntype nholl nmat
'neutrons x's, g-prod. for Sn-nat from FENDL-2 '/
'processed with NJOY94.105 '/
'n' 'g'/hpart
175 42/ngrp(part)
'nscat' 'ng' 'gscat' 'ntherm'/htype
 1 1 2 1/jinp
 1 2 2 1/joutp
'snnat' 5000 5000/hmat matno matgg
'stop'
```

Input file for producing data in ACE format

```
0
6
'moder'
20 -21
'reconr'
-21 -22
'Pendf tape for Sn-nat from FENDL-2'/
5000 2/ mat ncards ngrids
.002 0. 7 .007/ err tempr(0.) ndigit(7) errmax
'Sn-nat from FENDL-2'/
'processed at IAEA/NDS with NJOY94.105'/
0/
'broadr'
-21 -22 -23/
5000 1/ mat ntemp istart(0) itrap(0) temp(0.)
.002 2.e6 .01/ err emaxbr errmax(20*err) errin(.0001*err)
300./ temp
0/
'unresr'
-21 -23 -24
5000 1 6/
300./
1.e10 1.e4 1.e3 100. 10. 1./
0/
'thermr'
0 -24 -25
0 5000 8 1 1 0 1 221/
300.
0.001 1.0
'heatr'
-21 -25 -26 /in1 in2 out
5000 2 0 0 0 2 / mat npk(0) nqa(0) ntem(0) loc(0) ipr(1)
443 444/  kermatot damage
'gaspr'
-21 -26 -27
'acer' / generate the ace formatted library
-21 -27 0 30 31
1 0 1 .40 0/
'Sn-nat from FENDL-2 with NJOY94.105'/
5000 300./
0.01 1/
/
/
'stop'
```

e) NJOY input file used for processing Au-197 from FENDL-2 at IAEA/NDS

Input file for producing multigroup data

```
0
6
*moder*
20 -21
*reconr*
-21 -22
*Pendf tape for Au-197 from FENDL-2*/
7925 2/ mat ncards ngrids
.002 0. 7 .007/ err tempr(0.) ndigit(7) errmax
*Au-197 from FENDL-2*/
*processed at IAEA/NDS with NJOY94.105*/
0/
*broadr*
-21 -22 -23/
7925 1/ mat ntemp istart(0) itrap(0) temp(0.)
.002 2.e6 .01/ err emaxbr errmax(20*err) errin(.0001*err)
300./ temp
0/
*unresr*
-21 -23 -24
7925 1 8/
300./
1.e10 1.e4 1000. 300. 100. 30. 10. 1./
0/
*thermr*
0 -24 -25
0 7925 8 1 1 0 1 221/
300.
0.001 1.0
*heatr*
-21 -25 -26 /in1 in2 out
7925 2 0 0 0 2 / mat npk(0) nqa(0) ntem(0) loc(0) ipr(1)
443 444/  kermatot damage
*gaspr*
-21 -26 -27
*groupr*
-21 -27 0 28/in1 in2 gin out
7925 17 10 11 6 1 8 2/ mat ng(175) np(42) iwt lo nte ns
ipr
*Au-197 groupr n.ng data for fendl-2*/
300./
1.e10 1.e4 1000. 300. 100. 30. 10. 1./
3/
3 221/
3 251 *mubar*/
3 252 *xi*/
3 253 *gamma*/
3 259 *1/v*/
6/
6 221/
16/
21 /
22 /
23 /
24 /
25 /
0/
0/
*moder*
 40 -41
*reconr*
-41 -42/
*pendf tape for Au from endf/b-6 gamma-int*/
7900 2/
.002 /err tempr(0.) ndigit(7) errmax
* Au from endf/b-6*/
* processed at NDS with njoy94.105*/
0/
*gaminr*
-41 -42 0 43/
7900 10 3 6 1/
* 42 group Au photon interaction gaminr data endf6*/
-1/
0/
*matxsr*
28 43 44/
1 *matxs au197*/
2 3 2 1/npart ntype nholl nmat
*neutrons x's, g-prod. for Au-197 from FENDL-2 */
*processed at IAEA/NDS with NJOY94.105 */
*n* *g*/hpart
175 42/ngrp(part)
*nscat* *ng* *gscat* *ntherm*/htype
1 1 2 1/jinp
1 2 2 1/joutp
*au197* 7925 7900/hmat matno matgg
*stop*
```

Input file for producing data in ACE format

```
0
6
*moder*
20 -21
*reconr*
-21 -22
*Pendf tape for Au-197 from FENDL-2*/
7925 2/ mat ncards ngrids
.002 0. 7 .007/ err tempr(0.) ndigit(7) errmax
*Au-197 from FENDL-2*/
*processed at IAEA/NDS with NJOY94.105*/
0/
*broadr*
-21 -22 -23/
7925 1/ mat ntemp istart(0) itrap(0) temp(0.)
.002 2.e6 .01/ err emaxbr errmax(20*err) errin(.0001*err)
300./ temp
0/
*unresr*
-21 -23 -24
7925 1 8/
300./
1.e10 1.e4 1000. 300. 100. 30. 10. 1./
0/
*thermr*
0 -24 -25
0 7925 8 1 1 0 1 221/
300.
0.001 1.0
*heatr*
-21 -25 -26 /in1 in2 out
7925 2 0 0 0 2 / mat npk(0) nqa(0) ntem(0) loc(0) ipr(1)
443 444/  kermatot damage
*gaspr*
-21 -26 -27
*acer* / generate the ace formatted library
-21 -27 0 30 31
1 0 1 .40 0/
*Au-197 from FENDL-2 processed at IAEA/NDS, witn
NJOY94.105*/
7925 300./
0.01 1/
/
/
*stop*
```

Appendix E: Distribution of the FENDL library

(As recommended at the IAEA Advisory Group Meeting on FENDL, held in Del Mar, California, 5-9 Dec.1995)

The master copy of the FENDL-2 library resides with the Nuclear Data Section of the International Atomic Energy Agency. To facilitate user access to the library the official copy of FENDL-2 was distributed in March 1997 to the major nuclear data centres in Europe (NEA Data Bank, Paris), Japan (JNDC, Tokai-mura), Russia (CJD,Obninsk) and USA (NNDC, Brookhaven and RSIC, Oak Ridge). As agreed between data centers, sharing common FENDL information, the recipients are receiving now the same products from all above centers. The data are available and may be further distributed to the user community according to the customer service options given below. Each FENDL-2 sublibrary will be in a single data set, i.e. Activation, Decay, etc. in the 8 mm tape, 6 mm tape, 4 mm tape or standard 9 track magnetic tape (6250 bpi or 1600 bpi) and CD-ROM options. The interested scientists may request FENDL-2 (or parts of it) directly from the IAEA/NDS or from one of these centers.

Table 1. FENDL CUSTOMER SERVICE OPTIONS

MEDIA	FORMAT	By WHOM
Electronic	FTP	IAEA, NEADB, NNDC
4 mm tape	UNIX TAR	CJD, IAEA, NEADB, NNDC, RSIC
	VAX BACKUP	CJD, IAEA, NEADB, NNDC
	ASCII	NEADB
6 mm tape	UNIX TAR	NEADB
	VAX BACKUP	NEADB
	ASCII	NEADB
8 mm tape	UNIX TAR	NEADB, NNDC, RSIC
	VAX BACKUP	NEADB, NNDC
	ASCII	NEADB
9 track	ASCII	CJD, IAEA
	EBCDIC	CJD, IAEA
CD-ROM	UNIX TAR	RSIC
	ASCII	NEADB

Table notes
1) NNDC will distribute FENDL unprocessed data
2) RSIC will distribute FENDL processed data
3) RSIC offers cost free service to ITER customers