IEC-83

Present Status of JENDL Project

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1. JENDL-3 revision 2 (JENDL-3.2)

The revision work of JENDL-3 has been almost completed. By adding new evaluation for 16 nuclei, the evaluated data are given for 340 nuclei in JENDL-3.2. The γ-ray production data are stored for 66 nuclei. About 180 nuclei out of 340 have a new or revised data. Table 1 lists the nuclei and MAT numbers in JENDL-3.2. Table 2 summarizes main modifications for JENDL-3.2.

For the natural elements, many parts of data were constructed from the data of their isotopes. In the case where the resonance energy region are different from each other, the resonance regions are different even in the natural element data. In many cases, the data of natural element data are not consistent with those of isotopes. Since the natural elements have many experimental data than isotopes, the data constructed from isotopes were modified on the basis of those experimental data. Therefore, the users are recommended to use the data of natural elements instead of those of isotopes.

The ENDF-6 format is adopted for JENDL-3.2. So their MAT numbers are changed to the definition in the ENDF-6 format. In JENDL-3.1, data of some fission product nuclei are stored twice with different MAT numbers. It caused a confusion to the users. In JENDL-3.2, double entries of data exist no longer. The double-differential data (DDX) are important for fusion neutronics. However, JENDL-3.2 in the ENDF-6 format adopts the conventional MF4-MF5 representation of emitted neutron data instead of the MF6 format. We are preparing a special purpose file, JENDL Fusion File, which provides the DDX data in the MF6 format for fusion neutronics.

For the convenience of users, the ENDF-5 format version will be available on request, and pointwise data files will be provided, which are calculated at 0 K with RECENT or RESENDD.

So far the middle of April 1994, re-evaluation of data for ^{19}F, ^{150}Sm, ^{152}Sm and ^{154}Sm had not been finished. As to the γ-ray production data, Ca and Pb were under re-evaluation. JENDL-3.2 will be released May or June 1994.

Preliminary benchmark calculations have been made for important data. It has been confirmed that the revised data of Fe are in very good agreement with measured integral data. Data for main actinoids give better C/E values for fast and thermal reactor characteristics.

2. Future Subjects for JENDL

JENDL-3.2 gives much better evaluated data than JENDL-3.1. However, the following problems are still existing:

Covariance matrix

JENDL-3.2 has no covariance matrices. We recognize importance of the covariance matrices. A new working group has been organized in Japanese Nuclear Data Committee for study of evaluation method of the covariance matrices.

Energy Distributions

Energy distribution data in JENDL-3.2 should be reexamined. In particular, new representation method proposed in the ENDF-6 format has not been adopted in JENDL-3.2. So the energy balance is not always correct.

3. JENDL Special Purpose Files

The following evaluated data files other than JENDL-3.2 are being developed in Japan. Their status is given below. No progress has been made for the (α,n) data file and the decay data file.

JENDL Fusion File

JENDL Fusion File is made to provide precise double-differential neutron and charged particle emission data by using MF6 representation of the ENDF-6 format. The evaluation has been finished for the data of ^{27}Al, Si, Ca, Ti, Cr, ^{55}Mn, Fe, ^{59}Co, Ni, Cu, ^{75}As, Zr, ^{93}Nb, Mo, Sb, W, Pb and ^{209}Bi, and is under way for ^{19}F, Ge, Sn. SINCROS-II which consists of GNASH, DWUCK, CASTHY and several auxiliary programs is used for the theoretical calculation. Those results are examined by comparing with DDX measured at Tohoku and Osaka university. The data for JENDL Fusion File have been adopted in JENDL-3.2 after approximately changing the MF6 representation to MF4-MF5 representation.

JENDL Actinoid File

This file will provide 89 nuclei mainly in the range above Z=90. The data for 57 nuclei will be taken from JENDL-3.2. For the other nuclei, we need new evaluation. In the last year, the evaluation of ^{237}Pu and ^{244}Pu was made. At present, Dr. Konshin is investigating the parameters for fission cross section calculation for other nuclei. By adopting his results, new calculation will be made for other nuclides, and if needed the data taken from JENDL-3.2 for some minor actinoids will be updated.

JENDL Dosimetry File

The first version of JENDL Dosimetry File was released in 1991 with the data for 61 reactions. In the last year, re-evaluation of several cross sections was made by members of Dosimetry Integral Test Working Group of JNDC. New dosimetry reactions to be added to the 61 reactions are also investigated. In the first version of JENDL Dosimetry File, the covariance matrices were taken from other files such as IRDF-85. The evaluation of own covariance matrices is also the subjects to this working group.

JENDL Activation Cross Section File

The preliminary file has been compiled and its validation test started with integral data measured at JAERI/FNS. The test shows that JENDL Activation Cross Section File gives the most preferable results among libraries tested. The test and revision of the preliminary file is in progress.

JENDL High Energy Files

The evaluation of data for high energy neutrons and charged particles have been initiated in JNDC. They will make data files for neutrons and protons up to 50 MeV and about 1.5 GeV. The former files will be used for the ESNIT project promoted in JAERI. The evaluation of neutron data up to 50 MeV has been made for several structural materials. However, the compilation of data files has not yet completed. The latter files will be used for design of transmutation systems of high-level waste. The evaluation of data for Al, Pb and Bi was made for proton induced reactions up to 1 GeV. The proton-induced reaction data of Cr, Fe, Ni and Cu isotopes were also evaluated up to 15 MeV.

JENDL PKA/KERMA File

This file will store the spectra of primary knock-on atoms (PKA) and KERMA factors. The data to be stored are created from the data files up to 50 MeV made for the ESNIT project. A couple of codes to create the file have been developed and tested. However, the data compilation work has not yet started.

JENDL Photonuclear Data File

The evaluation has been almost finished for C, N, O, Al, Cu, Bi and U in the γ-ray energy range up to 140 MeV. Their compilation in the ENDF-6 format is in progress. The evaluation for Ti, Fe, Ta, W and Pb is also in progress.

A bibliographic index to the photonuclear data was published as JAERI-M 93-195.

Table 1 Nuclides and MAT numbers stored in JENDL-3.2

H - 1	125	H - 2	128	He- 3	225	He- 4	228
Li- 6	325	Li- 7	328	Be- 9	425	B - 10	525
B - 11	528	C - 12	625	N - 14	725	N - 15	728
O - 16	825	F - 19	925	Na- 23	1125	Mg- 0	1200
Mg- 24	1225	Mg- 25	1228	Mg- 26	1231	Al- 27	1325
Si- 0	1400	Si- 28	1425	Si- 29	1428	Si- 30	1431
P - 31	1525	S - 0	1600	S - 32	1625	S - 33	1628
S - 34	1631	S - 36	1637	Cl- 0	1700	Cl- 35	1725
Cl- 37	1731	Ar- 40	1837	K - 0	1900	K - 39	1925
K - 40	1928	K - 41	1931	Ca- 0	2000	Ca- 40	2025
Ca- 42	2031	Ca- 43	2034	Ca- 44	2037	Ca- 46	2043
Ca- 48	2049	Sc- 45	2125	Ti- 0	2200	Ti- 46	2225
Ti- 47	2228	Ti- 48	2231	Ti- 49	2234	Ti- 50	2237
V - 51	2328	Cr- 0	2400	Cr- 50	2425	Cr- 52	2431
Cr- 53	2434	Cr- 54	2437	Mn- 55	2525	Fe- 0	2600
Fe- 54	2625	Fe- 56	2631	Fe- 57	2634	Fe- 58	2637
Co- 59	2725	Ni- 0	2800	Ni- 58	2825	Ni- 60	2831
Ni- 61	2834	Ni- 62	2837	Ni- 64	2843	Cu- 0	2900
Cu- 63	2925	Cu- 65	2931	Ga- 0	3100	Ga- 69	3125
Ga- 71	3131	Ge- 0	3200	Ge- 70	3225	Ge- 72	3231
Ge- 73	3234	Ge- 74	3237	Ge- 76	3243	As- 75	3325
Se- 74	3425	Se- 76	3431	Se- 77	3434	Se- 78	3437
Se- 79	3440	Se- 80	3443	Se- 82	3449	Br- 79	3525
Br- 81	3531	Kr- 78	3625	Kr- 80	3631	Kr- 82	3637
Kr- 83	3640	Kr- 84	3643	Kr- 85	3646	Kr- 86	3649
Rb- 85	3725	Rb- 87	3731	Sr- 86	3831	Sr- 87	3834
Sr- 88	3837	Sr- 89	3840	Sr- 90	3843	Y - 89	3925
Y - 91	3931	Zr- 0	4000	Zr- 90	4025	Zr- 91	4028
Zr- 92	4031	Zr- 93	4034	Zr- 94	4037	Zr- 95	4040
Zr- 96	4043	Nb- 93	4125	Nb- 94	4128	Nb- 95	4131
Mo- 0	4200	Mo- 92	4225	Mo- 94	4231	Mo- 95	4234
Mo- 96	4237	Mo- 97	4240	Mo- 98	4243	Mo- 99	4246
Mo-100	4249	Tc- 99	4331	Ru- 96	4425	Ru- 98	4431
Ru- 99	4434	Ru-100	4437	Ru-101	4440	Ru-102	4443
Ru-103	4446	Ru-104	4449	Ru-106	4455	Rh-103	4525
Rh-105	4531	Pd-102	4625	Pd-104	4631	Pd-105	4634
Pd-106	4637	Pd-107	4640	Pd-108	4643	Pd-110	4649
Ag- 0	4700	Ag-107	4725	Ag-109	4731	Ag-110m	4735
Cd- 0	4800	Cd-106	4825	Cd-108	4831	Cd-110	4837
Cd-111	4840	Cd-112	4843	Cd-113	4846	Cd-114	4849
Cd-116	4855	In-113	4925	In-115	4931	Sn-112	5025
Sn-114	5031	Sn-115	5034	Sn-116	5037	Sn-117	5040
Sn-118	5043	Sn-119	5046	Sn-120	5049	Sn-122	5055
Sn-123	5058	Sn-124	5061	Sn-126	5067	Sb- 0	5100
Sb-121	5125	Sb-123	5131	Sb-124	5134	Sb-125	5137

Te-120	5225	Te-122	5231	Te-123	5234	Te-124	5237
Te-125	5240	Te-126	5243	Te-127m	5247	Te-128	5249
Te-129m	5253	Te-130	5255	I -127	5325	I -129	5331
I -131	5337	Xe-124	5425	Xe-126	5431	Xe-128	5437
Xe-129	5440	Xe-130	5443	Xe-131	5446	Xe-132	5449
Xe-133	5452	Xe-134	5455	Xe-135	5458	Xe-136	5461
Cs-133	5525	Cs-134	5528	Cs-135	5531	Cs-136	5534
Cs-137	5537	Ba-130	5625	Ba-132	5631	Ba-134	5637
Ba-135	5640	Ba-136	5643	Ba-137	5646	Ba-138	5649
Ba-140	5655	La-138	5725	La-139	5728	Ce-140	5837
Ce-141	5840	Ce-142	5843	Ce-144	5849	Pr-141	5925
Pr-143	5931	Nd-142	6025	Nd-143	6028	Nd-144	6031
Nd-145	6034	Nd-146	6037	Nd-147	6040	Nd-148	6043
Nd-150	6049	Pm-147	6149	Pm-148	6152	Pm-148m	6153
Pm-149	6155	Sm-144	6225	Sm-147	6234	Sm-148	6237
Sm-149	6240	Sm-150	6243	Sm-151	6246	Sm-152	6249
Sm-153	6252	Sm-154	6255	Eu- 0	6300	Eu-151	6325
Eu-152	6328	Eu-153	6331	Eu-154	6334	Eu-155	6337
Eu-156	6340	Gd-152	6425	Gd-154	6431	Gd-155	6434
Gd-156	6437	Gd-157	6440	Gd-158	6443	Gd-160	6449
Tb-159	6525	Hf- 0	7200	Hf-174	7225	Hf-176	7231
Hf-177	7234	Hf-178	7237	Hf-179	7240	Hf-180	7243
Ta-181	7328	W - 0	7400	W -182	7431	W -183	7434
W -184	7437	W -186	7443	Pb- 0	8200	Pb-204	8225
Pb-206	8231	Pb-207	8234	Pb-208	8237	Bi-209	8325
Ra-223	8825	Ra-224	8828	Ra-225	8831	Ra-226	8834
Ac-225	8925	Ac-226	8928	Ac-227	8931	Th-227	9025
Th-228	9028	Th-229	9031	Th-230	9034	Th-232	9040
Th-233	9043	Th-234	9046	Pa-231	9131	Pa-232	9134
Pa-233	9137	U -232	9219	U -233	9222	U -234	9225
U -235	9228	U -236	9231	U -237	9234	U -238	9237
Np-236	9343	Np-237	9346	Np-238	9349	Np-239	9352
Pu-236	9428	Pu-238	9434	Pu-239	9437	Pu-240	9440
Pu-241	9443	Pu-242	9446	Am-241	9543	Am-242	9546
Am-242m	9547	Am-243	9549	Am-244	9552	Am-244m	9553
Cm-241	9628	Cm-242	9631	Cm-243	9634	Cm-244	9637
Cm-245	9640	Cm-246	9643	Cm-247	9646	Cm-248	9649
Cm-249	9652	Cm-250	9655	Bk-249	9752	Bk-250	9755
Cf-249	9852	Cf-250	9855	Cf-251	9858	Cf-252	9861
Cf-254	9867	Es-254	9914	Es-255	9915	Fm-255	9936

Table 2 Main modifications for JENDL-3.2

C-12	capture cross section in 100 eV to 5 MeV, total above 8 MeV, some inelastic scattering cross sections
N-14	total cross section in 5 to 10 MeV, inelastic scattering cross sections
O	capture and inelastic scattering cross sections
Na-23	total cross section above 1 MeV
Al-27	inelastic scattering above 5.6 MeV, angular and energy distributions
Si	angular and energy distributions, small modification to inelastic scattering cross sections
P-31	total cross section above 544 keV
S	All cross sections above resonance region
K	total cross sections above 200 keV
Ca, Ti, V-51	almost all cross sections, angular and energy distributions, γ production data
Cr	total cross section in 300 keV to 4 MeV, angular and energy distributions of neutrons
Mn-55	part of inelastic scattering, angular and energy distributions
Fe	total cross section in 350 keV to 12 MeV, inelastic scattering cross section, angular and energy distributions, γ production data
Co-59	capture, inelastic scattering cross sections, angular and energy distributions, γ production data
Ni	total cross section in 557 keV to 5 MeV, angular and energy distributions, γ production data
Cu	cross sections in 50 to 153 keV, angular and energy distributions, γ production data
Zr	Almost all data
Nb-93	resonance parameters, inelastic scattering, angular and energy distributions
Mo	inelastic scattering, angular and energy distributions
Cd	resonance parameters, inelastic and capture cross sections
Sb	almost all cross sections, angular and energy distributions
Other Fission Products	resonance parameters, inelastic scattering and capture cross sections
W	almost all cross sections, angular and energy distributions
Pb	resonance parameters, almost all cross sections, angular and energy distributions
U-233	resonance parameters, fission, inelastic scattering, (n,2n) reaction cross sections, fission spectrum
U-235	resolved and unresolved resonance parameters, fission spectrum, ν_p
U-238	resolved and unresolved resonance parameters, inelastic scattering cross section
Pu-239	resolved resonance parameters, fission spectrum
Pu-241	resolved resonance parameters