IEL-20

JAPANESE EVALUATED NUCLEAR DATA LIBRARY, VERSION-3
— JENDL-3 —

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Abstract: The third version of Japanese Evaluated Nuclear Data Library (JENDL-3) has been developed aiming at really general applications such as fission, fusion and shielding calculations. The general purpose file of JENDL-3 contains neutron nuclear data for 324 nuclides in the ENDF-5 format. In the JENDL-3 evaluation, much effort has been devoted to improve reliability of high-energy data for fusion application, which were not satisfactory in JENDL-2, and to include gamma-ray production data. Some advanced nuclear theoretical models were adopted and recent experimental data of energy-angle double-differential cross sections (DDX) mainly measured in Japan were taken into account. Various benchmark tests have so far been made in order to verify the applicability of JENDL-3 to various fields. For fast reactor calculations, JENDL-3 gives satisfactory results for most of characteristics. Particularly, space dependences of reaction rates, sodium void coefficients and control rod worths, which were significant with JENDL-2, nearly disappear. This suggests that the JENDL-3 data are well balanced. Satisfactory applicability has been also proved for the other field such as thermal reactor calculations, fusion neutronics, shielding and dosimetry.

(JENDL-3, general purpose file, fission reactors, fusion neutronics, simulations evaluation, direct and pre-equilibrium processes, benchmark tests.)

Introduction

The Japanese Evaluated Nuclear Data Library (JENDL) has been developed by JAERI Nuclear Data Center in cooperation with Japanese Nuclear Data Committee since 1970's. Its first version was mainly aimed at fast reactor applications and was completed in 1977. The second version, which was completed in 1982, was applicable to all the fission reactor calculations, but was proved to be unsatisfactory for fusion neutronics. The third version (JENDL-3) has been compiled aiming at really general applications such as fission, fusion and shielding calculations. The general purpose file of JENDL-3 contains neutron nuclear data for 324 nuclides in the ENDF-5 format and was completed in 1989[1]. The specifications for JENDL-1, -2 and -3 are given in Table 1.

Table 1 Specification for each version of JENDL

	JENDL-1	JENDL-2	JENDL-3
Purpose	Fast Reactor	Fission Reactor	General
Completion	1977	1982	1989
Maximum Energy	15 MeV	20 MeV	20 MeV
Number of Nuclides*	72	181	324(53)
Light (Z=1~9)	4	6	14(10)
Medium light (Z=10~30)	23	33	56(23)
Fission Product (Z=31~69)	34	101	178(8)
Medium heavy (Z=70~87)	1	12	19(9)
Heavy (Z=88~94)	9	19	31(3)
Transplutonium (Z=95~100)	1	8	26(0)

( ) Number of nuclides with γ-ray production data

In the JENDL-3 evaluation, much effort has been devoted to improve reliability of high-energy data for fusion application, which were not satisfactory in JENDL-2, and to include gamma-ray production data. Some advanced nuclear theoretical models such as the coupled-channel model, DWBA, the pre-equilibrium model etc. have played an important role to achieve these purposes. Recent experimental data of energy-angle double-differential cross sections (DDX) mainly measured in Japan[2,3] were taken into account in the evaluation of emitted neutron spectra. The detailed evaluation will be given for each mass region.

The evaluation of JENDL-3 was completed in 1987, and a test version of JENDL-3, called JENDL-3T, was distributed for data validation. Since then various benchmark tests were made in order to verify the applicability of the data. The results of the benchmark tests were informed to the JENDL-3 compilation group in JNDC. The compilation group examined the results as a whole and informed the evaluators, if some modifications were to be made. The evaluators re-examined their evaluation by taking account of the comments from the compilation group. The final version of JENDL-3 was released in 1989, adopting the revised data.

Evaluation

Light Nuclides

The evaluation of light nuclides were performed mainly on the basis of experimental data, if they are available. A particular care was paid for DDX. The R-matrix theory was applied to the resonance structure.

Lithium: The tritium production cross sections of Li isotopes are important as a candidate material of the fusion blanket. The (n,t) cross section of ^{6}Li was evaluated with the R-matrix theory below 1 MeV and on the basis of the experimental data above 1 MeV. The (n,n't) cross section of ^{7}Li was evaluated on the basis of the newly measured data and is shown in Fig. 1. Energy distributions of both the isotopes were calculated with the phase-space model, and were given by using about 30 pseudo levels in the file.

Beryllium: The (n,2n) reaction cross section of ^{9}Be was evaluated on the basis of available experimental data. The 14 MeV value was based on the recent measurements in Japan, and is by 4% lower than that of JENDL-2 as shown in Fig. 2. The agreement with the experimental data is satisfactory.

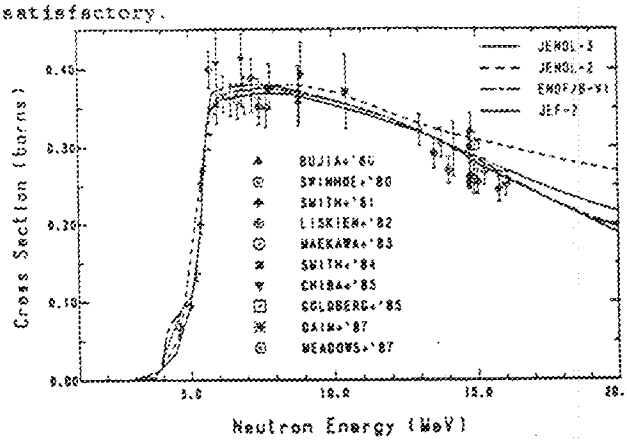

Fig. 1 ^{7}Li(n,n'α) T reaction cross section

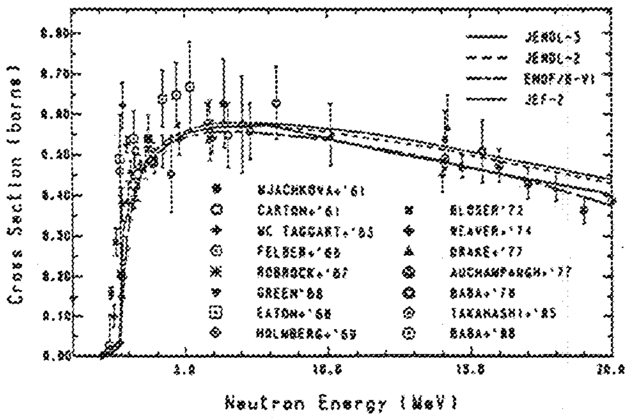

Fig. 2 ^{9}Be(n,2n) reaction cross section

Medium Nuclides

Among nuclides in the region of Z=10~67, fission products nuclides (Z=33~65) have been evaluated independently and systematically by a working group of JNDC, and the results are presented elsewhere in this proceedings. Hence the following are for the other nuclides.

Theoretical calculation: The spherical optical model and statistical model played an important role in the evaluation. In addition to these conventional models, more advanced models such as the coupled-channel optical model, DWBA and the pre-equilibrium model were applied particularly in the high energy region. Systematic trends of various parameters such as optical potentials and level densities were investigated. Figure 3 compares the DDXs of natural iron of JENDL-3 with the experimental data of Osaka and Tohoku Universities[2,3]. The JENDL-3 data reproduce the measured data well, while the JENDL-2 data, which were evaluated only with the conventional statistical model, give poor agreement.

Natural elements: In JENDL-3, the data of natural elements were evaluated on the basis of the experimental data of the natural elements independently of the evaluation of isotopic data, because there exist much more experimental data for the natural elements. If there exist some discrepancies between the natural element data thus evaluated and those composed of the isotopic data, we left them. Hence we recommend to use the natural element data for the natural elements. On the other hand, ENDF/B-VI and JEF-2 give only the isotopic data.

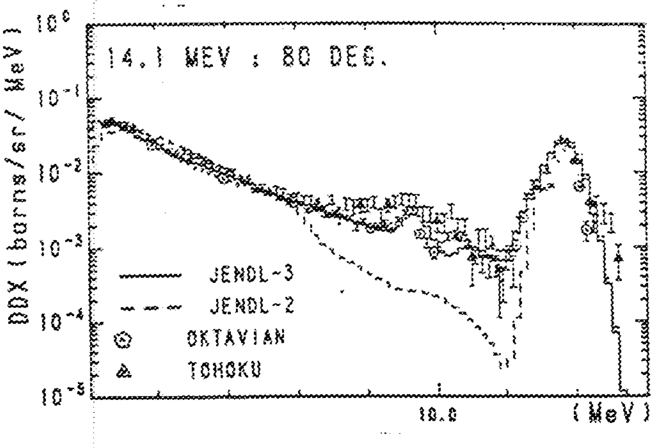

Fig. 3 DDX of Fe at 14 MeV

Threshold reaction cross sections are important for fusion and dosimetory applications. In most cases, they were calculated with the statistical model including the pre-equilibrium effects and were normalized to the experimental data.

However discrepancies still remain in some of the threshold reactions. Figure 4 compares the recently evaluated data of ^{56}Fe(n,α) reaction cross section. They agree at 14 MeV where the experimental data exist, but disagree in the other energy regions. To resolve these discrepancies, new measurements of the (n,α) cross sections are under planning at JAERI Tandem accelerator for main structural materials.

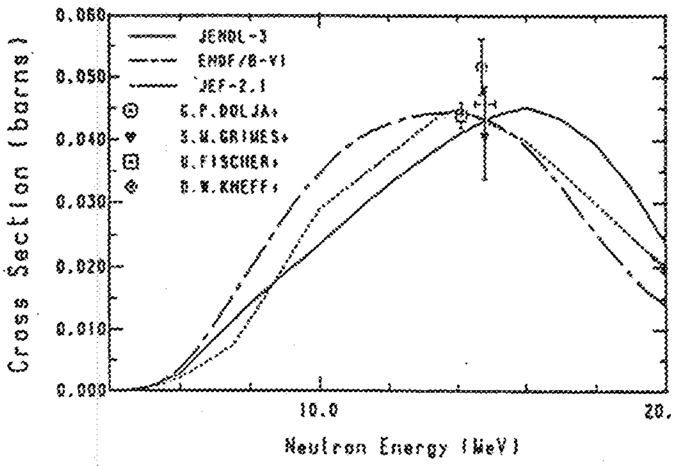

Fig. 4 ^{56}Fe(n,α) reaction cross section

Heavy Nuclides

Fifty-seven nuclides from ^{223}Ra and ^{255}Fm are contained as heavy nuclides. Among them a special care was taken to main fissile and fertile materials.

Simultaneous evaluation was applied[4] above 50 keV for the fission cross sections of ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu and the capture cross sections of ^{238}U and ^{197}Au. All the absolute measurements and the ratio measurements such as $\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$ were fitted by the generalized least squares method by a B-spline function. Covariance data required for this method were estimated from the experimental conditions. The evaluated results of ^{235}U and ^{239}Pu are shown in Fig. 5.

Capture cross section of ^{238}U has been carefully investigated. The values obtained with the simultaneous evaluation were found to be higher than the latest measurements of Kazakov et al.[5] in the energy range from 50 keV to 300 keV.

The results of the benchmark tests favor the smaller cross section, and Fröhner's theoretical calculation[6] also supports the lower values. Hence we adopted the lower values on the basis of the data of Kazakov et al. Thus the values of JENDL-3 are by 10% smaller than those of JENDL-2 in this energy region but agree with those of ENDF/B-VI and JEF-2 as seen in Fig. 6.

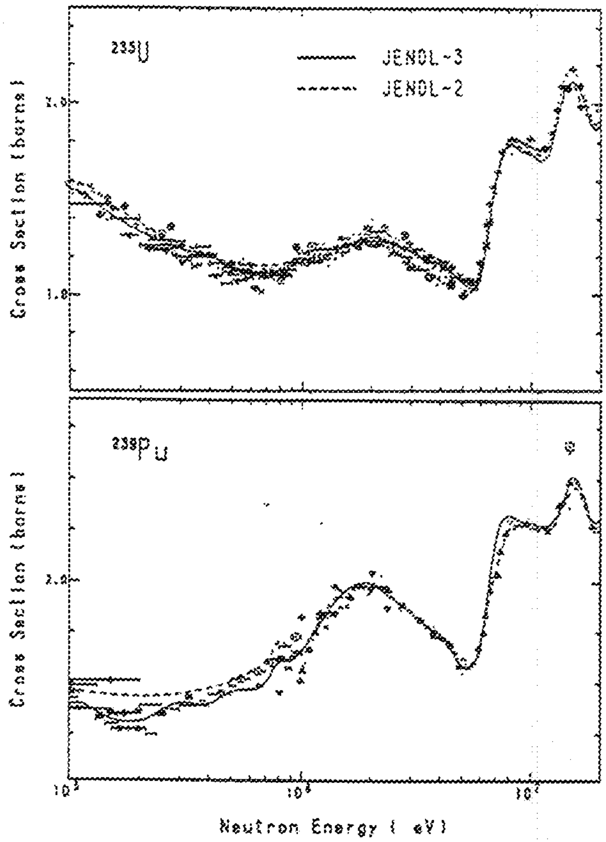

Fig. 5 Fission cross sections of ^{235}U and ^{239}Pu

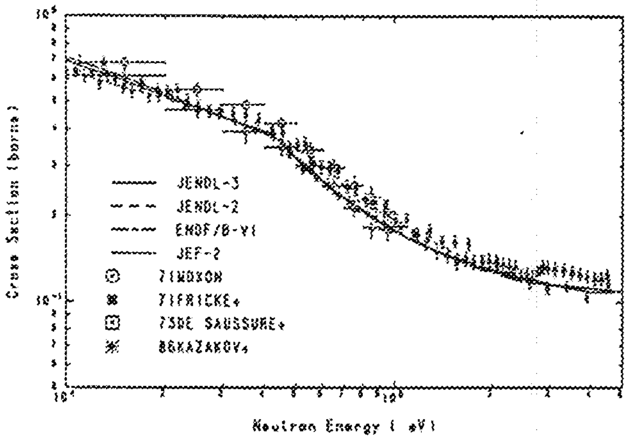

Fig. 6 Capture cross section of ^{238}U

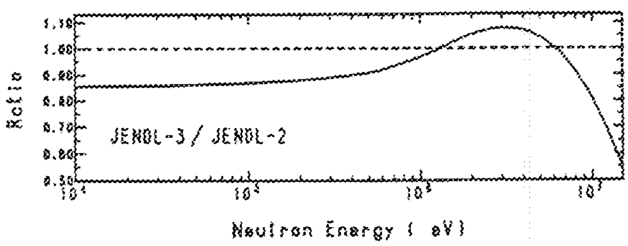

Fig. 7 Fission neutron spectrum of ^{239}Pu

<u>Fission neutron spectrum</u>: The prompt fission neutron spectrum formula of Madland and Nix[7] were adopted for ^{232}U, ^{234}U, ^{235}U, ^{238}U, ^{239}Pu and ^{240}Pu. This type of spectrum has larger average neutron energy than the Maxwellian or Watt types. The spectra for ^{239}Pu are shown in Fig. 7. The Maxwellian spectra were adopted for the other nuclides.

<u>Gamma-ray Production Data</u>

The gamma-ray production data were evaluated mainly on the basis of the statistical model with the pre-equilibrium effects. As will be described later, the gamma-ray spectra thus calculated was found to reproduce the measured data fairly well except the thermal neutron capture gamma-ray. Hence the gamma-ray data caused by the thermal neutron capture were re-evaluated on the basis of the available experimental data.

<u>Benchmark Tests</u>

<u>Simple Fast Benchmark Assemblies</u>[8]

Fast assemblies of simple geometry and simple components are useful to know the balance of data for main fissile nuclides. Eight assemblies such as JEZEBEL were chosen. The calculation was made with the ANISN code with S_{16} and P_5 approximation by using the group cross sections of 175 group structure. The C/E values are given in Table 2 for k-eff and central fission rate ratio of ^{238}U to ^{235}U. Very satisfactory results were obtained for the Pu and ^{235}U cores. For the ^{233}U cores, however, the k-eff values and the ^{238}U/^{235}U fission rate ratios are overestimated by 2 % and by 7 %, respectively. The sensitivity analysis revealed that the fission neutron spectrum of ^{233}U might be too hard and that the inelastic scattering cross sections might be too high. Some modification should be made on these quantities in future.

Table 2 C/E values of k-eff and fission rate ratio of ^{238}U to ^{235}U

Core		k-eff	$<^{28}\sigma_f> / <^{25}\sigma_f>$
Pu	JEZEBEL	1.0001	0.9944
	JEZEBEL-PU	0.9963	1.0063
	FLATTOP-PU	0.9974	1.0117
	THOR	0.9985	1.0034
^{235}U	GODIVA	1.0066	1.0006
	BIG TEN	1.0038	1.0195
	FLATTOP-25	1.0033	1.0697
^{233}U	JEZEBEL-23	1.0206	1.0619
	FLATTOP-23	1.0175	1.0696

<u>Fast Critical Assemblies</u>

For fast reactor benchmark tests of JENDL-1 and -2, 21 assemblies with one-dimensional model have been used, which consists of 18 international benchmark assemblies, two MOZART cores (MZA and MZB) and FCA-V-2. In the present tests, the JUPITOR reference core (ZPPR-9) and FCA-IX cores were added*. Most of the results were already published elsewhere[9].

* FCA-IX series consist of 7 EU cores. FCA-IX-1 ~ 3 are graphite moderated, IX-4 ~ 6 are SUS moderated and IX-7 is a 20 % EU metal core. Trends of prediction with JENDL-3 are very different between graphite moderated and SUS moderated cores. Hereafter FCA-IX-C represents the average of IX-1~3 and IX-S that of IX-4~6.

Effective multiplication factor(k-eff):Table 3 The C/E values of JENDL-3 are satisfactory as seen in Table 3. It is found, however, that the k-eff values of U cores are lower by 0.7 % than those of Pu cores. As to FCA IX cores, the C/E values are much underestimated for IX-G cores, which are graphite moderated, while satisfactory C/E values are observed for SUS moderated cores.

Table 3 C/E values of k-eff

Cores	JENDL-2	JENDL-3
21 Benchmark cores	1.009	1.002
Pu cores	1.004	1.004
U cores	1.005	0.997
ZPPR-9	0.999	1.006
FCA-IX-C	0.992	0.984
FCA-IX-S	1.012	1.006
FCA-IX-7	1.009	1.007

Central reaction rate ratios: Table 4
The C/E values of important ratios are given in Table 4. JENDL-3 gives better prediction for fission rate ratios of ^{239}Pu and ^{235}U except for FCA-IX-C. This good balance may come from our adoption of simultaneous evaluation. As for the ratio of ^{238}U fission to ^{235}U fission, JENDL-3 looks to overestimate the ratio for 21 benchmark cores and FCA cores. This ratio is, however, sensitive to the measurements, i.e., perturbation caused by the detectors. Most of these measurements were made with micro fission chambers and the perturbation must be serious. On the other hand, the recent measurements in ZPPR-9 were made with foil techniques and very precise analyses were made. As the C/E value of ZPPR-9 is near unity, we believe that this quantity is well predicted. The ratio of ^{238}U capture to ^{239}Pu fission is well predicted for the benchmark cores, but is overestimated by 4 % for ZPPR-9. As this ratio is an important parameters for breeding ratio, further study will be required.

Table 4 C/E values of central reaction rate ratios

Quantities*	Assemblies	JENDL-2	JENDL-3
$\frac{F(^{239}Pu)}{F(^{235}U)}$	21 Benchmark		
	Pu cores	0.97	0.99
	U cores	0.99	0.99
	All cores	0.97	0.99
	ZPPR-9	0.98	1.00
	FCA-IX-C	1.03	1.05
	FCA-IX-S	0.97	0.98
	FCA-IX-7	1.02	1.02
$\frac{F(^{238}U)}{F(^{235}U)}$	21 Benchmark		
	Pu cores	1.05	1.12
	U cores	0.98	1.04
	All cores	1.03	1.10
	ZPPR-9	0.94	1.00
	FCA-IX-C	1.22	1.30
	FCA-IX-S	1.12	1.19
	FCA-IX-7	1.05	1.11
$\frac{C(^{238}U)}{F(^{239}Pu)}$	21 Benchmark		
	Pu cores	1.02	1.00
	U cores	0.96	0.94
	All cores	0.99	0.98
	ZPPR-9	1.05	1.04

* F means fission and C means capture

Doppler coefficients: The C/E values of Doppler coefficients of natural UO_2 samples in ZPPR-9 were calculated. The average C/E value with JENDL-3 is 0.94 and better than that of 0.91 with JENDL-2.

Sodium void coefficients: Fig. 8
The sodium void coefficients were much overestimated (by 20 %) with JENDL-2, and the overestimation becomes larger when the voided volume increases as seen in Fig. 8. With JENDL-3, the overestimation is much reduced and the C/E values show little dependence on the voided volume.

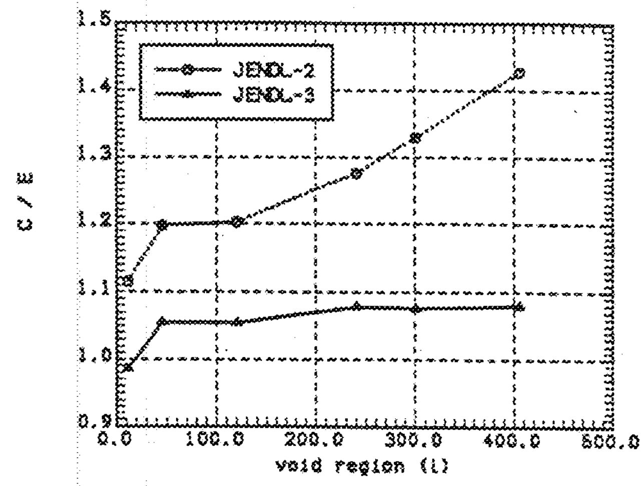

Fig. 8 Sodium void coefficients in ZPPR-9

Space dependence: With JENDL-2, considerable space dependences were observed for the C/E values of reaction rates, control rod worths and sodium void coefficients. These space dependences are much reduced with JENDL-3. For the ^{239}Pu fission rate distribution in ZPPR-9, the C/E values remain ± 2% even in the outer core, while they reach 6% in the outer core with JENDL-2. As to the control rod worths in ZPPR-9, the maximum C/E deviation is 2% with JENDL-3, while it is 4.5% with JENDL-2 as is seen in Fig. 9.

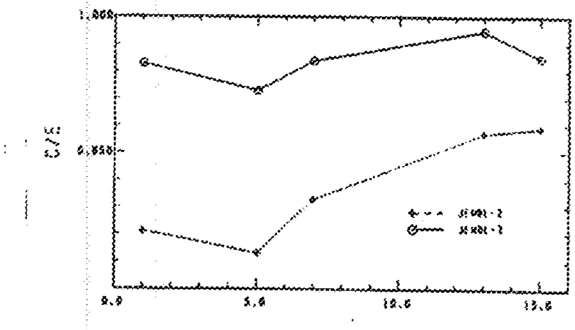

Fig. 9 Control rod worths in ZPPR-9

Thermal Benchmark Tests

Various thermal critical assemblies have been analyzed. The benchmark calculations were performed with SRAC, a thermal reactor standard code for reactor design and analyses. Some results were already published [8].

Here the results on TRX-1, -2, BAPL-1, -2 and -3 are shown, which are U- cores selected for 'WIMS Library Update' project proposed by IAEA. Table 5 shows the C/E values of k-eff and lattice parameters calculated with JENDL-3 and JENDL-2. Both JENDL-2 and -3 slightly underestimate the k-eff values but predict well the lattice parameters.

Table 5 The C/E values of k-eff and lattice parameter for thermal assemblies

Core	Quantities+	JENDL-2	JENDL-3
TRX-1	k-eff	0.9939	0.9956
	ρ-28	1.02	1.02
	δ-25	1.00	0.98
	C*	1.01	1.00
TRX-2	k-eff	0.9963	0.9979
	ρ-28	1.00	1.00
	δ-25	0.99	0.97
	C*	1.00	0.99
BAPL-1	k-eff	0.9956	0.9953
	ρ-28	1.01	1.01
	δ-25	1.00	0.98
BAPL-2	k-eff	0.9966	0.9960
	ρ-28	1.04	1.04
	δ-25	1.01	0.99
BAPL-3	k-eff	0.9979	0.9977
	ρ-28	1.01	1.01
	δ-25	1.01	0.99

ρ-28: Ratio of epithermal to thermal ^{238}U capture.
δ-25: Ratio of epithermal to thermal ^{235}U fission.
C* : Ratio of ^{238}U capture to ^{235}U fission.

Fusion Neutronics

As one of the main aims of JENDL-3 is its fusion application, intensive testing has been made for fusion neutronics benchmarking. Most of benchmark experiments were selected from those made in Japan such as TOF neutron spectrum measurements, tritium breeding and neutron multiplication experiments and gamma-ray leakage spectrum measurements performed in FNS and OKTAVIAN facilities. The precise discussion was published in Ref. [10].

TOF Spectrum Experiments are useful for direct validation of inelastic scattering and neutron emission reaction data. Very satisfactory agreement has been observed for most of nuclides. Figure 10 shows the spectrum of Oxygen.

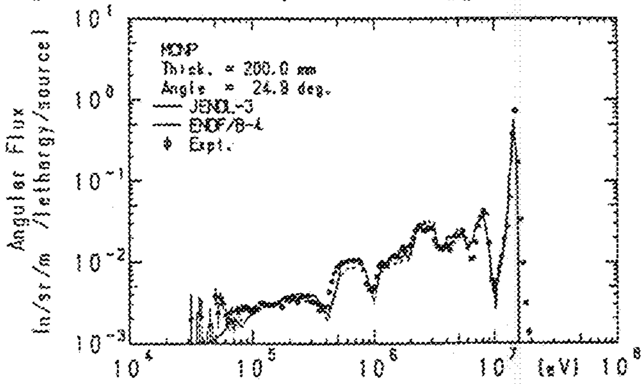

Fig. 10 Leakage neutron spectrum from liquid Oxygen slab

Integral Experiments on tritium breeding rate or neutron multiplication by Be or Pb were performed in FNS and OKTAVIAN facilities and analyzed with JENDL-3. It was found that the tritium breeding rate could be predicted within error of 5 ~ 10 % for a blanket consists of Li and Be.

Gamma-ray benchmarking: The gamma-ray leakage spectrum and gamma-ray heating experiments were analyzed. The results were satisfactory for Li_2O, C, Al and Si but some problems were pointed out for Be and W. Figure 11 shows the results of gamma-ray heating in a Be assembly.

Conclusion: As to the neutron data, JENDL-3 is satisfactory for most of cases. For some important nuclides such as Be, it was pointed out that the angle and energy distribution of inelastic scattering and (n,2n) reaction should be further modified. The gamma-ray benchmark tests are still going on.

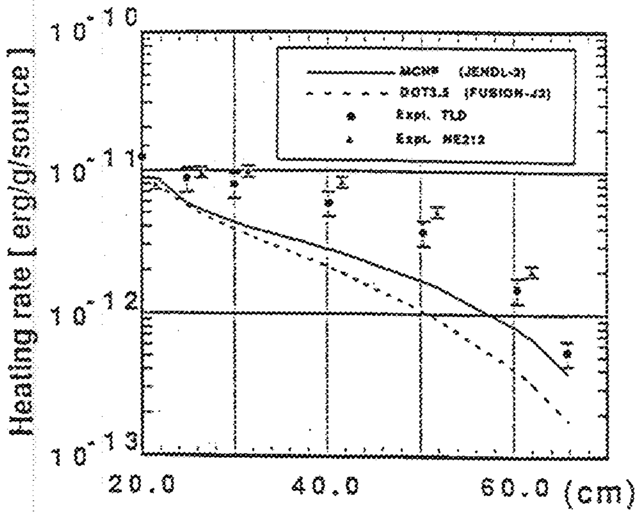

Fig. 11 Gamma-ray heating rate in Be slab

Shielding

As for neutron shielding, typical 9 benchmark experiments have been analyzed, such as Broomstick, ASPIS, ORNL, KFK etc. The results were satisfactory as a whole. Figure 12 shows neutron fluxes in iron of the ASPIS experiment.

For secondary gamma-ray shielding problems, studies are going on in cooperation with the fusion neutronics benchmarking group.

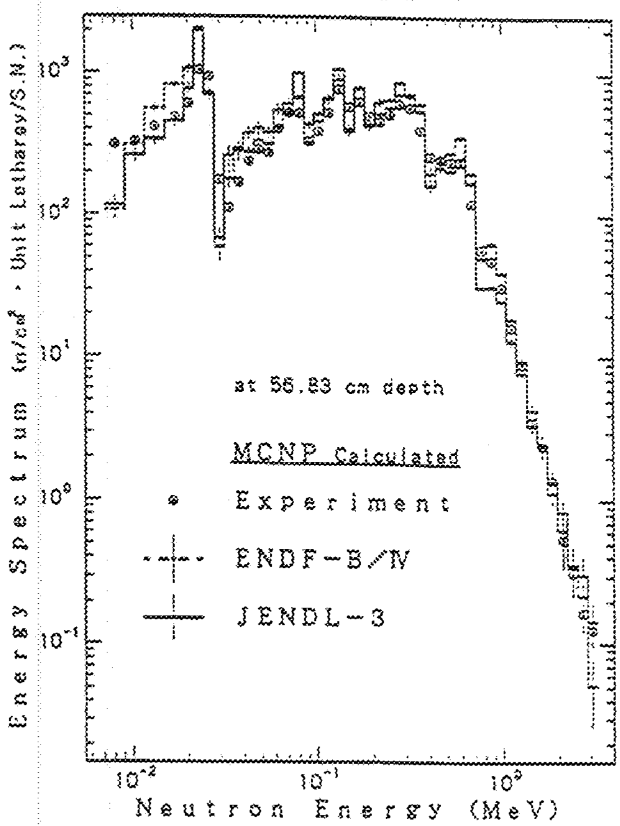

Fig. 12 Neutron energy spectra at 56.83-cm-depth in the iron shield of the ASPIS experiment

Gamma-ray Production Data

Emitted gamma-ray spectra arising from thermal neutron capture and fast neutron reactions were calculated and compared [11] with the experimental data measured at ORNL Tower Shielding Facility. The agreement was satisfactory for gamma-rays arising from fast neutron reactions. It was found, however, that the gamma-ray spectra caused by thermal neutron capture could not be well reproduced with JENDL-3T. Hence re-evaluation works were made and the modified data were adopted in JENDL-3.

Dosimetry

As one of JENDL special purpose files, a dosimetry file was provided and its applicability has been tested. The cross section data were mainly taken from JENDL-3 general purpose file and the covaricance data were taken from IRDF-85. Total of 61 reactions were adopted.

The data were tested with the activation data measured in various standard spectra. Figure 13 shows the C/E values in ^{252}Cf spontaneous fission spectrum (NBS standard) and CFRMF spectrum. Most of C/E values fall on unity within the experimental errors. Disagreement is remarkable, however, for ^{58}Ni(n,2n), ^{63}Cu(n,2n) etc. Both the cross sections and the integral experiments are being studied for these cases.

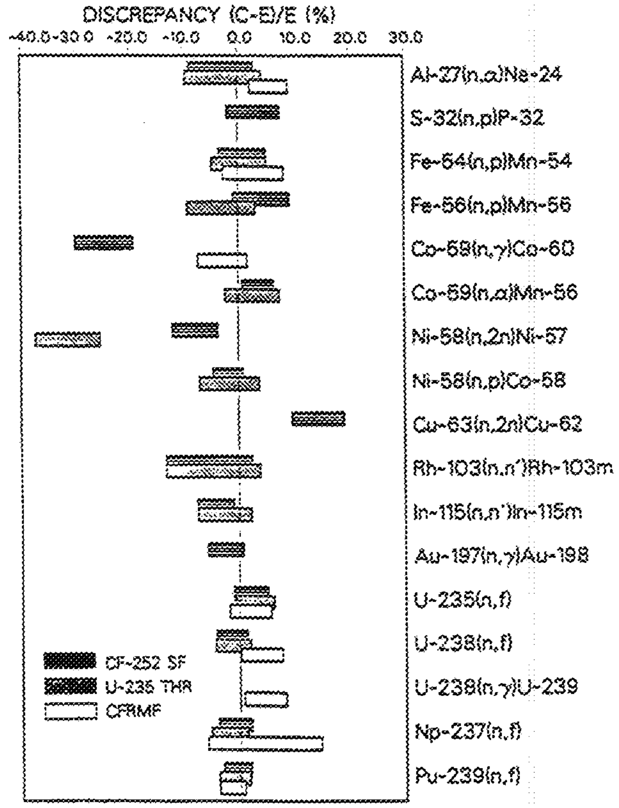

Fig. 13 C/E values of dosimetry reaction rates in ^{252}Cf fission and CFRMF spectra

Table 6 JENDL Special Purpose Files

File	Contents*	Completion
Dosimetry	61 R	1991
Gas production	23 N	1991
Activation	1000 R	1991
(α,n)	11 E	1991
KERMA/DPA	46 E	1992
Actinide	73 N	?

* R: Reaction, N: Nuclides, E: Elements

Future Scope

Revision of JENDL-3

The results of the benchmark tests have been satisfactory as a whole. JENDL-3 will be used widely for general purposes. For some nuclides, however, drawbacks have been already pointed out. More problems will be revealed in near future. The revision works have already started. The revised version of JENDL-3 (JENDL-3 Revision 2*) will be released at the end of 1993.

Special Purpose File

Aside from JENDL general purpose file, various special purpose files will be provided. They are tabulated in Table 6.

High Energy Nuclear Data Files

Recently high energy accelerator techniques become to attract much attention particularly from the viewpoint of transactinide burning, accelerator breeding, high intensity neutron sources etc. These techniques need nuclear data for high energy incident particles, which are beyond the scope of JENDL-3. These high energy nuclear data will be one of the main challenging targets in future.

Evaluation of neutron data up to 50 MeV was already started for the conceptional design of ESNIT project (d-Li neutron source energy selective irradiation test facility). Evaluation of high energy neutron and charged particle (up to a few hundred MeV) induced nuclear data is now under planning. The photo reaction cross section data up to 100 MeV is now going on.

* Revision 1 was already released in 1990 after correcting trivial compilation errors.

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