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FENDL-3 Library – Summary documentation

R.A. Forrest, R. Capote, N. Otsuka,

T. Kawano, A.J. Koning, S. Kunieda, J-Ch. Sublet and Y. Watanabe

December 2012

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Introduction

History of the FENDL libraries

Nuclear fusion is recognised as a long-term energy source. The IAEA has played an important role in nurturing the work on this future energy source by providing support for the exchange of scientific and technical information on fusion research through conferences, meetings and projects. The most important initiative on fusion research is currently the ITER project, and in order to design this and ensure safe operation a wide range of Nuclear Data information is fundamental. Realisation that the needs of nuclear data for fusion are different from those of fission meant that it was appropriate to produce a specific data library to address these needs.

The Fusion Evaluation Nuclear Data Library (FENDL) was the response of the IAEA to the need for a data library specifically designed for fusion applications. An initial meeting was held in 1989 [1] and following the creation and testing of FENDL-1 in 1995 [2] work started on FENDL-2. This work culminated in the release of the library FENDL-2 [3] containing evaluations judged to be the best available in February 1997.

Following testing and discussion of the way forward [4] the next version (FENDL-2.1) was released in 2004 [5], and this was extensively used for ITER material studies (ITER Project Management and Quality Programme: Quality Assurance in Neutronic Analyses). It had long been recognised that the neutron fluxes achievable in ITER would not be sufficient to investigate and qualify materials for future fusion power plants, and it would be therefore necessary to construct a facility to test candidate fusion reactor materials under high neutron radiation dose conditions approximating those to be found in a fusion reactor. This facility – International Fusion Materials Irradiation Facility (IFMIF) – involves accelerating high currents (up to 250 mA) of deuterons to 40 MeV and impinging them on a liquid lithium target to produce neutrons. Deuterons that strike elements of the accelerator transport system, as well as various target materials, would induce radioactivity that needs to be considered in the safe operation of this facility as well as in its eventual decommissioning. The status of energy differential deuteron cross section data from a few MeV up to 40 MeV is considered by the IFMIF development community to be inadequate for the purposes of assessing the facility with respect to operational safety and licensing issues. In particular, the FENDL-2.1 library does not contain data for incident charged particles (e.g. protons and deuteron), while the maximum energy for incident neutrons is limited to 20 MeV.

Recognizing these difficulties in March 2006, the International Nuclear Data Committee (INDC) recommended the extension of the FENDL library to cover the nuclear data needs of the IFMIF community. A Technical Meeting aiming at identifying possible detailed objectives for a CRP was held at IAEA, Vienna, on 31 October – 2 November 2007 [6]. The CRP was approved by IAEA Research Program Advisory Committee on December 2007.

As is usual for a CRP there were three Research Coordination Meetings; [7,8,9] and these led to the production of the FENDL-3 library.

Components of FENDL-3

FENDL-3 consists of General Purpose (GP) and Activation parts, both of which contain neutron-, proton- and deuteron-induced files with a maximum energy of the incident particle at least equal to 60 MeV. The GP neutron library contains high-fidelity covariance data where these are available; to complement these data a shadow library based on TENDL-2011 is provided so that uncertainty calculations can be made with a complete library. The quality of TENDL-2011 uncertainties ranges from low- to medium-fidelity [10], but the library is complete and may suffice for many applications. In addition to the evaluated files a set of processed files are available for use in applications and for testing. Table 1 shows the materials and details of the GP neutron library.

Table 1. Details of all materials in the neutron-induced GP library.

Num	MAT	Material	Laboratory	Date	Source	E_{max}(eV)	Size
1	125	1-H-1	LANL	OCT05	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	58kb
2	128	1-H-2	LANL	FEB97	ENDF/B-VII.0	1.50E+08	85kb
3	131	1-H-3	LANL	NOV01	ENDF/B-VII.0+Ad-Hoc	6.00E+07	99kb
4	225	2-He-3	JAERI	JUN87	JENDL-4+Ad-Hoc	6.00E+07	41kb
5	228	2-He-4	LANL	SEP10	ENDF/B-VII.1+Ad-Hoc	6.00E+07	227kb
6	325	3-Li-6	LANL	APR06	ENDF/B-VII.0+TENDL-2010	2.00E+08	1432kb
7	328	3-Li-7	LANL	AUG88	ENDF/B-VII.0+TENDL-2010	2.00E+08	1066kb
8	425	4-Be-9	LLNL,LANL	OCT09	ENDF/B-VII.0+TENDL-2010	2.00E+08	1961kb
9	525	5-B-10	LANL	APR06	ENDF/B-VII.0+TENDL-2010	2.00E+08	1483kb
10	528	5-B-11	LANL	MAY89	ENDF/B-VII.0+TENDL-2010	2.00E+08	1519kb
11	625	6-C-12	KYUSHU	JUL03	JENDL-4.0+JENDL/HE-2007	1.50E+08	6082kb
12	628	6-C-13	NRG	NOV10	TENDL-2010	2.00E+08	1632kb
13	715	7-N-15	LANL	DEC05	RUSFOND+TENDL-2010	2.00E+08	1249kb
14	725	7-N-14	JNDC	JUN89	JENDL-4.0+JENDL/HE-2007	1.50E+08	8185kb
15	825	8-O-16	LANL	DEC05	ENDF/B-VII.1	1.50E+08	2986kb
16	828	8-O-17	NRG	NOV10	TENDL-2010	2.00E+08	2134kb
17	831	8-O-18	NRG	NOV10	TENDL-2010	2.00E+08	1933kb
18	925	9-F-19	CNDC,ORNL	OCT03	ENDF/B-VII.0+JENDL/HE-2007	1.50E+08	5963kb
19	1125	11-Na-23	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	6073kb
20	1225	12-Mg-24	KYUSHU	DEC03	JENDL-4.0+JENDL/HE-2007	1.50E+08	11481kb
21	1228	12-Mg-25	KYUSHU	DEC03	JENDL-4.0+JENDL/HE-2007	1.50E+08	11423kb
22	1231	12-Mg-26	KYUSHU	DEC03	JENDL-4.0+JENDL/HE-2007	1.50E+08	11291kb
23	1325	13-Al-27	LANL	FEB97	JEFF-3.1.1	1.50E+08	2195kb
24	1425	14-Si-28	LANL,ORNL	DEC02	ENDF/B-VII.0	1.50E+08	1870kb
25	1428	14-Si-29	LANL,ORNL	JUN97	ENDF/B-VII.0	1.50E+08	1846kb
26	1431	14-Si-30	LANL,ORNL	JUN97	ENDF/B-VII.0	1.50E+08	1570kb
27	1525	15-P-31	NRG	OCT10	TENDL-2010	2.00E+08	2443kb
28	1625	16-S-32	NRG	OCT10	TENDL-2010	2.00E+08	2514kb
29	1628	16-S-33	NRG	NOV10	TENDL-2010	2.00E+08	2685kb
30	1631	16-S-34	NRG	NOV10	TENDL-2010	2.00E+08	2322kb
31	1637	16-S-36	NRG	NOV10	TENDL-2010	2.00E+08	2093kb
32	1725	17-Cl-35	ORNL,LANL	OCT03	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	10829kb
33	1731	17-Cl-37	ORNL,LANL	OCT03	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	7423kb
34	1825	18-Ar-36	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5719kb
35	1831	18-Ar-38	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5703kb
36	1837	18-Ar-40	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5539kb
37	1925	19-K-39	NRG	FEB12	TENDL-2012	2.00E+08	2958kb
38	1928	19-K-40	NRG	FEB12	TENDL-2012	2.00E+08	3197kb
39	1931	19-K-41	NRG	FEB12	TENDL-2012	2.00E+08	2913kb
40	2025	20-Ca-40	SAEI	MAY03	JENDL-4.0+JENDL/HE-2007	1.50E+08	6164kb
41	2031	20-Ca-42	SAEI	MAY03	JENDL-4.0+JENDL/HE-2007	1.50E+08	6350kb
42	2034	20-Ca-43	SAEI	MAY03	JENDL-4.0+JENDL/HE-2007	1.50E+08	6413kb
43	2037	20-Ca-44	SAEI	MAY03	JENDL-4.0+JENDL/HE-2007	1.50E+08	6003kb
44	2043	20-Ca-46	SAEI	MAY03	JENDL-4.0+JENDL/HE-2007	1.50E+08	5689kb
45	2049	20-Ca-48	SAEI	MAY03	JENDL-4.0+JENDL/HE-2007	1.50E+08	5394kb
46	2125	21-Sc-45	NRG	OCT04	JEFF-3.1.1	2.00E+08	2204kb
47	2225	22-Ti-46	LANL	FEB09	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	5984kb
48	2228	22-Ti-47	LANL	FEB09	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	6160kb

Num	MAT	Material	Laboratory	Date	Source	E_{max}(eV)	Size
49	2231	22-Ti-48	LANL,ORNL	AUG10	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	5935kb
50	2234	22-Ti-49	LANL	FEB09	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	6002kb
51	2237	22-Ti-50	LANL	FEB09	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	5762kb
52	2325	23-V-50	JAEA	MAR10	JENDL-4.0+TENDL-2010	2.00E+08	3222kb
53	2300	23-V-51	SAEI	MAY03	JENDL-4.0+JENDL/HE-2007	1.50E+08	7285kb
54	2425	24-Cr-50	KIT	MAR10	KIT-2010	2.00E+08	18537kb
55	2431	24-Cr-52	FZK/INR	APR09	ENDF/B-VII.1+JENDL/HE-2007	1.50E+08	12766kb
56	2434	24-Cr-53	KIT	MAR10	KIT-2010	2.00E+08	16892kb
57	2437	24-Cr-54	KIT	MAR10	KIT-2010	2.00E+08	9710kb
58	2525	25-Mn-55	IAEA	FEB11	IAEA (as ENDF/B-VII.1)	6.00E+07	6282kb
59	2625	26-Fe-54	LANL,ORNL	SEP96	ENDF/B-VII.0	1.50E+08	1877kb
60	2631	26-Fe-56	NRG	FEB04	JEFF-3.1.1+TENDL-2010	1.50E+08	9068kb
61	2634	26-Fe-57	LANL,ORNL	SEP96	ENDF/B-VII.0	1.50E+08	2169kb
62	2637	26-Fe-58	NRG	OCT04	JEFF-3.1.1	2.00E+08	1816kb
63	2725	27-Co-59	ANL,ORNL	JUL89	ENDF/B-VII.0+JENDL/HE-2007	1.50E+08	5631kb
64	2825	28-Ni-58	LANL,ORNL	SEP97	ENDF/B-VII.0	1.50E+08	17187kb
65	2831	28-Ni-60	LANL,ORNL	SEP97	ENDF/B-VII.0	1.50E+08	16571kb
66	2834	28-Ni-61	LANL,ORNL	SEP97	ENDF/B-VII.0	1.50E+08	2090kb
67	2837	28-Ni-62	LANL,ORNL	SEP97	ENDF/B-VII.0	1.50E+08	1842kb
68	2843	28-Ni-64	LANL,ORNL	SEP97	ENDF/B-VII.0	1.50E+08	1848kb
69	2925	29-Cu-63	LANL,ORNL	FEB98	ENDF/B-VII.0	1.50E+08	2034kb
70	2931	29-Cu-65	LANL,ORNL	FEB98	ENDF/B-VII.0	1.50E+08	1954kb
71	3025	30-Zn-64	SIT.SHIMZ	AUG07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6800kb
72	3031	30-Zn-66	SIT.SHIMZ	AUG07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6082kb
73	3034	30-Zn-67	SIT.SHIMZ	AUG07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6326kb
74	3037	30-Zn-68	SIT.SHIMZ	AUG07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6235kb
75	3043	30-Zn-70	SIT.SHIMZ	AUG07	JENDL-4.0+JENDL/HE-2007	1.50E+08	5731kb
76	3125	31-Ga-69	SIT.SHIMZ	MAY07	JENDL-4.0+JENDL/HE-2007	1.50E+08	5466kb
77	3131	31-Ga-71	SIT.SHIMZ	MAY07	JENDL-4.0+JENDL/HE-2007	1.50E+08	5476kb
78	3225	32-Ge-70	NRG	DEC04	JEFF-3.1.1	2.00E+08	2335kb
79	3231	32-Ge-72	NRG	DEC04	JEFF-3.1.1	2.00E+08	2091kb
80	3234	32-Ge-73	NRG	DEC04	JEFF-3.1.1	2.00E+08	2266kb
81	3237	32-Ge-74	NRG	OCT04	JEFF-3.1.1	2.00E+08	1882kb
82	3243	32-Ge-76	NRG	OCT04	JEFF-3.1.1	2.00E+08	1702kb
83	3525	35-Br-79	JAEA	AUG09	JENDL-4.0+TENDL-2010	2.00E+08	2191kb
84	3531	35-Br-81	JAEA	AUG09	JENDL-4.0+TENDL-2010	2.00E+08	2127kb
85	3925	39-Y-89	BNL-LANL	AUG06	ENDF/B-VII.0+TENDL-2010	2.00E+08	5091kb
86	4025	40-Zr-90	JNDC	AUG89	JENDL-4.0+JENDL/HE-2007	1.50E+08	6572kb
87	4028	40-Zr-91	JAEA	JUL07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6591kb
88	4031	40-Zr-92	JAEA	JUL07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6608kb
89	4037	40-Zr-94	JAEA	JUL07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6645kb
90	4043	40-Zr-96	JAEA	JUL07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6615kb
91	4125	41-Nb-93	JAEA	JUL07	JENDL-4.0+JENDL/HE-2007	1.50E+08	6623kb
92	4225	42-Mo-92	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5417kb
93	4231	42-Mo-94	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5418kb
94	4234	42-Mo-95	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5582kb
95	4237	42-Mo-96	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5498kb
96	4240	42-Mo-97	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5589kb
97	4243	42-Mo-98	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5412kb
98	4249	42-Mo-100	SIT.SHIMZ	MAY06	JENDL-4.0+JENDL/HE-2007	1.50E+08	5470kb
99	4525	45-Rh-103	CAD,BRC,+	FEB05	JEFF-3.1.1+TENDL-2011	2.00E+08	5605kb
100	4725	47-Ag-107	JAERI,BNL	MAR05	ENDF/B-VII.0+TENDL-2010	2.00E+08	2260kb
101	4731	47-Ag-109	BNL,KAERI	FEB06	ENDF/B-VII.0+TENDL-2010	2.00E+08	3551kb
102	4825	48-Cd-106	IRMM,JNDC	OCT10	IRMM+TENDL-2010	2.00E+08	2032kb
103	4831	48-Cd-108	IRMM,UA,ANL	OCT10	IRMM+TENDL-2010	2.00E+08	1726kb
104	4837	48-Cd-110	IRMM,UA,ANL	OCT10	IRMM+TENDL-2010	2.00E+08	1724kb
105	4840	48-Cd-111	IRMM,JNDC	OCT10	IRMM+TENDL-2010	2.00E+08	1998kb
106	4843	48-Cd-112	IRMM,UA,ANL	OCT10	IRMM+TENDL-2010	2.00E+08	1661kb
107	4846	48-Cd-113	BNL,CNDL	MAR05	ENDF/B-VII.0+TENDL-2010	2.00E+08	2316kb
108	4849	48-Cd-114	IRMM,UA,ANL	OCT10	IRMM+TENDL-2010	2.00E+08	1646kb
109	4855	48-Cd-116	IRMM,UA,ANL	OCT10	IRMM+TENDL-2010	2.00E+08	1613kb
110	5025	50-Sn-112	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3341kb
111	5031	50-Sn-114	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3515kb
112	5034	50-Sn-115	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3759kb
113	5037	50-Sn-116	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3313kb
114	5040	50-Sn-117	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3212kb
115	5043	50-Sn-118	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3302kb

Num	MAT	Material	Laboratory	Date	Source	E_{max}(eV)	Size
116	5046	50-Sn-119	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3080kb
117	5049	50-Sn-120	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	2976kb
118	5055	50-Sn-122	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	2814kb
119	5061	50-Sn-124	JAEA	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	2653kb
120	5125	51-Sb-121	CNDC,BNL	DEC04	ENDF/B-VII.0+TENDL-2010	2.00E+08	2107kb
121	5131	51-Sb-123	CNDC,BNL	DEC04	ENDF/B-VII.0+TENDL-2010	2.00E+08	2024kb
122	5325	53-I-127	LANL,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	3215kb
123	5525	55-Cs-133	JAEA+	APR09	JENDL-4.0+TENDL-2010	2.00E+08	5220kb
124	5625	56-Ba-130	JNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1793kb
125	5631	56-Ba-132	JNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1792kb
126	5637	56-Ba-134	JNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1819kb
127	5640	56-Ba-135	JNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1824kb
128	5643	56-Ba-136	JNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1800kb
129	5646	56-Ba-137	JNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1779kb
130	5649	56-Ba-138	CNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1771kb
131	5725	57-La-138	NRG	NOV10	TENDL-2010	2.00E+08	3242kb
132	5728	57-La-139	NRG	OCT10	TENDL-2010	2.00E+08	3049kb
133	5825	58-Ce-136	BNL	MAR06	ENDF/B-VII.0+TENDL-2010	2.00E+08	3552kb
134	5831	58-Ce-138	BNL	MAR06	ENDF/B-VII.0+TENDL-2010	2.00E+08	3700kb
135	5837	58-Ce-140	JNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1813kb
136	5843	58-Ce-142	JNDC,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1766kb
137	6425	64-Gd-152	JAEA+	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	4016kb
138	6431	64-Gd-154	JAEA+	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3567kb
139	6434	64-Gd-155	JAEA+	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	4084kb
140	6437	64-Gd-156	JAEA+	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3588kb
141	6440	64-Gd-157	JAEA+	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3407kb
142	6443	64-Gd-158	JAEA+	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	3091kb
143	6449	64-Gd-160	JAEA+	DEC09	JENDL-4.0+TENDL-2010	2.00E+08	2748kb
144	6825	68-Er-162	TIT	SEP00	ENDF/B-VII.0+TENDL-2010	2.00E+08	2147kb
145	6831	68-Er-164	TIT	SEP00	ENDF/B-VII.0+TENDL-2010	2.00E+08	2094kb
146	6837	68-Er-166	TIT,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	2178kb
147	6840	68-Er-167	TIT,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	2247kb
148	6843	68-Er-168	TIT,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	2151kb
149	6849	68-Er-170	TIT,BNL	JAN05	ENDF/B-VII.0+TENDL-2010	2.00E+08	1977kb
150	7125	71-Lu-175	NRG	NOV10	TENDL-2010	2.00E+08	3282kb
151	7128	71-Lu-176	NRG	NOV10	TENDL-2010	2.00E+08	3185kb
152	7225	72-Hf-174	JAEA	JUL09	JENDL-4.0+TENDL-2010	2.00E+08	2198kb
153	7231	72-Hf-176	JAEA	JUL09	JENDL-4.0+TENDL-2010	2.00E+08	2193kb
154	7234	72-Hf-177	JAEA	JUL09	JENDL-4.0+TENDL-2010	2.00E+08	2079kb
155	7237	72-Hf-178	JAEA	AUG09	JENDL-4.0+TENDL-2010	2.00E+08	2216kb
156	7240	72-Hf-179	JAEA	JUL09	JENDL-4.0+TENDL-2010	2.00E+08	2167kb
157	7243	72-Hf-180	JAEA	JUL09	JENDL-4.0+TENDL-2010	2.00E+08	2193kb
158	7328	73-Ta-181	JAEA	JUL06	JENDL-4.0+JENDL/HE-2007	1.50E+08	6408kb
159	7425	74-W-180	IAEA	AUG06	IAEA	1.50E+08	9153kb
160	7431	74-W-182	IAEA	AUG06	IAEA	1.50E+08	9549kb
161	7434	74-W-183	IAEA	AUG06	IAEA	1.50E+08	10160kb
162	7437	74-W-184	IAEA	AUG06	IAEA	1.50E+08	8494kb
163	7443	74-W-186	IAEA	AUG06	IAEA	1.50E+08	9618kb
164	7525	75-Re-185	NRG	NOV10	TENDL-2010	2.00E+08	3175kb
165	7531	75-Re-187	NRG	NOV10	TENDL-2010	2.00E+08	3089kb
166	7825	78-Pt-190	NRG	NOV10	TENDL-2010	2.00E+08	3306kb
167	7831	78-Pt-192	NRG	NOV10	TENDL-2010	2.00E+08	3155kb
168	7837	78-Pt-194	NRG	NOV10	TENDL-2010	2.00E+08	3006kb
169	7840	78-Pt-195	NRG	OCT10	TENDL-2010	2.00E+08	3077kb
170	7843	78-Pt-196	NRG	NOV10	TENDL-2010	2.00E+08	2867kb
171	7849	78-Pt-198	NRG	NOV10	TENDL-2010	2.00E+08	2719kb
172	7925	79-Au-197	LANL	JAN84	ENDF/B-VII.0+JENDL/HE-2007	1.50E+08	7170kb
173	8225	82-Pb-204	NRG	DEC04	JEFF-3.1.1	2.00E+08	2405kb
174	8231	82-Pb-206	NRG	DEC04	JEFF-3.1.1	2.00E+08	2430kb
175	8234	82-Pb-207	NRG	DEC04	JEFF-3.1.1	2.00E+08	2212kb
176	8237	82-Pb-208	NRG	DEC04	JEFF-3.1.1	2.00E+08	2009kb
177	8325	83-Bi-209	NRG	DEC04	JEFF-3.1.1	2.00E+08	2632kb
178	9040	90-Th-232	IAEA	FEB06	IAEA	6.00E+07	13592kb
179	9228	92-U-235	ORNL,LANL,+	SEP06	ENDF/B-VII+JENDL/HE-2007	1.50E+08	69737kb
180	9237	92-U-238	ORNL,LANL+	SEP06	ENDF/B-VII+JENDL/HE-2007	1.50E+08	48649kb

The materials for the proton-induced GP library are identical to the neutron-induced except that ^{4}He is missing. The materials for the deuteron-induced GP library are identical to the neutron-induced except that ^{1}H, ^{2}H, ^{3}H, ^{3}He and ^{4}He are missing. As can be seen in Table 1 the neutron-induced library is assembled from several sources and also in cases where the original data source does not extend to sufficiently high energy it was necessary to merge data with a high energy source. Details of the main source evaluations used and the merging process are given in the following sections.

Isotopes from ENDF/B-VII.0 and ENDF/B-VII.1

ENDF/B-VII.1 [11] was released in December 2011, enabling FENDL-3 to benefit from the most recent evaluation works under the ENDF nuclear data library development. There are 65 neutron evaluations imported from ENDF/B-VII.1 in FENDL-3. An extensive review of ENDF/B-VII.1 evaluation is given in reference 11, and detailed information for the individual nuclei may be found in the references, or in the comment section in the evaluated files. Note that some of the files in FENDL-3 include “merged” evaluations, such as ENDF/B-VII below 20 MeV and TENDL above 20 MeV. Table 1 shows clearly the individual sources for such merged files. In addition, some of the data files in ENDF/B-VII.1 come from different sources. For example, the barium evaluations are based on JENDL-3.3, but the resonance parameters were updated by BNL. A document of provenance of each evaluation is available at the BNL web site [12]. This section briefly summarizes these 65 evaluations in FENDL-3.

Light elements

Eleven files were imported from ENDF/B-VII.1; they are ^{1,2,3}H, ^{4}He, ^{6,7}Li, ^{9}Be, ^{10,11}B, ^{16}O, and ^{19}F. For production of the light element nuclear data files, the use of R-matrix analysis is essential. The R-matrix analysis often includes not only the neutron induced reaction (incoming channel), but also inverse charged particle induced reactions that form the same A+1 system. Although the R-matrix analysis can be very precise if high quality experimental data are provided, the analysis will be difficult in the higher energy region due to the number of total open channels, increasing number of partial waves, lower energy resolution in the experimental data, etc. The light element data files often consist of both the R-matrix results at low energies and other evaluation methods in the higher energy region.

There are three files that contain the Standards cross sections evaluated at IAEA: ^{1}H(n,n), ^{6}Li(n,t), and ^{10}B(n,α) reactions, which are based on the R-matrix analysis by Hale of LANL. In ENDF/B-VII.1, a new R-matrix evaluation for ^{4}He was performed by Hale, and this recent evaluation was also used for FENDL-3. The triton and ^{4}He data are limited to 20 MeV in the original ENDF files, and they were extended to 60 MeV by A. Trkov for FENDL-3 [9].

Structural materials

In the mass range from ^{27}Al to ^{65}Cu, the evaluated files have been produced by ORNL and LANL, except for a relatively old evaluation of ^{59}Co by ANL [13]. These evaluations consist of the resonance parameters evaluated at ORNL, the cross sections in the MeV region evaluated with GNASH at LANL or TNG at ORNL, and the high energy data above 20 MeV at LANL as a part of the LA-150 library. These evaluations include the most recent Reich-Moore resonance parameters evaluated at ORNL for ^{35,37}Cl, Ti isotopes, and ^{58,60}Ni which include covariance data. The evaluations above the resonance region are basically the statistical Hauser-Feshbach model calculations with pre-equilibrium emission. The nuclear reaction modelling adopted includes secondary particle angle and energy distributions, as they are primarily important for radiation shielding calculations for fusion applications.

The ^{59}Co evaluation should be particularly noted. The statistical model calculations were performed with the CADE code [14], for which few documents are available nowadays. The particle energy and angular distributions are stored separately in MF=4 and 5, not in MF=6 that are commonly used for fusion applications. This evaluation is planned to be upgraded in the next release of ENDF/B library.

Above the resonance regions, many of the data files have not been updated since ENDF/B-VI, except for the (n,α) cross section of Cr, Fe, Co, and Ni isotopes. There was a long-standing issue concerning the cluster emission in the pre-equilibrium process, where phenomenological alpha-particle knock-out models do not reproduce the recent measurements of alpha-particle production at LANSCE/LANL. Since gas-production in the structural material causes serious embrittlement problems in the fusion reactors, a better modelling of the cluster emission is crucial. This problem was resolved by applying an improved Iwamoto-Harada model [15], and the data files for these isotopes were upgraded by including new calculations carried out by Kunieda. The updated model also impacts on the calculated alpha-particle energy spectra.

Resolved resonances are given up to relatively higher energies (several hundred keV), and strong fluctuations still persist up to several MeV. Since these structures in the total and elastic scattering cross sections are very important for neutron shielding calculations, these resonance-like shapes should be retained in the evaluations. The evaluated total cross sections are often obtained by tracing high-resolution measurements to take these fluctuations into account; evaluations were typically performed with the SOK code [16].

Fission products region

The evaluated files adopted from ENDF/B-VII.1 in the fission product regions cover the Y, Ag, Cd, Sb, Ba, Ce, and Er isotopes. These evaluations often combine recommendations by the WPEC fission product data evaluation, and new resonance parameter evaluations made at BNL. There are also some new evaluations made at BNL with the EMPIRE code in collaboration with KAERI, which include ^{109}Ag in 2010 and ^{136,138}Ce in 2006. The Cd data files also include new resonance parameters evaluated at IRMM and BNL.

Although ^{197}Au is outside the fission product region, this contains the important Standards capture cross section. This file includes the updated resonance parameters evaluated at BNL.

Actinides

The two evaluations for the major actinides, ^{235}U and ^{238}U, were adopted from ENDF/B-VII.1, which are almost identical to those in ENDF/B-VII.0, except for a minor change to the delayed neutron yield and spectra. These two files include the Standards fission cross sections. The details of uranium evaluations are given by Young et al. [17], and these files have been extensively tested in the past [18, 19], as they are the key nuclides for fission energy applications.

Although the ENDF community has devoted a lot of effort to an upgrade of the major actinide evaluations, there might be some rooms to improve these files in the future release of ENDF. Issues regarding the prompt fission neutron spectra are actively discussed under a current IAEA CRP [20], and new Monte Carlo approaches have become a common technique to calculate the spectra not only for the prompt neutrons but also for the prompt gamma-rays. Another rather striking observation was a large discrepancy in the inelastic scattering cross section among ENDF, JENDL, and JEFF in the fast energy range [21], despite all the libraries performing very well in criticality benchmark testing. These problems should be addressed both by new experiments and theoretical developments in the near future. These nuclides are of limited relevant to fusion studies, since the use of actinides in the fusion reactor is basically only for diagnostic purposes.

Isotopes from JEFF libraries

In the FENDL-3 general purpose neutron file, 15 isotopic evaluations are taken from the Joint Evaluated Fission and Fusion file JEFF-3.1.1, they are: ^{27}Al, ^{45}Sc, ^{56,58}Fe, ^{70,72,73,74,76}Ge, ^{103}Rh, ^{204,206,207,208}Pb and ^{209}Bi.

The JEFF project involves evaluation efforts that cover the main nuclear data needs in the fields of fission and fusion applications. This initiative has provided the means for co-operative activities between participating countries while ensuring the most rational and efficient use of available resources. JEFF is an OECD/NEA Data Bank project and essentially involves only scientists from European countries. Development of the JEFF libraries is indirectly financed by the voluntary contributions of participating individuals and organizations. Staff at the NEA Data Bank ensure the maintenance of the JEFF library, and twice yearly JEFF meetings bring together experts in all areas of nuclear data.

The latest large scale release of the JEFF library took place in May 2005, with JEFF-3.1 [22]. Since then, a new general purpose library JEFF-3.1.1 was released [23], as well as an upgrade to the activation data library and an update to the fission yield and decay data libraries.

In total, JEFF-3.1.1 now consists of the following sub-libraries:

- neutron general purpose library (381 isotopes)
- neutron activation library (774 isotopes)
- thermal scattering law library (9 materials)
- decay data library (3852 isotopes)
- fission yield data library (19 isotopes)
- proton special purpose library (26 isotopes)

In addition, new evaluation efforts are underway to contribute to JEFF-3.2.

The isotopic evaluations adopted from JEFF for FENDL-3.1 all extend up to 150 or 200 MeV. One, ^{27}Al, is originally from the ENDF/B-VI.8 library. The others are NRG evaluations, made with TALYS.

Isotopes from TENDL libraries

In the FENDL-3 general purpose neutron file, 23 isotopic evaluations stem from the TALYS Evaluated Nuclear Data library, TENDL, version 2010 [24], they are: ^{13}C, ^{17,18}O, ^{31}P, ^{32,33,34,36}S, ^{39,40,41}K, ^{138,139}La, ^{175,176}Lu, ^{185,187}Re and ^{190,192,194,195,196,198}Pt.

The basic idea behind TENDL is that, in principle, complete nuclear model calculations can be performed for all nuclides, reaction channels and energies, important or not. In fact, the basis for TENDL is formed by complete nuclear data libraries, all with essentially the same structure, for all nuclides. The first version of the library, TENDL-2008, was actually just that: the isotopic nuclear data libraries were made by global TALYS calculations only, without any further adjustment of any model parameters or inclusion of experimental data. In more recent versions of TENDL, progressively more effort is invested in nuclide specific parameters and experimental data, while for the most important channels, e.g. $^{238}U(n,\gamma)$, nuclear model calculations are completely normalized by experimental data, existing evaluated data, or differential data inferred from integral measurements, in certain energy ranges. Note that even for such nuclides; TALYS is still used for all other energy ranges and reaction channels. In sum, the contents of TENDL ranges from detailed channel-by-channel evaluation work to libraries based on default parameters.

The TENDL-2010 library contains sub-libraries for incident neutrons, protons, deuterons, tritons, helium-3's, alphas, photons and a fission yield sub-library. For all types of incident particles, nuclear data libraries for 2430 isotopic materials are produced. These are all isotopes, in either ground or metastable state, with a half-life longer than 1 sec. from Z = 3 (Li) to Z = 110 (Ds). This is about a factor of 8 more nuclides than in any other world library. All libraries extend up to 200 MeV.

The isotopic evaluations adopted from TENDL-2010 for FENDL-3 concern mostly nuclides for which no other evaluated file exists, or for which the existing evaluations are very old.

Isotopes from JENDL-4.0 and JENDL/HE-2007

JENDL-4.0 [25] is the latest version of the Japanese Evaluated Nuclear Data Library which was released in 2010. FENDL-3 adopted data for 65 isotopes from JENDL-4.0, they are: ^{3}He, ^{12}C, ^{14}N, ^{23}Na, $^{24,25,26}Mg$, $^{36,38,40}Ar$, $^{40,42,43,44,46,48}Ca$, $^{50,51}V$, $^{64,66,67,68,70}Zn$, $^{69,71}Ga$, $^{79,81}Br$, $^{90,91,92,94,96}Zr$, ^{93}Nb, $^{92,94,95,96,97,98,100}Mo$, $^{112,114,115,116,117,118,119,120,122,124}Sn$, ^{133}Cs, $^{152,154,155,156,157,158,160}Gd$, $^{174,176,177,178,179,180}Hf$ and ^{181}Ta.

JENDL-4.0 is a general purpose library that contains neutron cross sections for 406 nuclei up to 20 MeV. The library includes new/updated resolved resonance parameters. It also features smooth cross sections (including double differential neutron spectra and gamma-ray production data) that were evaluated with the coupled-channel optical model code OPTMAN [26] and modern nuclear reaction model codes CCONE [27] and POD [28]. The benchmark calculations were performed mainly for fission reactors, and reasonable performances were reported [29].

The JENDL high-energy file is one of the special purpose files for e.g., accelerator applications. The latest version is JENDL/HE-2007 [30] that was released in 2007. It includes the data of neutron- and proton-induced reactions for 106 nuclei up to 3 GeV. For the incident energy range lower than 150 or 200 MeV, the cross sections were evaluated with the OPTMAN / ECIS and the GNASH code [31]. It should be noted that neutron data below 20 MeV are identical with those in JENDL-3.3. The library gives double differential cross sections for emitted neutrons, protons, deuterons, tritons, helium-3s, alpha-particles and gamma-rays. It also includes isotope production cross sections and pion production cross sections. Neutron shielding benchmarks were carried out, e.g., for TIARA experiment at 43 and 68 MeV [32,33], and reasonable performance was obtained.

Data Merging: High-energy extension of neutron library

The sections above describe the most important data sources for the neutron-induced data library. For each material one of those evaluated data files was assigned based on the cross sections and the benchmark results. However, ENDF/B-VII.1 and JEFF-3.1.1 do not have high-energy (E_n > 20 or 30 MeV) data for most nuclei. Also, high-energy data were not included in JENDL-4.0 and RUSFOND.

It was considered essential that each material have high energy data that extend at least to 60 MeV and so for a nuclide where it was absent a choice of data from either JENDL/HE-2007 or TENDL-2010 (MT=5) was made. This was done keeping the original data untouched, except for a few cases:

- All the data above 150 MeV, and pion production cross sections were removed for JENDL/HE-2007.
- For ^{235,238}U, it was agreed to use ENDF/B-VII.0 for low energy and JENDL/HE-2007 for high energy. However, there is a difference in data formatting for fission neutron angle-energy distribution. In FENDL-3, (MF,MT)=(6,18) was adopted both for low and high energy regions.

All the data files prepared were checked by CHECKR-8.05 and FIZCON-8.03, and no serious problems were reported. Note that the codes sometimes detected errors/warnings, but the same errors/warnings were seen in the original files.

Activation data

In the fusion research programme, safety and environmental issues are of great importance in the continuing development of power plants. In support of this programme, a sound, complete and reliable neutron-induced activation cross-section data library is required. The European Activation File (EAF) project has been an on-going process performed through European and Worldwide co-operation that has led to the creation of succeeding EAF versions. The latest release EAF-2010 [34] also includes cross section data for deuteron- and proton-induced reactions. As with EAF-2005 and -2007 all cross sections have an upper energy limit of 60 MeV. EAF-2010 has benefited from the generation, maintenance, verification and validation of comprehensive activation files. Cross section validation exercises against both experimental data, systematics and integral data, which were started in 1995, nearly 20 years ago, enabling a comprehensive and multi-faceted assessment of the data. Verification and validation reports are available on line on the EASY web site [35].

The EAF-2010 library in its ENDF-6 formatted version has been selected to be the FENDL/A-3.0 neutron activation library. It contains 66,256 excitation functions involving 816 different materials from ^{1}H to ^{257}Fm, atomic numbers 1 to 100, in the energy range 10^{-5} eV to 60 MeV. When produced isomeric states are treated up to the second level and termed: ground (g), first (m) and second (n). Uniquely, uncertainties are also provided that quantify the degree of confidence placed on the data for each of the numerous open reaction channels. The original EAF-2010 library in EAF format and its associated uncertainty file has been translated into ENDF-6 format, taking advantage of the new high-energy format extension approved during the CSWEG meeting in November 2010. New versions of the Utility codes CHECKR and FIZCON have been used with success to validate the format of the newly constructed library. The cross sections represent materials that are infinitely dilute, no self-shielding is included and the temperature for Doppler broadening is 294 K.

The FENDL/A-3.0 neutron activation library encompasses many data forms to satisfy broad user requirements:

- Original ENDF-6 formatted files
- PENDF's NJOY style, point-wise files (294 K)
- GENDF's 211 groups files

- ACE dosimetry files, for the LANL Monte Carlo code MCNP
- PENDF's LLNL Generalized Nuclear Data format files (xml)

Two other activation libraries have also been assembled for proton and deuteron incident particles; FENDL/A-3.0 proton and FENDL/A-3.0 deuteron. The most recent TENDL-2011 [36] data files have been selected for the 803 materials needed. The nuclear data energy range has been extended up to 200 MeV. Concerning the deuteron induced library files for all but the first 16 lightest nuclei (Z < 10), the (d,p) channels have been changed by renormalisation based on phenomenological systematics of A. Ignatyuk [37,38]. This was done at the PENDF file level so this modification is not in the original ENDF-6 formatted files.

The FENDL/A-3.0 proton and FENDL/A-3.0 deuteron activation libraries encompass several data forms to satisfy broad user requirements:

- Original ENDF-6 formatted files (original only)
- PENDF's PREPRO style point-wise files (original and renormalized)
- GENDF's 162 groups files (original and renormalized)

For the neutron-induced Activation library data are provided for the 816 materials shown in Table 2.

Table 2. Details of all materials in neutron-induced Activation library.

H-1	V-48	Se-82	Ru-98	Te-118	Ce-137	Tb-156	Hf-180	Au-195	Pa-234
H-2	V-49	Br-76	Ru-99	Te-119	Ce-137m	Tb-156m	Hf-180m	Au-196	U-230
H-3	V-50	Br-77	Ru-100	Te-119m	Ce-138	Tb-156n	Hf-181	Au-196n	U-231
He-3	V-51	Br-79	Ru-101	Te-120	Ce-139	Tb-157	Hf-182	Au-197	U-232
He-4	Cr-48	Br-81	Ru-102	Te-121	Ce-140	Tb-158	Ta-175	Au-198	U-233
Li-6	Cr-50	Br-82	Ru-103	Te-121m	Ce-141	Tb-159	Ta-176	Au-198m	U-234
Li-7	Cr-51	Kr-76	Ru-104	Te-122	Ce-142	Tb-160	Ta-177	Au-199	U-235
Be-7	Cr-52	Kr-78	Ru-105	Te-123	Ce-143	Tb-161	Ta-179	Au-200m	U-236
Be-9	Cr-53	Kr-79	Ru-106	Te-123m	Ce-144	Dy-153	Ta-180	Hg-193	U-237
Be-10	Cr-54	Kr-80	Rh-99	Te-124	Pr-141	Dy-154	Ta-180m	Hg-193m	U-238
B-10	Mn-52	Kr-81	Rh-99m	Te-125	Pr-142	Dy-155	Ta-181	Hg-194	U-240
B-11	Mn-53	Kr-82	Rh-100	Te-125m	Pr-143	Dy-156	Ta-182	Hg-195	U-241
C-12	Mn-54	Kr-83	Rh-101	Te-126	Nd-140	Dy-157	Ta-183	Hg-195m	Np-234
C-13	Mn-55	Kr-84	Rh-101m	Te-127	Nd-141	Dy-158	Ta-184	Hg-196	Np-235
C-14	Fe-52	Kr-85	Rh-102	Te-127m	Nd-142	Dy-159	W-178	Hg-197	Np-236
N-14	Fe-54	Kr-86	Rh-102m	Te-128	Nd-143	Dy-160	W-180	Hg-197m	Np-236m
N-15	Fe-55	Rb-82m	Rh-103	Te-129	Nd-144	Dy-161	W-181	Hg-198	Np-237
O-16	Fe-56	Rb-83	Rh-105	Te-129m	Nd-145	Dy-162	W-182	Hg-199	Np-238
O-17	Fe-57	Rb-84	Pd-100	Te-130	Nd-146	Dy-163	W-183	Hg-200	Np-239
O-18	Fe-58	Rb-85	Pd-101	Te-131m	Nd-147	Dy-164	W-184	Hg-201	Pu-234
F-19	Fe-59	Rb-86	Pd-102	Te-132	Nd-148	Dy-165	W-185	Hg-202	Pu-236
Ne-20	Fe-60	Rb-87	Pd-103	I-123	Nd-149	Dy-166	W-186	Hg-203	Pu-237
Ne-21	Co-55	Sr-82	Pd-104	I-124	Nd-150	Ho-163	W-187	Hg-204	Pu-238
Ne-22	Co-56	Sr-83	Pd-105	I-125	Pm-143	Ho-164	W-188	Tl-199	Pu-239
Na-22	Co-57	Sr-84	Pd-106	I-126	Pm-144	Ho-164m	Re-181	Tl-200	Pu-240
Na-23	Co-58	Sr-85	Pd-107	I-127	Pm-145	Ho-165	Re-182	Tl-201	Pu-241
Na-24	Co-58m	Sr-86	Pd-108	I-128	Pm-146	Ho-166	Re-182m	Tl-202	Pu-242
Mg-24	Co-59	Sr-87	Pd-109	I-129	Pm-147	Ho-166m	Re-183	Tl-203	Pu-243
Mg-25	Co-60	Sr-88	Pd-110	I-130	Pm-148	Er-160	Re-184	Tl-204	Pu-244
Mg-26	Ni-56	Sr-89	Pd-112	I-131	Pm-148m	Er-161	Re-184m	Tl-205	Pu-245
Mg-28	Ni-57	Sr-90	Ag-105	I-133	Pm-149	Er-162	Re-185	Pb-200	Pu-246
Al-26	Ni-58	Sr-91	Ag-106m	I-135	Pm-150	Er-164	Re-186	Pb-201	Pu-247
Al-27	Ni-59	Y-86	Ag-107	Xe-122	Pm-151	Er-165	Re-186m	Pb-202	Am-239

Si-28	Ni-60	Y-87	Ag-108m	Xe-124	Sm-144	Er-166	Re-187	Pb-203	Am-240
Si-29	Ni-61	Y-87m	Ag-109	Xe-125	Sm-145	Er-167	Re-188	Pb-204	Am-241
Si-30	Ni-62	Y-88	Ag-110m	Xe-126	Sm-146	Er-168	Re-189	Pb-205	Am-242
Si-31	Ni-63	Y-89	Ag-111	Xe-127	Sm-147	Er-169	Os-182	Pb-206	Am-242m
Si-32	Ni-64	Y-90	Cd-106	Xe-128	Sm-148	Er-170	Os-183	Pb-207	Am-243
P-31	Ni-66	Y-91	Cd-107	Xe-129	Sm-149	Er-171	Os-183m	Pb-208	Am-244
P-32	Cu-63	Y-93	Cd-108	Xe-129m	Sm-150	Er-172	Os-184	Pb-209	Cm-240
P-33	Cu-64	Zr-86	Cd-109	Xe-130	Sm-151	Tm-165	Os-185	Pb-210	Cm-241
S-32	Cu-65	Zr-88	Cd-110	Xe-131	Sm-152	Tm-166	Os-186	Pb-212	Cm-242
S-33	Cu-67	Zr-89	Cd-111	Xe-131m	Sm-153	Tm-167	Os-187	Bi-203	Cm-243
S-34	Zn-62	Zr-90	Cd-112	Xe-132	Sm-154	Tm-168	Os-188	Bi-204	Cm-244
S-35	Zn-64	Zr-91	Cd-113	Xe-133	Sm-156	Tm-169	Os-189	Bi-205	Cm-245
S-36	Zn-65	Zr-92	Cd-113m	Xe-133m	Eu-145	Tm-170	Os-190	Bi-206	Cm-246
Cl-35	Zn-66	Zr-93	Cd-114	Xe-134	Eu-146	Tm-171	Os-191	Bi-207	Cm-247
Cl-36	Zn-67	Zr-94	Cd-115	Xe-135	Eu-147	Tm-172	Os-191m	Bi-208	Cm-248
Cl-37	Zn-68	Zr-95	Cd-115m	Xe-136	Eu-148	Tm-173	Os-192	Bi-209	Cm-249
Ar-36	Zn-69m	Zr-96	Cd-116	Cs-127	Eu-149	Yb-166	Os-193	Bi-210	Cm-250
Ar-37	Zn-70	Zr-97	In-111	Cs-129	Eu-150	Yb-168	Os-194	Bi-210m	Bk-245
Ar-38	Zn-72	Nb-90	In-113	Cs-131	Eu-150m	Yb-169	Ir-185	Po-206	Bk-246
Ar-39	Ga-66	Nb-91	In-114m	Cs-132	Eu-151	Yb-170	Ir-186	Po-207	Bk-247
Ar-40	Ga-67	Nb-91m	In-115	Cs-133	Eu-152	Yb-171	Ir-187	Po-208	Bk-248
Ar-41	Ga-69	Nb-92	Sn-112	Cs-134	Eu-152m	Yb-172	Ir-188	Po-209	Bk-248m
Ar-42	Ga-71	Nb-92m	Sn-113	Cs-135	Eu-153	Yb-173	Ir-189	Po-210	Bk-249
K-39	Ga-72	Nb-93	Sn-114	Cs-136	Eu-154	Yb-174	Ir-190	At-210	Bk-250
K-40	Ge-68	Nb-93m	Sn-115	Cs-137	Eu-155	Yb-175	Ir-191	At-211	Cf-246
K-41	Ge-69	Nb-94	Sn-116	Ba-128	Eu-156	Yb-176	Ir-192	Rn-211	Cf-248
K-42	Ge-70	Nb-95	Sn-117	Ba-129	Eu-157	Lu-169	Ir-192n	Rn-222	Cf-249
K-43	Ge-71	Nb-95m	Sn-117m	Ba-130	Gd-146	Lu-170	Ir-193	Ra-223	Cf-250
Ca-40	Ge-72	Nb-96	Sn-118	Ba-131	Gd-147	Lu-171	Ir-193m	Ra-224	Cf-251
Ca-41	Ge-73	Mo-92	Sn-119	Ba-132	Gd-148	Lu-172	Ir-194	Ra-225	Cf-252
Ca-42	Ge-74	Mo-93	Sn-119m	Ba-133	Gd-149	Lu-173	Ir-194n-	Ra-226	Cf-253
Ca-43	Ge-76	Mo-93m	Sn-120	Ba-133m	Gd-150	Lu-174	Ir-196m	Ra-228	Cf-254
Ca-44	Ge-77	Mo-94	Sn-121	Ba-134	Gd-151	Lu-174m	Pt188	Ac-225	Es-250
Ca-45	As--71	Mo-95	Sn-121m	Ba-135	Gd-152	Lu-175	Pt189	Ac-226	Es-251
Ca-46	As--72	Mo-96	Sn-122	Ba-135m	Gd-153	Lu-176	Pt190	Ac-227	Es-252
Ca-47	As--73	Mo-97	Sn-123	Ba-136	Gd-154	Lu-177	Pt191	Ac-228	Es-253
Ca-48	As--74	Mo-98	Sn-124	Ba-137	Gd-155	Lu-177m	Pt192	Th-227	Es-254
Sc-44m	As--75	Mo-99	Sn-125	Ba-138	Gd-156	Hf-170	Pt193	Th-228	Es-254m
Sc-45	As--76	Mo-100	Sn-126	Ba-139	Gd-157	Hf-171	Pt193m	Th-229	Es-255
Sc-46	As--77	Tc-95	Sb-119	Ba-140	Gd-158	Hf-172	Pt194	Th-230	Es-256m
Sc-47	Se-72	Tc-95m	Sb-120m	La-135	Gd-159	Hf-173	Pt195	Th-231	Es-257
Sc-48	Se-73	Tc-96	Sb-121	La-137	Gd-160	Hf-174	Pt195m	Th-232	Fm-252
Ti-44	Se-74	Tc-97	Sb-122	La-138	Tb-151	Hf-175	Pt196	Th-234	Fm-253
Ti-45	Se-75	Tc-97m	Sb-123	La-139	Tb-152	Hf-176	Pt197	Pa-228	Fm-255
Ti-46	Se-76	Tc-98	Sb-124	La-140	Tb-153	Hf-177	Pt198	Pa-229	Fm-257
Ti-47	Se-77	Tc-99	Sb-125	La-141	Tb-154	Hf-178	Pt200	Pa-230	
Ti-48	Se-78	Tc-99m	Sb-126	Ce-134	Tb-154m	Hf-178n	Pt202	Pa-231	
Ti-49	Se-79	Ru-96	Sb-127	Ce-135	Tb-154n-	Hf-179	Au-193	Pa-232	
Ti-50	Se-80	Ru-97	Sb-128	Ce-136	Tb-155	Hf-179n	Au-194	Pa-233	

Processing

In order to produce nuclear application libraries suitable for advanced fusion systems, the FENDL/MC-3.0 library - point-wise continuous-energy cross section data in ACE format for Monte Carlo calculations - and the FENDL/MG-3.0 library – multi-group cross section data up to 55 MeV for deterministic transport codes have been produced by the IAEA under a Special Service Agreement to D. López Aldama. The FENDL/MC-3.0 and FENDL/MG-3.0 libraries will be freely available from the Nuclear Data Section webpage (http://www-nds.iaea.org). The package ACEDOP [39], which allows Doppler broadening of ACE-formatted cross sections files in the resolved resonance range, is also available as for FENDL-2.1.

The work carried out to produce FENDL-3.0 application libraries is described in detail in Reference 44. The continuous-energy cross section data files in ACE format and the multi-group cross section data in MATXS format were produced by NJOY-99.364+ [40] at IAEA-NDS. Point-wise cross section data files in ENDF-6 format were generated using the PREPRO-2010 code package [41]. Two libraries of processed files were produced: FENDL/MC-3.0 and FENDL/MG-3.0.

FENDL/MC-3.0 is an ACE-formatted library suitable for use by the MCNP family of Monte-Carlo codes [42]. FENDL/MG-3.0 is a multi-group-formatted library, intended for use in deterministic transport codes like DORT and TORT [43].

A qualitative verification of processed files was also performed and the main findings and recommendations were given in Reference 44. Note that this reference describes the processing that was done on a preliminary version of FENDL-3, following the verification errors were corrected and the processing was repeated to give the version used for distribution.

Processing of the proton-induced GP library was also carried out by D. López Aldama, and involved the production the FENDL-P/E-3.0 and the FENDL-P/MC-3.0 libraries. The latter is a continuous-energy cross section library in ACE format for Monte Carlo calculations at energies lower than 60 MeV. The ^{6}Li, ^{7}Li and ^{10}B evaluations were extended up to 200 MeV using the data of TENDL-2010 library. For these isotopes TENDL-2010 data were adopted at high energy for MT=2 and MT=5 and the rest of reactions that were included in low energy file from FENDL-P/E-3.0 were kept up to the lower energy limit of TENDL-2010 library. At low energies, cross section data for MT=2 and MF=6 were converted from nuclear amplitude expansion (LTP=1) to nuclear plus interference distribution with the proton interference linear in μ (LTP=12). For ^{3}H and ^{3}He a cross section value of zero was set from the energy limit in the low energy file (usually 20 MeV) up to 60 MeV.

The processing sequence for generating the FENDL-P/MC-3.0 is shown in Figure 1.

A summary of the main processing options is presented below for completeness:

- Reconstruction tolerance in RECONR: 0.1%.
- Resonance-integral-check tolerance in RECONR: 0.2%.
- Maximum resonance integral error in RECONR: 5.0E-08 (default for 0.1%).
- Temperature: 300 K = 2.5852E-08 MeV.
- Temperature: 300 K = 2.5852E-08 MeV.
- Thinning tolerance in BROADR: 0.1%.
- Integral criterion tolerance in BROADR: 0.2%.
- Integral thinning tolerance in BROADR: 5.0E-0.8 (default for 0.1%).
- Maximum energy in BROADR: 20 MeV.
- No thermal data.
- No thinning in ACER.
- ACE-type 1 file.
- Suffix for ZAID in ACER: .30
- New cumulative angle distributions in ACER.

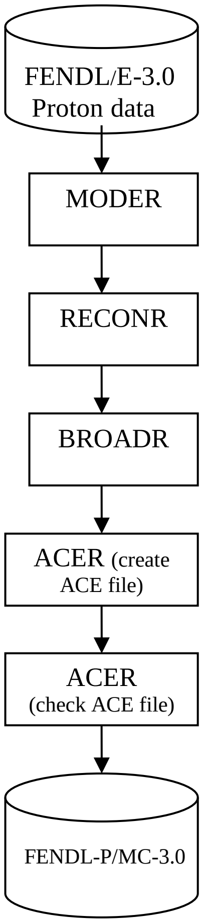

Figure 1: NJOY processing sequence for FENDL-P/MC-3.0

Since the entire deuteron-induced GP library was taken from TENDL-2011, it was also appropriate to use the processed library (ACE) produced by NRG [36].

The activation files were processed by J.Ch Sublet. In the case of neutron induced files, these are available as ACE, GENDF and PENDF files and also in the XML format termed GND [45]. In the case of proton-induced and deuteron-induced files these are available as GENDF and PENDF files.

Summary of files

A summary of the available FENDL-3 files is given in Table 3.

Table 3. Details of FENDL-3 files.

	General purpose (FENDL-3.0)					**Activation (FENDL/A-3.0)**			
	neutron	photo-atomic	neutron shadow	proton	deuteron	neutron	proton	deuteron	deuteron renormalized
Number of Materials	180	59	174	179	175	816	803	803	803
ENDF (FENDL/E-3.0)									
PENDF									
GND									
ACE (FENDL/MC-3.0)									
GENDF (FENDL/MG-3.0)									
MATXS (FENDL/MG-3.0)									

Verification

It is expected that further verification and validation will be carried out once the FENDL-3 data are freely available. In addition to the verification carried out as part of the processing additional work was presented at the 3rd RCM meeting. Details are available in the presentations [37]. A study of validation of the FENDL-3 neutron-induced activation file using integral data has been published [46], providing additional work on validation.

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