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BENCHMARK TESTINGS OF EVALUATED DATA FILES ON FISSION REACTORS AND SHIELDING PROBLEMS

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Abstract: Description of the JENDL-3 project, which is now going on in Japan aiming at standard domestic library of microscopic cross sections, are given stressing on the benchmark tests. Historical review of JENDL development is given first, and next scope of JENDL-3 and high-light of JENDL-3 evaluation are described. JENDL-3T, an adhoc file, is compiled for the testing of JENDL-3. Applicability of JENDL-3T nuclear data file to fission reactors(thermal and fast reactors) and shielding applications is checked through the wide range of the benchmark test performed by JNDC. From the results, overall predictability of JENDL-3T found to be satisfactory, however, some problems were also pointed out. By the reflection of these feedback information forth coming official version of JENDL-3 is expected to be greatly improved.

(JENDL-3, nuclear data, cross-sections, benchmark test, integral test, FBR, LWR, HCLWR, Shielding)

Introduction

There are prominent nuclear data files in the world. These files have been always being maintained according to the needs of those who want to predict neutronic characteristics of reactor cores, such as K_{eff} or sodium-void reactivity coefficients, very accurately or to determine the optimal shielding configurations. Requests from the users are very severe so that only a few data in the files are fulfilled. Thus revision or version up of data file is inevitable.

In Japan JENDL-3(Japanese Evaluated Nuclear Data Library version 3) project is now going on. This project is started to realize the highly consistent files relying both on the latest differential data and very sophisticated theoretical calculations for the cross-sections. This project is now in the final stage for the release of JENDL-3. Recompilation work is progressing by the feedback information obtained from the benchmark tests.

In this paper, firstly historical overview of the JENDL development is described, next scope of the JENDL-3 and high-light of the evaluation in JENDL-3 are presented. Then the results of the benchmark test performed for the confirmation of the applicability of JENDL-3T, an adhoc file of JENDL-3, are described briefly for the fission reactors(LWR, HCLWR and FBR) and shielding applications.

Here presented works were performed by all members of the Subcommittee on Nuclear Data and those of the Subcommittee of Reactor Constants of Japanese Nuclear Data Committee(JNDC).

Historical review of the development of JENDL

The **Japanese Evaluated Nuclear Data Library (JENDL)** has been developed as the standard domestic library of microscopic cross sections by JAERI Nuclear data Center with cooperation of the Japanese Nuclear Data Committee (JNDC). In 1970 JNDC decided to make JENDL for the general purpose library, and **JENDL-1** project started in 1974. It's first version JENDL-1 /1/ was released in 1977, mainly aimed to provide data for fast reactor calculations. In this version only 72 nuclides are contained including 28 fission products(FP) nuclides. Energy range is 1.0E-5 eV to 15.MeV. ENDF/B-IV format system is adopted for the storing format. After the benchmark test for FBRs it was released but the number of nuclides is insufficient for applications other than FBR.

The second version (**JENDL-2**) /2/ project started in 1977. This version has aimed at wider applications such as thermal reactor, radiation shielding, fusion neutronics, nuclear fuel cycle etc. It was completed in April 1983. The number of nuclides are enlarged up to 181 including 100 FPs. The energy range is also enlarged up to 20 MeV. Adopted format system is the same as JENDL-1. In this evaluation, data of the structural materials and fissionable materials are improved. After the benchmark test of FBRs, it was released in 1982.

JENDL-2 was used for JUPITER (Japanese United States Program of Integral Tests and Experimental Researches), a joint USA-Japan mock-up experiments of large fast reactors using ZPPR facilities, ZPPR-9,10 and 13. The results were very good except Na-void reactivity coefficients, space dependence of C/E(ratio of the calculation to experimental value) for control-rod worth and power distributions /3/. Reliability was proved.

Parallel with the compilation work of JENDL-2, the **JENDL-3** project was started to enlarge the applicability for more general applications like Thermal, FBR, Fusion, Shielding, Dosimetry, Burn-up physics or Fuel cycle. The number of nuclides is expected to be enlarged up to 304 including 170 FPs and adopted format is decided to be ENDF/B-V.

For fusion neutronics applications, the data in the JENDL-2 were, however, found to be seriously inadequate in the energy range above several MeV . According to the strong requests from the analysts of the Japan-US cooperative fusion experiments using FNS (Fusion Neutronic Source) of JAERI and the university jointed programs on fusion experiments using OKTAVIAN of Osaka University, a preliminary version for JENDL-3, **JENDL-3PR1** /4/ was prepared at the end of 1983, in which only 8 nuclides; Li6, Li7, Be-9, C-12, O-16, Cr, Fe and Ni are contained. In 1985 **JENDL-3PR2** /5/ was released which was the revised one of JENDL-3PR1 only for the nuclides of Li-6,Li-7 and C-12. For fusion neutronics application, these two files have been mainly used up to now.

In 1987 May, JENDL-3T file is completed /6/ for the sake of benchmark testing, the file contains 73 nuclides with photon production cross section of 32 nuclides. After the benchmark test, within the 1988 fiscal year, JENDL-3 is expected to be released. Current status of JENDL-3T is given in the paper of Dr. Asami /7/ of this conference.

Scope of JENDL-3

Characters of JENDL-3 are described as follows:
1. The number of nuclides is enlarged up to 304 which is comparable to that of ENDF/B-V which has 289 nuclides.
2. Photon production cross-sections are newly introduced for 32 nuclides.
3. Improvements of high energy neutron data are highly considered taking account of direct and pre-equilibrium process for particle emissions.
4. Double differential cross section data measured at Osaka Univ. and Tohoku Univ. are highly considered at the evaluation stage. These data are essential for fusion neutronics.
5. Data refinements through the systematic studies for nuclear model parameters and simultaneous evaluations for fission and capture for important heavy nuclides.
6. Storing the data for individual isotopes even for the stable natural element is made, for example Fe-natural and Fe-54,56,57,58. This is for the convenience of the calculations of induced radio-activity, radiation damage or KERMA factors.
7. No error-file was supplied, due to the man power problem. Although the requests from the users are very keen.
8. Throughout the evaluation of JENDL-3, we do not take the standard file concept, i.e., no standard file is supplied. Instead we use the simultaneous evaluation methods for main heavy fissionable nuclides. Recommended from the simultaneous evaluation (fission, capture for main fissile material) are the standard data in JENDL-3.

High-light of the evaluation in JENDL-3

1. Simultaneous evaluations

Up to JENDL-2 evaluation, independent individual evaluations are used using standard file concept. In JENDL-3, simultaneous evaluation is made using generalized least-square fitting by B-spline functions /8/. Used data in this processing are fission cross-sections for U-235, U-238, Pu-239, Pu-240, Pu-241, capture cross-sections for U-238 and Au-197, and ratio data for F28/F25, F49/F25, C28/F25 and C28/F49 (F: fission, C: capture, 25: U-235, 28: U-238, 49:Pu-239). Energy range adopted for the simultaneous evaluations are from 50 keV to 20 MeV. Recent measurement data are mostly used. The results are shown in Fig.1 and 2. From these figures, JENDL-3T results traces the measured data quite well. But distinct difference are clearly seen between JENDL-2 and JENDL-3T. JENDL-3T evaluation for fission data of U-235 and Pu-239 indicates systematically low values compared with JENDL-2 evaluation. And for U-238 capture evaluations rather larger values than JENDL-2 is resulted. This might cause serious problem for FBR benchmarks. Because predictability of JENDL-2 was quite well, 1--2% less reactive profile for FBR cores are foreseen.

For the results from the simultaneous evaluations, two different point of view are arose. Evaluator's stand point: From the recent measurements, JENDL-3T data are the best. On the other hand for the benchmark tester's standpoint: Are these data so reliable? We cannot accept such a poor criticality predictable data. Because so many experimental data were rejected in the selection of the input data. Are these data so bad to have to be rejected? And we cannot neglect systematic errors associated to each individual experiment between the different measurements. To this points, simultaneous procedure is vital enough? Big discussions are still continuing.

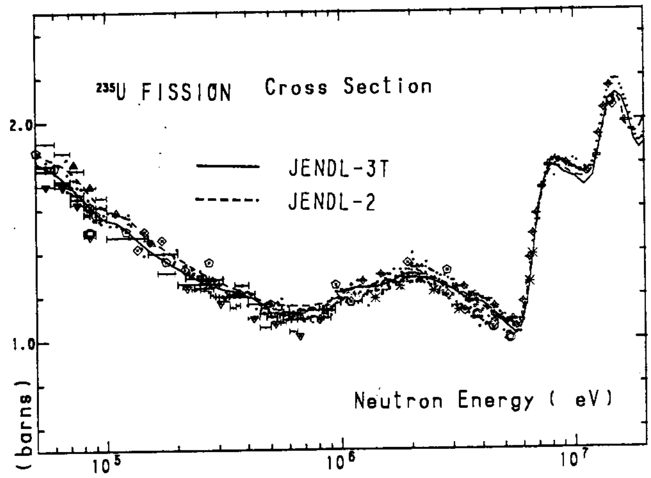

Fig.1 U-235 fission cross section

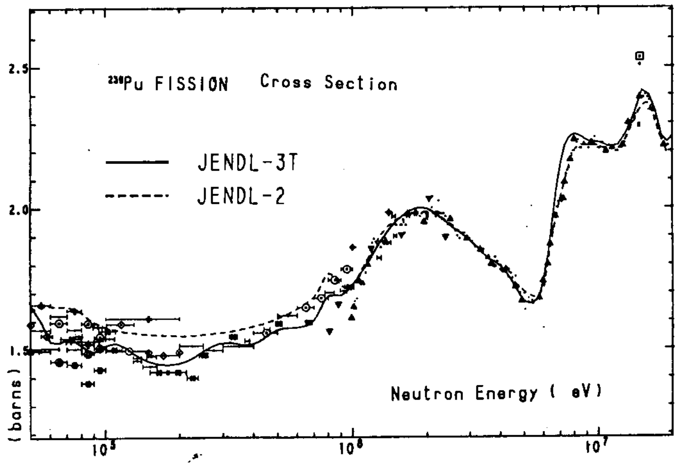

Fig.2 Pu-239 fission cross-section

2. Inelastic cross section

In high energy range, direct process represented as the collective motion and single particle transition is essential for the inelastic cross-sections. For such calculation a coupled channel code: ECIS /9/ and/or a DWBA code: DWUCK-4 /10/ is used. Up to JENDL-2, these processes are not taken into account.

For example, U-238 inelastic cross-sections in JENDL-2, no direct process is considered. Therefore all cross-sections in discrete level above 2.6 MeV were set to 0. Above 2.6 MeV, only continuum treatment is employed. In JENDL-3T, direct process is considered by ECIS and DWUCK-4 and so obtained data are largely different from JENDL-2 as shown in Fig. 3. JENDL-3T data should have been improved compared with that of JENDL-2 from the stand point of considering the direct process. But how accurate? It is still question. Because no experimental data are available for this total inelastic cross-section, the evaluated data are only constructed from the so sophisticated calculations. We have no means to assess so obtained data by the experimental data. Thus benchmark test will play an important role for the assessment of these data.

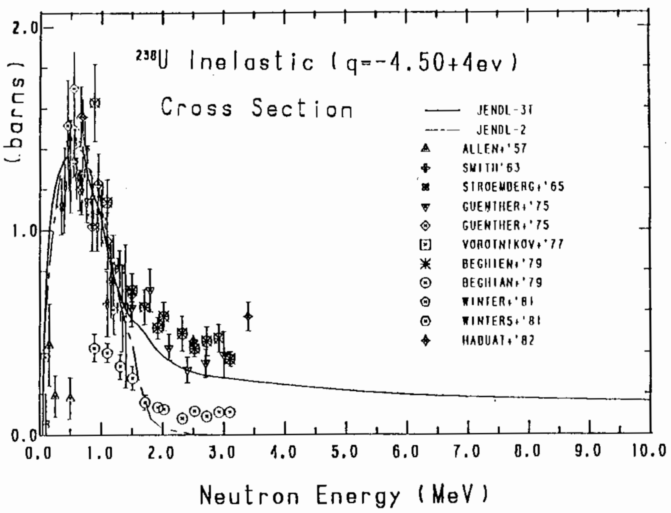

Fig.3 U-238 inelastic scattering(1-st level) cross section

3. High energy region cross sections

The special emphases have been put on the high energy neutron data such as charged particle cross sections, gas production cross sections, neutron emission cross sections or (n,2n) cross sections in the evaluation. Double differential data: DDX(neutron emission data, the energy-angle double differential cross-sections) are carefully checked with the experimental data. Also threshold reaction data are carefully evaluated, because these data are essential for fusion and dosimetry applications. Relatively large change have been made from JENDL-2 to JENDL-3.

4. Resonance parameters

J-unknown state assignments:

J-unknown state assignment, this ia a trick made by an invalid J number assignment to indicate J-unknown state, which have been a local definition valid only for JENDL-2, is completely removed in JENDL-3. This local assignments had been the largest problematic problem in JENDL-2.

Resonance region enlargement

Resonance region is expanded due to the availability of the new experimental data. For example, U-238 of JENDL-3 a new parameters obtained by Olsen of ORNL /11/. Number of resonances are considerably increased, for example, s-wave resonances increases from 187 to 356, p-wave from 265 to 485. High end of resolved range is moved from 4keV in JENDL-2 to 9.5 KeV in JENDL-3T. This is made according to remove the ambiguity due to the unresolved resonance parameter representation which sometimes brings significantly different self-shielding factors by calculation method. Because no self-shielding information was used in deriving the unresolved resonance parameters. Increase of the number of parameters affects the computing time considerably for the reconstruction of the resonances shape cross sections. In this case more than 1-hour of CPU time by FACOM M/780, is necessary really time consuming calculations.

Reich-Moor multi-level parameters

For Pu-239 resonance parameters Reich-Moor multi-level parameters given by Derrien et al. /12/, are adopted as JENDL-3T evaluation. But Reich-Moor parameter is deactivated options in ENDF/B-V format, so we must use ENDF/B-IV format only for this parameters. Only a few code such as RESENDD /13/ can accept this parameter. Processing code may be one of the problems for this parameter. Up to JENDL-2, SLBW(Single level Breit Wigner) parameters were utilized, but the reproducibility of the resonance cross section shape (especially fission) between the resonances was rather poor for this SLBW parameter representation due to the interference effects.

Benchmark testing project of JENDL-3

To confirm the applicability of JENDL-3, a project of benchmark testing is now in progress in Japancsc Nuclear Data Committee. To cover very wide range of the application fields foreseen to this file, sub working groups for each testing fields are organized in the Subcommittee on Reactor Constants of JNDC with the cooperation between Nuclear Code Committee and Research Committee of Reactor Physics of Japan. Testing fields and responsible sub-working groups are listed in Table 1.

Table 1. Benchmark testing fields

applications fields	sub-working group	group leader
LWR	LWR-S.W.G.	H.Takano (JAERI)
HCLWR	LWR-S.W.G.	"
FBR	FBR-S.W.G.	T.Takeda (OSAKA U)
Shielding	Shielding S.W.G.	M.Kawai (NAIG)
Fusion neutronics	Fusion neutronics S.W.G.	H.Maekawa(JAERI)
Dosimetry	Dosimetry data S.W.G.	M.Nakazawa(U TOKYO)

As to the other benchmark tests except above described, for example, criticality safety, facility safety or spent fuel cask applications are also considered. Because these benchmark test are somewhat application oriented and the work is thought outside the scope of JNDC, it has decided to be managed by the cooperation between Nuclear Code Committee, especially working group on the nuclear safety code and JNDC.

JENDL-3T file, containing only major nuclides for nuclear energy applications, has been prepared so as to be applied to the benchmark testing prior to the fixing of the final version of JENDL-3. Thus JENDL-3T is a temporary version of JENDL-3, where 'T' stands for temporary, testing or tentative version, and it is destined to be the official version of JENDL-3. To confirm the applicability of the evaluated data delivered from the evaluator, this benchmark testing is planned and performed. Thus the results from this benchmark test will be reflected to the final version through the discussions by the compilation group of JENDL and the evaluator.

Up to now first round testing have been finished and so many observations have been obtained for the primary nuclides important for the nuclear energy applications. Some results of the benchmark tests from the fission reactor related applications are described here after.

As to the applicability to the fusion neutronics or to the reactor dosimetry, we cannot mention at all in this paper. For the former, please refer to the Dr. Maekawa's paper /14/ , and the latter, to the Prof. Nakazawa's paper/51/.

LWR(Light Water Reactor) and HCLWR(High Conversion LWR) benchmarks

For the JNDC, these benchmark tests are new faces. Up to the JENDL-2 release, no feedback are made from thermal reactor benchmarks. Because not so many users were foreseen for this fields, benchmark tests were not performed. Therefore insufficient thermal data which were stored in JENDL-2 file have been pointed out several times by users.

Especially for LWR, users possess their own libraries tested by a number of their startup physics data or operational data of real power reactors. Because they thought their library is perfect for the moment, they do not want to replace their own library having been tuned by the existing reactors to the new data base like JENDL-2 or -3T. However this situation is gradually changing, new trends such as longer cycle of operation, high burn-up operation, Pu recycling or new type reactor development represented by HCLWR , are moving in the nuclear industries, there are increasing necessity for the accurate cross section data for these purposes which have never been experienced yet. There have been no confident data base for these new applications.

Benchmark calculations were performed using SRAC, a thermal reactor standard code system for reactor design and analysis/15/, by the thermal and high conversion reactor benchmark test sub working group(SWG) of JNDC leaded by H.TAKANO of JAERI. Due to the space problems, we only mention the brief results here, the detailed discussions are also given in the paper by Dr. H.TAKANO et. al.,/16/ please read their paper.

LWR: Light water reactor benchmarks

The following benchmark assemblies are selected as benchmark test problems: a number of critical experiments with different fuels of U-235, U-233 and Pu-239 of Jakins/17/, two water moderated lattice(TRX-1 and 2) /18/ and a large number of uniform water moderated lattices collected by Strawbridge and Barry /19/.

The same calculations are applied for both of the files i.e., JENDL-3T and -2. The cell spectrum calculations were performed by the collision probability method. Criticality calculations were performed with P_1-S_8 by one-dimensional Sn transport code ANISN/20/.

Results and discussions

Comparison of statistics for the obtained multiplication factors for Strawbridge and Barry's benchmarks is shown in Table 2.

Table 2. Statistics of Keff for Strawbridge et.al.

averaged keff		JENDL-2	JENDL-3T
UO2 lattices	(55 cases)	0.983(0.011)	0.991(0.012)
U-metal	(61 cases)	0.989(0.009)	0.992(0.009)

n.b. the number in the parenthesis shows standard deviation

From this table, about 0.8% increase in K_{eff} is found for JENDL-3T. The magnitude of under-estimation becomes large along with the increase of the H/U (number density ratio of hydrogen to U) ratio.

U-235 fueled assemblies

K_{eff} increase by 0.3 % in JENDL-3T is observed. This is due to the nu data in thermal energy range of U-235. For well moderated case agreements are quite well in JENDL-3T. However for the metal fuel assemblies with a hard spectrum, as large as 2% K_{eff} overestimation is observed. The same result is obtained for the heavy water moderated case and H/U=0. case of Jakins. This is responsible for the too large nu data in the fast energy range and a hard Madland and Nix's fission spectra.

Pu-239 fueled assemblies

About 0.6% constant less reactive profile along with H/Pu-239 ratio is obtained for JENDL-3T compared with that of JENDL-2. This is due to lower fission cross section of Pu-239 resolved resonance range.

U-233 fueled assemblies

Agreements by JENDL-3T becomes quite well. From the analysis of lattice cell parameters for TRX and ETA, Th-232 capture related parameters such as Th-232 capture epithermal to thermal ratio or Th-232 capture to U-235 fission ,etc., are greatly improved in JENDL-3T. This is due to the large Th-232 capture cross section data compared with that of JENDL-2.

HCLWR: High Conversion Light Water reactors

Selected benchmark cores are following two assemblies:

The first one is the PROTEUS core series/21/, which is a tight lattice experiments with the moderator-to-fuel volume ratio of 0.5 simulating high conversion light water reactors. There are two series of experiments of which fissile Pu enrichment are about 6 and 8 % . In each cores, three different H_2O voidage states were measured ,i.e., 0, 42.5 and 100%, to check the void reactivity coefficient, which is a key parameters for the safety aspects of the reactors.

The second one is the FCA-XIV-1 core/22/, which is a plate type experiment fueled with the enrichment of 6.5 % U-235 and the moderator-to-fuel volume ratio of 0.6.

Results and discussions

Pu-239 core : PROTEUS

K_{inf} is well predicted by JENDL-3T , however at the progressive void stages predictability of K_{inf} is not sufficient compared with that of JENDL-2.

For the C/E of the reaction rate ratio for C28/F25(U-238 capture to U-235 fission), corresponding to the conversion ratio of the core, JENDL-3T result becomes worse than that of JENDL-2. The same tendency is found for the C/E values of F28/F25(U-238 fission to U235 fission) for JENDL-3T. That is, spectral indices in high energy region seems to be too large for JENDL-3T results. The similar results are obtained in the FBR benchmarks.

U-235 core: FCA-XIV-1

No improvements are observed for all items analyzed using JENDL-3T than those by JENDL-2. Over-estimation in F28/F25, C28/F25 observed in JENDL-2 is more enhanced in JENDL-3T. This enhancement of C/E values are significantly attributed to the nu values of U-235 data in fast energy range. Thus improvements are foreseen by the modification of the nu-values of U-235 in JENDL-3T.

FBR benchmark test

The benchmark test for FBR has been performed using 21 cases of one-dimensional model and several two-dimensional models of fast critical assemblies. Selected integral data are K_{eff}, reaction rate ratio, sodium void worth and Doppler reactivity worth, which are key design parameters for the economical and safe FBRs.

This benchmark test was performed by H.Takano (JAERI) and T.Takeda (Osaka Univ.) at the FBR benchmark test SWG of JNDC. Their presentation on this subject is also scheduled in this conference, for details please also refer to their papers /16, 23/.

Benchmark test with one-dimensional model

The benchmark tests were made with the same manner as used in the test of JENDL-1 /24/ and -2 /25/. The selected assemblies are listed in Table-3 with their characteristic features, 18 of which are those selected by Hardie et al./26/, two are MOZART cores and one is a FCA core. The same calculations are made using JENDL-3T and JENDL-2. The results are shown mostly in C/E basis for each integral data items.

Table 3. Benchmark problems adopted for FBR tests

	Assembly	Fuel	Fer/Fiss	Vol(l)	comments
1	VERA-11A	Pu	0.05	12.	Pu+C,No U in core
2	VERA-1B	U	0.07	30.	94% EU+C
3	ZPR-3-6F	U	1.1	50.	
4	ZEBRA-3	Pu	8.6	60.	Hard spectrum:80% abv. 100keV
5	ZPR-3-12	U	3.8	100.	Soft spectrum, due to added C
6	SNEAK-7A	Pu	3.0	110.	
7	ZPR-3-11	U	7.5	140.	
8	ZPR-3-54	Pu	1.6	190.	Similar to 3-53 Fe reflector
9	ZPR-3-53	Pu	1.6	220.	U reflector
10	SNEAK-7B	Pu	7.0	310.	
11	ZPR-3-50	Pu	4.5	340.	(ZPR-3-48) with additional C
12	ZPR-3-48	Pu	4.5	410.	Soft spectrum due to added C
13	ZEBRA-2	U	6.2	430.	
14	ZPR-3-49	Pu	4.5	450.	(ZPR-3-48) without Na
15	ZPR-3-56B	Pu	4.6	610.	Ni reflector
16	ZPR-6-7(Ref)	Pu	6.5	3100.	L/D=0.9
17	ZPR-6-6A	U	5.0	4000.	L/D=0.8
18	ZPPR-2	Pu	5.1	2400.	L/D=0.5 2-zone equal vol.
19	MZA'	Pu	3.9	570.	
20	MZB	Pu	5.2	1800.	
21	FCA-V-2	Pu+(U)	2.3	200.	Pu/EU=1/3

K_{eff}: effective multiplication factor

In Table-4, calculated K_{eff} by diffusion model is shown. For the Pu-fueled cores the JENDL-3T reduces K_{eff} by 2.0 -- 0.3 % depending on core size compared with that of JENDL-2 with an exception of ZPR-3-56B which is a Ni reflector core. In the latter assembly , significant contribution from the Ni elastic cross section are found to be attributed to this difference. For U-fueled cores the JENDL-3T results increase K_{eff} by 1.3 -- 0.5 % core to cores, 0.8 % in average. As a results, a very large difference in K_{eff}, more than 2 %, is found for JENDL-3T between different fuel types as seen in Table 5.

Table 5. Average K_{eff} for different Fuel types

	Pu-core	U-core	Difference
JENDL-2	0.99800	1.00337	0.5%
JENDL-3T	0.99025	1.01118	2.2%
JENDL-3T(replacing nu data of U-235 to JENDL-2)	0.989	0.989	0.0%

Break down of the components contributing to

Table 4. Comparison of K_{eff} values calculated by JENDL-2 and JENDL-3T and their difference

No.	Assembly	Exp.	JENDL-2	JENDL-3	J3-J2 (%) Pu-core	J3-J2 (%) U-core
1	VERA-11A	1.00000	0.99496	0.97938	-1.60	
2	VERA-1B	1.00000	0.99952	1.00648		+0.69
3	ZPR-3-6F	1.00000	1.01285	1.02642		+1.32
4	ZEBRA-3	1.00000	0.99977	0.97954	-2.06	
5	ZPR-3-12	1.00000	1.00630	1.01448		+0.81
6	SNEAK-7A	1.00000	1.00578	0.99582	-1.06	
7	ZPR-3-11	1.00000	1.00496	1.01264		+0.76
8	ZPR-3-54	1.00000	0.96373	0.96258	-0.07	
9	ZPR-3-53	1.00000	0.99585	0.99214	-0.38	
10	SNEAK-7B	1.00000	1.00377	0.99200	-1.19	
11	ZPR-3-50	1.00000	1.00025	0.99547	-0.48	
12	ZPR-3-48	1.00000	1.00627	0.99636	-1.00	
13	ZEBRA-2	1.00000	0.99247	0.99775		+0.527
14	ZPR-3-49	1.00000	1.00896	0.99633	-1.27	
15	ZPR-3-56B	1.00000	0.99622	0.99685	+0.06	
16	ZPR-6-7	1.00000	0.99919	0.99301	-0.63	
17	ZPR-6-6A	1.00000	1.00408	1.00915		+0.509
18	ZPPR-2	1.00000	1.00569	0.99866	-0.71	
19	MZA	1.01080	0.99982	0.99303	-0.68	
20	MZB(1)	1.00400	0.99651	0.99212	-0.45	
21	FCA-5-2	1.00000	0.99277	0.98993	-0.29	
	average		0.99953	0.00623	-0.783	+0.773

this K_{eff} difference between JENDL-3T and -2 are shown in Table 6. From this table, nu value of U-235 in JENDL-3T has large contribution to the difference between U-fueled and Pu-fueled cores. Too high values are assigned for JENDL-3T. If nu values of U-235 of JENDL-2 data are used in JENDL-3T, the K_{eff} difference in fuel type will be completely removed as seen in the last line of Table 5. No systematic difference between Pu and U fueled core is resulted, and only 1 % less reactive profile is observed.

Table 6. JENDL-3T U-235 and Pu-239 primary reaction data contribution to the K_{eff}

U-235 (for U-core)		Pu-239 (for Pu-core)	
nu ν	+2.19 %	nu ν	+0.72 %
chi χ	+0.35	chi χ	+0.42
fission σ	-1.58	fission σ	-2.19
sum	+0.96 %	sum	-1.06 %

n.b. standard: JENDL-2

Also from the table 6, a relatively large effect is observed for Madland Nix's fission spectrum data. In average 0.4 % increase is attributed to this data. This data also contribute to the over estimation of the spectral indices for the threshold reaction. In this connection, re-evaluation of the fission spectrum is pointed out.

Contribution from U-238 inelastic cross section is also analyzed, because so large differences are seen between the evaluations of JENDL-2 and -3T. In the table 7, effects to the spectral indices are also shown as well as K_{eff}. A very large contribution is observed for 2--5 MeV energy range. This cross section data is not fixed yet due to the difficulty of measurements, only calculation is the main evaluation methods. There still remain a very big ambiguity in the cross sections and as a results for the K_{eff}. From this table, K_{eff} and threshold reaction spectral indices are trade off relations. There is no solution to fill the request both for the K_{eff} and threshold reaction spectral indices.

Table 7. JENDL-3T U-238 inelastic cross section sensitivity

20% inelastic cross section(MT=4) increase for each energy group

Energy	group	K_{eff}	F28/F25	F40/F25
10-5MeV	1--3	-0.05%	-0.4%	-0.1%
5-3MeV	4--5	-0.13%	-1.3%	-0.5%
3-2MeV	6--7	-0.25%	-2.6%	-1.0%
2-0.6MeV	8-11	+0.05%	-1.4%	-2.0%

n.b. standard data: JENDL-3T

Central reaction rate ratio

The reaction rate ratios calculated by JENDL-3T and JENDL-2 are compared in Table 8 together with ENDF/B-4.

The ratio of U-238 capture to Pu-239 fission rate, C28/F49, which is an important parameter for the estimation of the breeding ratio, is even increased by 4% for Pu-fueled cores by JENDL-3T. In JENDL-2, this value is already overestimated by about 6 % for relatively large cores, therefore 10 % over-estimation is resulted for JENDL-3T. It is too large for the design margin of the FBR.

The spectral indices for the threshold reaction such as F28/F25 (U-238 fission to U-235 fission), F40/F25 (Pu-240 fission to U-235 fission) are also worse for JENDL-3T than that of JENDL-2. Over-estimation of C/E is enhanced by about 6%. Sensitivity analysis revealed that this is due to the Madland and Nix's hard fission spectra for U-235 and Pu-239. A question of the adoption of these fission spectra in JENDL-3T is raised.

Table 8. C/E statistics for the primary integral data of central nuclear characteristics

	JENDL-1	JENDL-2	JENDL-3T	ENDF/B-IV
Keff				
Pu core(15)	1.001(0.011)	0.998(0.010)	0.990(0.009)	----
U core(6)	1.007(0.010)	1.003(0.006)	1.011(0.009)	----
All core(21)	1.003(0.011)	1.000(0.010)	0.996(0.013)	0.997
Central reaction rate ratio				
F28/F25	1.00 (0.08)	1.03 (0.09)	1.11 (0.10)	1.04
F49/F25	0.97 (0.04)	0.98 (0.02)	0.99 (0.02)	0.99
F40/F25	1.01 (0.11)	1.07 (0.12)	1.12 (0.14)	1.08
C28/F25	0.98 (0.03)	0.96 (0.07)	1.01 (0.07)	0.97
C28/F49	1.01 (0.05)	0.99 (0.05)	1.03 (0.06)	0.98
Central material worth				
U-235	1.03 (0.06)	1.01 (0.06)	1.04 (0.06)	1.01 (0.06)
U-238	1.10 (0.20)	1.00 (0.29)	1.03 (0.28)	0.95 (0.13)
B-10	0.95 (0.11)	0.85 (0.13)	0.88 (0.14)	0.84 (0.12)
Cr	0.95 (0.17)	1.07 (0.11)	1.12 (0.12)	1.36 (0.21)
Fe	0.88 (0.10)	1.13 (0.30)	1.19 (0.29)	1.11 (0.28)
Ni	1.12 (0.20)	1.24 (0.15)	1.17 (0.20)	1.17 (0.20)
Mn	1.25 (0.69)	1.15 (0.31)		

( ) : STD

Central sample reactivity worth

The C/E of the central material worth is also given in Table 8, the data are normalized to the worth of Pu-239 so as to remove the problems of the reactivity scale between the different assemblies. Over-estimation of sample worth for Mn which is well-known in JENDL-2 is corrected. However, improvements for other major nuclides are not attained for JENDL-3T.

Benchmark test with two-dimensional model

In order to simulate more sophisticated real FBR core emphasizing on the engineering aspects, two-dimensional benchmark test was performed using the fast critical assemblies FCA-VI-2 and ZPPR-9. The Doppler reactivity worth, sodium void reactivity worth and reaction rate distribution were calculated.

The UO_2 Doppler worth is increased by about 6% for JENDL-3T and approaches to the experimental value, as seen in Table 9. Improvement by JENDL-3T is significant.

Table 9. C/E of ZPPR-9 Natural UO_2 Doppler Reactivity worth

Temperature	JENDL-2	JENDL-3T
298 K -- 478 K	0.879	0.938
298 K -- 1087 K	0.888	0.951

The overestimation of sodium void reactivity coefficient which has been one of the drawbacks in JENDL-2 is cured remarkably in JENDL-3T. C/E values for the progressive void coefficients in ZPPR-9 are shown in Table 10. This is partly due to the improvements of Pu-239 fission cross sections around 1 keV in JENDL-3T.

The reaction rate distribution in ZPPR-9 is improved by JENDL-3T a little. The overestimation of the distribution in the outer core is improved a little (1%). For the reaction rate distribution, space dependence of C/E observed in JENDL-2 has been one of the problems for the FBR designers.

Space dependence of C/E values for control rod worth observed in JENDL-2 by the JUPITER(Japanese-United States Program of Integral Tests and Experimental Researches) analyses /3/ has been the serious problems for the safe and economical design for the demonstration FBR. Unfortunately this integral experiment is not yet analyzed for JENDL-3T. But from the results of reaction rate distribution, the possibility for the improvement is quite confident.

Table 10. C/E of ZPPR-9 Sodium-void Reactivity Coefficient

Void region	JENDL-2	JENDL-3T
9 drawers x 40.6 h	1.035	0.846
37 drawers x 40.6 h	1.096	0.895
97 drawers x 40.6 h	1.093	0.891
97 drawers x 81.28h	1.171	0.927
97 drawers x 101.60h	1.266	0.972
97 drawers x 137.16h	1.431	1.070

Recommendation to the JENDL-3T from fission reactor benchmarks

From these benchmark tests, reaction data asked for the re-evaluation are as follows:

U-235: nu value, fission cross-section, fission spectrum,

Pu-239: fission cross section, fission spectrum,

U-238: inelastic scattering cross-section, capture cross section,

Pu-240: capture and inelastic scattering cross section,

Al,Ni: capture and elastic scattering cross section.

Shielding Benchmark Test

Shielding design accuracy depends on the accuracy of nuclear data and adequacy of the calculational method used. Especially for shielding benchmark tests, these two items are deeply interrelated each other. Only for the cases in which these two items are sufficient, we can conclude that the agreement is quite well.

Shielding application covers a very wide range of nuclear industries such as fuel processing/reprocessing facility, transport cask of spent fuel, accelerator, space shuttle,..., as well as power reactors of thermal, fast or fusion. There are no application fields without having the shielding equipments in nuclear related industries. For the shielding materials, a variety of substances are used, therefore a number of nuclides for which shielding characters should be identified are born.

Rather different aspects are important for the shielding calculations compared with that of criticality calculations. In the criticality calculation, reaction balance is the most important in the calculation, i.e., key points are how accurately reaction balance is estimated by the calculation. Peak value and its shape of the resonance cross section has large contribution to the reaction balance in the core calculation thus self-shielding factor has important aspects. On the other hand in the shielding calculation, how accurately source is estimated and how accurately scattering is estimated are the key points. Therefore total cross section and scattering cross section play an important role. The shape of the valley between the resonances, i.e., cross section minima, has very important aspect. This is completely contrary to the case of criticality calculation.

Integral test of the cross section in JENDL-3T for the shielding application have been made for the following items listed in Table-11. The selected problems are picked up by the reason for which we can identify the problem in the cross section data and feedback these information to the evaluators. They are a good geometry experiments of single materials, penetration experiments for the single layers, leakage spectrum measurements for fusion applications and secondary gamma-ray production cross section check experiments.

Table 11. Adopted benchmark test for the shielding calculation

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A. Total cross section check by Broomstick's experiment
- Oxygen (152.4cm) /27/
- Nitrogen (91.44cm) /28/
- Sodium (60.6 cm) /29/
- Iron (20.3 & 30.5 cm) /30/
- SUS310 (20.3 cm) /31/

B. Fe (& SUS) data
- ASPIS Fe deep penetration experiment (140cm)
  source: fission neutron /32/
- KfK leakage spectrum measurement (15 -40 cm)
  source: Cf-252 /33/
- ORNL neutron transmission experiment(30-90cm)
  source: TSF uncollided beam /34/

C. Na data
- ORNL TSF sodium deep penetration experiment
  2.5 ft. - 15 ft. /35/
- ISPRA EURACOS-II deep penetration experiment
  400 cm /36/

D. Carbon data
- ASPIS Graphite shielding experiment /37/
- Profio experiment by RPI /38/

E. Others: Light nuclides and structural materials relating for fusion neutronics
- HANSEN experiment (Li-6, Li-7, C, O, Fe) /39/
- Linac TOF leakage spectrum experiment by KURRI (Li, Fe, Cr, Ni, SUS) /40/

F. Fe secondary gamma-ray production cross section data
- SUS-304 ORNL 14MeV neutron penetration experiment /41/

Benchmark Calculations

In this benchmark test, for comparisons ENDF/B-4, JENDL-2 and JENDL-3PR1 are also used as well as JENDL-3T. The analyses are performed by several neutron transport codes of Sn and Monte Carlo, such as ANISN-Jr(1-D) /42/, DIAC(1-D) /43/, DOT3.5(2-D) /44/, MORSE-CG /45/ with more than 100 energy calculations. Group constants are prepared by PROF-GROUCH-G/B /46/ or RADHEAT-V4 /47/. In some analyses we use the JSSTDL library system /48/, which is a new library system for the sake of common use in the shielding calculation, it enables the user to provide proper group structure library he needs starting from the JSSTDL-295 common library.

Because not enough spaces are allowed, first two benchmark tests in the Table 11 are described here. Only the problems revealed from these benchmark tests are shown. In consequence worse results are mainly shown for JENDL-3T, but we stresses here that the general agreements are quite well for JENDL-3T.

As to the detailed discussions for Na and Fe data, please refer also to the paper by M.KAWAI et al., of NAIG /49/ in this conference.

Total cross section check in MeV range by Broomstick experiment

This series experiments are uncollided spectrum measurements performed by Straker /27-31/ using TSF-II reactor at ORNL to investigate minima in the total cross section in MeV range (1 MeV -- 10 MeV) for the typical shielding materials, O, N, Na, Fe, SUS310.

This experiment is performed in good geometry, no transport calculation is needed for the analysis. Calculation proceeds firstly to determine a transmitted uncollided spectrum and next to fold the so calculated spectrum with the resolution function of NE-213.

Results and discussion

Overall results in C/E basis are given in Table 12.

Oxygen: C/E profile is given in Fig.4. Agreements are better for ENDF/B-4. JENDL-3T results are systematically lower than that of ENDF/B-4 by 6-7 %. About 5% change of the cross-section will compensate the gap as seen in the Fig. 4.

Nitrogen: Both of the evaluations are comparable.
Sodium: C/E is over-estimated by about 20% for all files as seen in Fig.5. There are no difference between JENDL-2 and -3T. Clear difference is

Table 12. Statistics of C/E values for broomstick's experiment

Oxygen	thickness:	154.2 cm	
		C/E (std.)	
	JENDL-3T	0.686 (0.218)	
	ENDF/B-4	0.756 (0.213)	
	(energy range:	2MeV -- 8MeV)	
Nitrogen	thickness:	91.44 cm	
	JENDL-3T	1.134 (0.297)	
	ENDF/B-4	1.138 (0.298)	
	(energy range:	800keV -10MeV)	
Sodium	thickness:	60.6 cm	
	JENDL-3T	1.243 (0.189)	
	JENDL-2	1.242 (0.188)	
	ENDF/B-4	1.217 (0.215)	
	(energy range:	800keV -11MeV)	
Iron	thickness:	20.3 cm	30.5 cm
	JENDL-3T	0.958 (0.160)	0.937 (0.254)
	JENDL-3P1	0.959 (0.163)	0.935 (0.255)
	JENDL-2	0.959 (0.163)	0.938 (0.254)
	ENDF/B-4	0.954 (0.086)	0.986 (0.176)
	(energy range:	1.2MeV -11MeV	1.MeV - 8MeV
SUS310	thickness:	20.3 cm	
	JENDL-3T	1.042 (0.276)	
	JENDL-3P1	1.104 (0.308)	
	ENDF/B-4	1.020 (0.254)	
	(energy range:	1.2MeV -11MeV)	

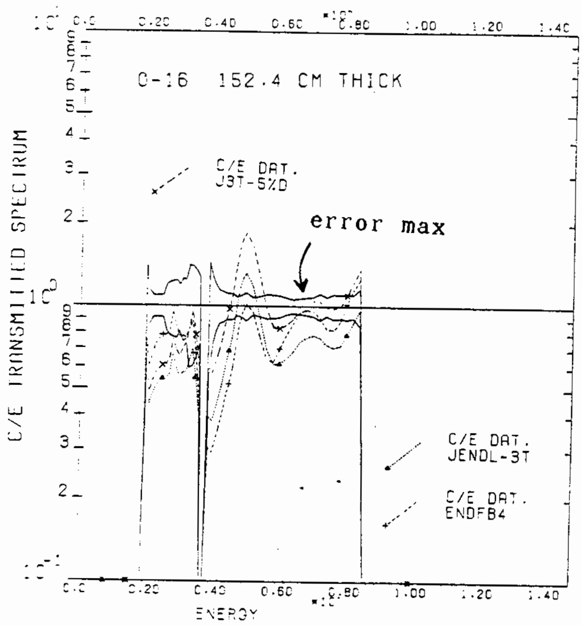

Fig. 4 C/E profile for O-16 transmitted spectrum

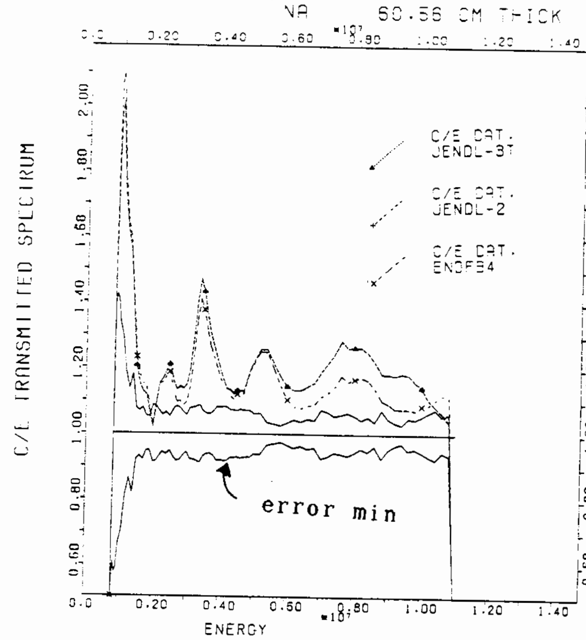

Fig. 5 C/E profile for Na-23 transmitted spectrum

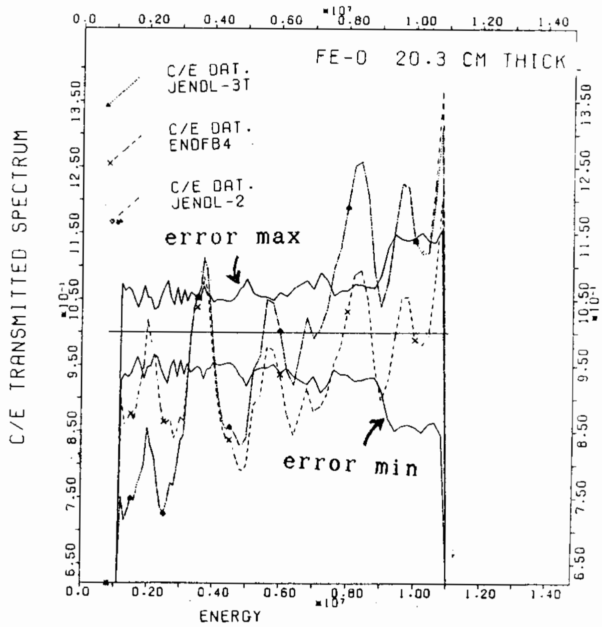

Fig.6 C/E profile for Fe transmitted spectrum

observed between JENDL and ENDF/B-4 in the energy range from 6 MeV to 10 MeV, where ENDF/B-4 is better. By the sensitivity calculation, this gap will be compensated by the increase of total cross section about 5%.

Iron: The average value of C/E is nearly the same among all of the files but the value of standard deviation for ENDF/B-4 is the best, about the half of the others. Fig. 6 shows the transmitted spectra and their C/E. A clear tendency is shown for JENDL-3T, below 3 MeV total cross section of JENDL-3T seems to be over-estimated and high energy range 8-10 MeV under-estimated since the spectrum and the cross sections are inverse proportional relation. Request for the re-evaluation is asked.

SUS310: Rough composition of this material is Fe(51w%),Ni(21w%),Cr(25w%),Si,Mn,C(rest). Overall agreements are better for ENDF/B-4. For JENDL-3T below 3MeV under-estimation of spectrum is found, the same as Fe case. The difference between JENDL-3T and 3PR1 comes from the total cross section of Cr and Ni, since there is no difference between the two for Fe data. Those nuclides of JENDL-3T are found to be improved.

From this analysis, total cross section in MeV range for these nuclides in JENDL-3T is not superior to JENDL-2 or ENDF/B-4 yet. Re-evaluation work is requested.

Natural iron cross section assessment through ASPIS deep penetration experiment

Iron is one of the typical shielding materials for general use. This experiment was selected because it was designed to provide information of benchmark quality for testing of data and calculational methods for deep penetration profile by natural iron shielding material.

Experimental configuration is as follows: a low-power natural converter plate, driven by the source reactor NESTOR, provided a large thin disk sources of fission neutrons is placed at the interface of a graphite moderator and extensive iron shield (=140 cm thickness) as shown in Fig. 7.

Calculation was performed by DOT3.5 with S-48 P_5 R-Z model(53X92 meshes) using 120 group library (BERMUDA 120 group structure)/50/. For natural iron cross sections, fully shielded data including higher Pl scattering matrices ($1/\sigma_t$ weight, σ_0=0.) were generated by PROF-GROUCH-G/B and used.

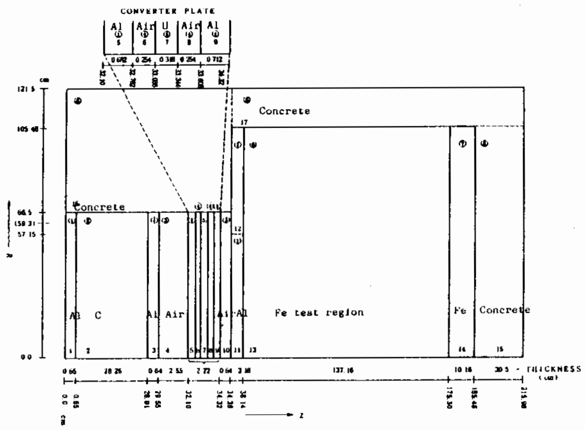

Fig. 7 Analytical model for ASPIS deep penetration experiment

Results and discussion

Axial attenuation profile is shown in Fig. 8. In general all results show good agreements.

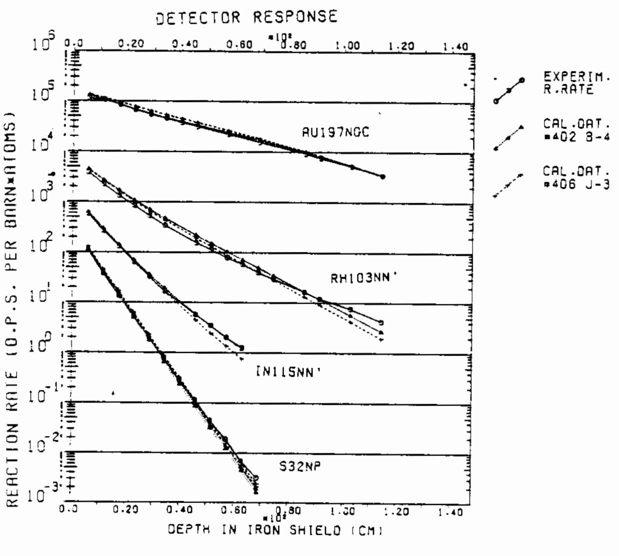

Fig. 8 Comparison of axial attenuation profile for several detector responses

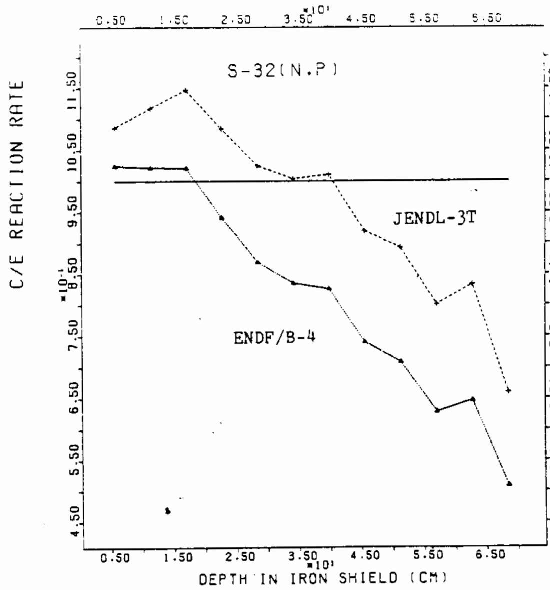

Fig. 9 Comparison of C/E profile for S-32(n,p) axial attenuation profile

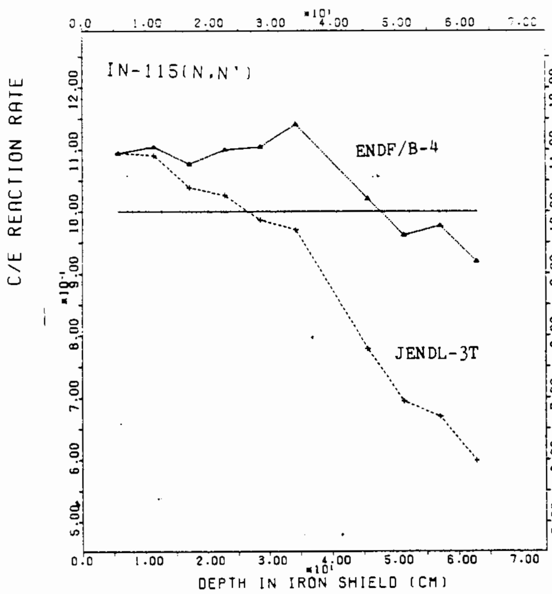

Fig. 10 Comparison of C/E profile for In-115(n,n') axial attenuation profile

The C/E profile for S-32(n,p), In-115(n,n') detectors are shown in Fig.9 and 10. The results for the spectrometer measurements at 86cm depth are shown in Fig. 11 and 12, in the latter figure C/E profile is plotted. From these figures, agreements of the fluxes after 24 keV resonances is quite good for JENDL-3T. The evaluation of this minima in JENDL-3T is concluded to be superior to that of ENDF/B-4. The tendency for the under-prediction of the fluxes at MeV range is clearly appeared. This tendency is amplified along with the penetration depth, it is consistent with the S-32(n,p) reaction rate C/E profile as seen in Fig. 9. For all nuclear data files having tested, the inelastic cross section from 2 to 5 MeV seems too large. And also from the detailed analysis of the In-115 reaction rate and spectrum profile, for the energy range from 600 keV to 1.2 MeV, some problems about the partitioning between elastic and inelastic cross-sections are pointed out.

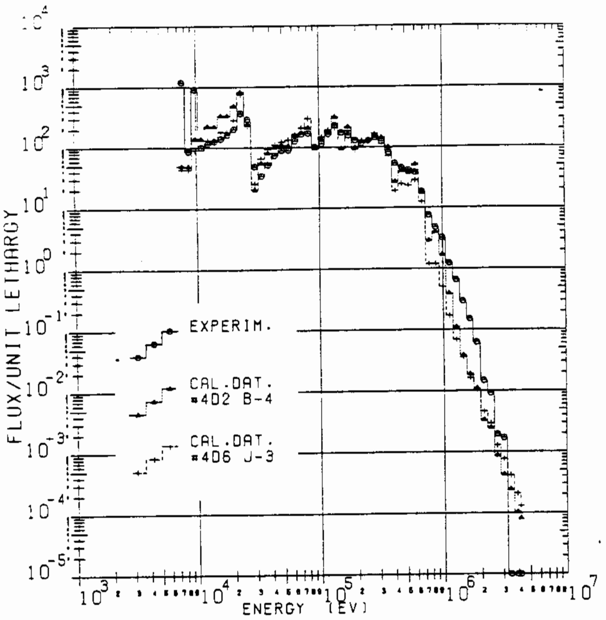

Fig. 11 Comparison of measured spectrum at 87 cm depth in the Iron shield

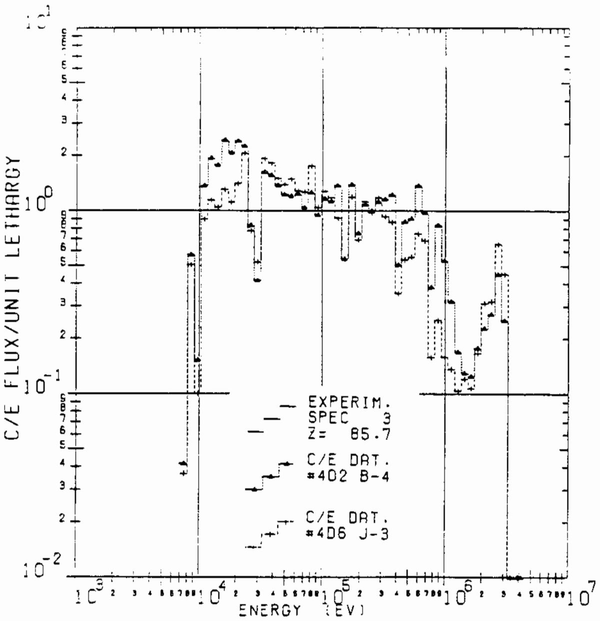

Fig. 12 C/E profile for measured spectrum at 87 cm depth in the Iron shield

Conclusion

Applicability of JENDL-3T nuclear data file to fission reactors and shielding calculations is confirmed through this wide range of benchmark tests performed by JNDC. Overall predictability of JENDL-3T showed satisfactory, however, some problems are also pointed out.

A schedule is foreseen for the release of the latest version of the general purpose Japanese Evaluated Nuclear Data Library within a several months for the sake of so wide needs to the fields even in the medicine or new material researches as well as nuclear energy technology for fission or fusion reactors.

By the reflection of thus obtained feedback information to the applicability of JENDL-3T, forth coming official version, JENDL-3, will be greatly improved.

Acknowledgment

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