11. Status of Standard Cross Section Library and Future Plan

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Abstract

JSSTDL-300 multi-group cross section library with 300 neutron energy groups coupled with 104 group γ-ray cross sections was developed for general users in nuclear reactor physics and/or design, whose source data is the evaluated nuclear data library JENDL-3.2. For the purpose of a standard or common use, several famous cross section libraries worldwide used, i.e., ABBN-25, GAM-123, VITAMIN-C/J(E+C), MGCL-137, BERMUDA-12 and FNS-125 for neutron, and LANL-12,-24-,-48, and CSEWG-94 for γ-ray, are consulted about setting the common energy group structure. Furthermore, in order to expand the applicability, the top energy is set on 20 MeV and the lowest energy is 10^{-5} eV. In the thermal neutron energy region, the JSSTDL-300 has about 20 energy groups. Besides, many utility codes for group collapsing and for data format transformation are provided for general users.

1 Introduction

Many cross section libraries based on different evaluated nuclear data files have been used in Japan for reactor physics and nuclear designs. In such a situation, evaluations of their reliabilities from nuclear data point of view are difficult and confused since the discrepancies of calculated and experimental integral data like criticality k_{eff} come from the errors of nuclear data themselves and also calculational model. Especially, the inter-comparisons or verification of the core performance parameters given by some designers cannot be easily performed.

Need of the standard multi-group cross section library has been requested so far but its realization has never been accomplished. However, some multi-group and Monte Calro libraries prepared by JAERI have been used like a standard cross section libraries, which are mentioned in Section 2 in more detail and frequently referred by many users in comparison with their own libraries. Users are obliged to be interpreted with extra labour since the energy group structures of the multigoup cross section libraries, even among the JAERI's libraries, are usually different each other depending on their targeted cores, e.g., LWR or FBR core.

The Committee of Group Constants of JNDC had recommended that "JNDC should supply commonly usable group cross-section library for primary users in the fast or fusion reactor design groups following the JENDL-3 release" in 1987[1]. The Standard Cross Section Library Working Group (SCSLWG) in Japan has been encouraged by the recommendation and leaned forward the first version of the library. According to the study on the need of

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standard libraries, it was found that there hardly existed any cross section library to be available for shielding calculation as a standard library. Therefore, at the first step, the Japan Shielding Standard Cross Section Library (JSSTDL) was produced, whose constants and specifications are shown in Section 2 and will be found in the following Home Page in detailed,

(http://ndc.tokai.jaeri.go.jp/ftpnd/jendl/j32p300K.html)

As the second step, the study of future plan for standard cross section library following the first version is in progress by the SCSLWG. In this presentation, the activity of the SCSLWG is shown in the Section 2 and the current status of the future plan is reported in Section 4.

2 Activity of Standard Cross Section Working Group

Responding to the recommendations, "Prepare the Standard Cross Section Library", from The Committee of Group Constants of JNDC, the SCSLWG has sent out questionnaires to all members to pick up demands from reactor design and/or shielding design groups regarding specifications of the common group constants.

The first draft of JSSTDL-300 document has been prepared and will be published in a few months as a JAERI-report. Consequently, a brief review of the JSSTDL-300 specifications and nuclear data processing is shown below.

2.1 Nuclear Data Processing

The 67 nuclides as a whole were processed from JENDL-3.2 general-purpose file and stored for JSSTDL system. Gamma-ray data are furnished for 40 nuclides out of the 67 nuclides. They cover almost all nuclides having gamma production data in the general-purpose file. The nuclides compiled in the JSSTDL-300 library are shown in **Table 1** for neutron and gamma data, respectively.

(a) Pre-processing

The "pre-processing" means to generate temperature dependent point-wise cross-sections from the original nuclear data file JENDL-3.2 compiled in ENDF/B-5 Format. In the first step to represent the data in most compact way, we generate a start up file only including linear-linear interpolation scheme (INT=2) in FILE 3 in comparison with the results based on general interpolation scheme (INT=1 $\sim$ 5) defined in original data, where LINEAR code was adopted. In the next step we generate an energy dependent point-wise cross section at $0\,^0K$ using RECENTJ or RESENDD for the nuclides having resonance parameters in the JENDL-3.2 file. Complete FILE3 data are given here. At this step, the unresolved resonance data are also written to FILE2 section for the later processing by PROF. GROUCH-G/B in the effective cross section calculation step. So many data points are generated in this step, i.e., more than 100000 points per a reaction are not rare. We usually produce cross sections at the condition that their relative error is less than 0.1%.

In the next step, we generate temperature dependent cross sections by SIGMA1 code[2] using Kernel Methods in Doppler broadening. In this step, generated cross section points will be significantly reduced by Doppler broadening. This step is repeated until all of the necessary temperatures are processed.

(b) Processing

Table 1: **Nuclides of JSSTDL-300 Neutron and Photon Libraries**

Neutron Library						Photon Library			
JSSTDL300.LIB1A		JSSTDL300.LIB2A		JSSTDL300.LIB3A		JSSTDL104.LIB1A			
Nuclide	ID.No.	Nuclide	ID.No.	Nuclide	ID.No.	Nuclide	ID.No.	Nuclide	ID.No.
H-1	110	K	1900	Bi-209	8390	H -1	110	V -51	2310
H-2	120	Ca	2000	Th-232	9020	He-3	230	Cr	2400
He-3	230	Ti	2200	U-233	9230	Li-6	360	Mn-55	2550
He-4	240	V-51	2310	U-234	9240	Li-7	370	Fe	2600
Li-6	360	Cr	2400	(U-235)	9250	Be-9	490	Co-59	2790
Li-7	370	Mn-55	2550	U-236	9260	B -10	500	Ni	2800
Be-9	490	Fe	2600	(U-238)	9280	B-11	510	Cu	2900
B-10	500	Co-59	2790	Np-237	9370	C-12	620	Zr	4000
B-11	510	Ni	2800	Np-239	9390	N-14	740	Nb-93	4130
C-12	620	Cu	2900	Pu-236	9460	O -16	860	Mo	4200
N-14	740	Ga	3100	Pu-238	9480	F-19	990	Cd	4800
O-16	860	Ge	3200	(Pu-239)	9490	Na-23	1130	Eu	6300
F-19	990	Zr	4000	(Pu-240)	9400	Mg	1200	Hf	7200
Na-23	1130	Nb-93	4130	Pu-241	9410	Al-27	1370	Ta-181	7310
Mg	1200	Mo	4200	Pu-242	9420	Si	1400	W	7400
Al-27	1370	Cd	4800	Am-241	9510	P-31	1510	Pb	8200
Si	1400	Eu	6300	Am-242m	9520	S	1600	Bi-209	8390
P-31	1510	Hf	7200	Am-243	9530	K	1900	U-235	9250
S	1600	Ta-181	7310	Cm-242	9620	Ca	2000	U-238	9280
Cl	1700	W	7400	Cm-243	9630	Ti	2200	Pu-239	9490
Ar-40	1800	Pb	8200	Cm-244	9640				
(U-235)†	9250	(U-235)	9250	Cm-245	9650				
(U-238)	9280	(U-238)	9280	Cm-246	9660				
(Pu-239)	9490	(Pu-239)	9490	Cm-247	9670				
(Pu-240)	9400	(Pu-240)	9400	Cm-248	9680				

†: Nuclide in parentheses means the same data containes in each file.

Table 2: **File allocation in Nuclear Data Processing**

File No.	Processing
FILE1	Prompt neutron yield ν and fission spectrum χ, and those for delayed neutron,
FILE2	Unresolved range processwed by UNRESR module (NJOY),
FILE3	Resolved resonance and smooth parts,
FILE4	Energy transfer matrices in P_l order and average P_l coefficients for scattering reactions,
FILE5	Energy transfer matrices in P_0 order for scattering or fission cross sections,
FILE6	Energy transfer matrices in P_l order.

PROF GROUCH-G/B code was used to generate group cross-sections, scattering matrices and self-shielding factors. This code is designed for the generation of group constants with any weight specified by the user. Processing is made for the following quantities for each FILE categories as shown in **Table 2**.

(c) Post processing

This step is to make the group cross section library JSSTDL-300 gathering huge output data made by previous processing system PROF. GROUCH-G/B, which is composed of 2 steps. The first step GLIBMK code is used to produce nuclide dependent binary working library for each processed nuclides. The next step is JLIBGB for neutrons, which is an editing routine to produce JSSTDL-300 library from the working library previously produced. For gamma library, NGTOJL calculates the multi-group γ-ray cross sections from data file XCOM[3].

The 75 nuclides from ^{1}H to ^{248}Cm are assigned for the neutron library as the three separate files in which main fuel isotopes ^{235}U, ^{238}U, ^{239}Pu and ^{240}Pu are contained in each file, and the 20 nuclides for photon library as shown **Table 1**.

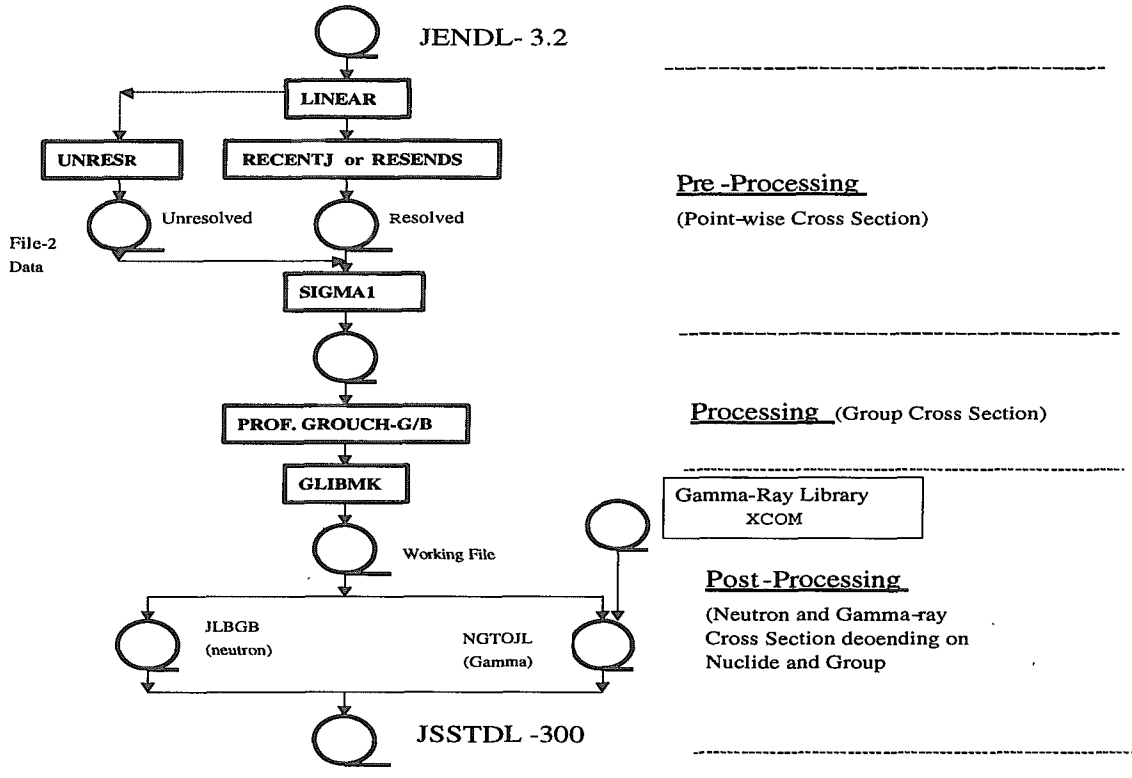

Figure 1: Nuclear Data Processing for JSSTDL-300 Library Production.

2.2 Energy Group Structure

The energy group structures of the JDDTSL-300 were determined by consulting with the group structures of SD-100(100 groups), VERMUDA-121(121), FNS-125(125), VITAMIN-C(171), VITAMIN-J(E+C)(175), GICX-42(42), ABBN-25(25), JFS-New(70), GAM-123(92), MGCL-137(91), WIMS-69(28), CSEWG-94(94), LANL-12(12), LANL-48(48), LANL-24(24), STEINER-21(21), STRAKER-22(22), BERMUDA-36(36) and HONEYCOMB-15(15). These group boundaries are included in the JSSTDL-300 group structure. Consequently, the neutron energy group widths (Δu's) are inhomogeneous as shown in **Fig. 1**. The number of photon energy groups is 104 from 1 keV to 50 MeV.

In order to investigate general feature of JSSTDL-300 energy group structure, it is superimposed on the typical Fast Breeder Reactor (FBR) Neutron spectrum emphasizing the individual resonance positions of constituent materials for the FBR. The group structure used for FBR spectrum calculation by $MC^2 - 2$ code[4] are 2040 groups with constant lethargy width $\Delta u = \frac{1}{120}$.

As shown in **Fig. 2**, the finer groups in the high energy region above about 20 keV are taken in the JSSTDL-300 and relatively coarse groups are in the fuel material resonance region below about 20 keV. The cross and dotted lines (at the same position) show the group boundaries of JSSTDL-300 Library and indicates the so-called group densities.

The finer group structure can be found in the resonance regions of structure materials such as ^{56}Fe as shown in **Fig. 2**. It implies that the JSSTDL-300 Library emphasizes the resonance effects of the structure materials.

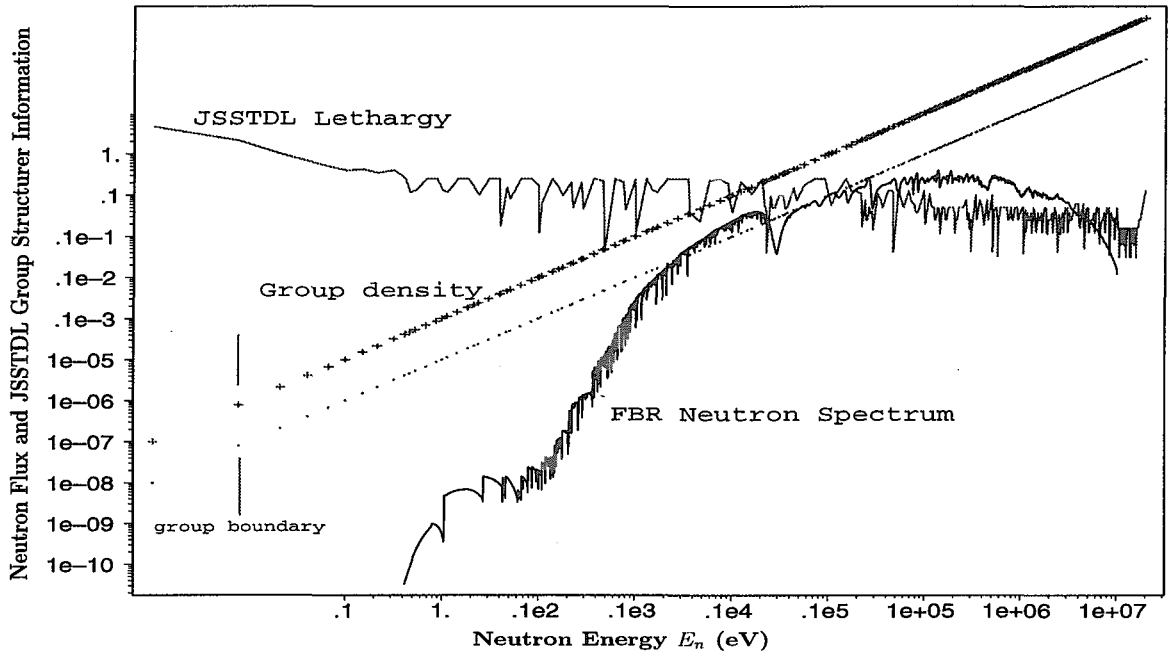

Figure 2: JSSTDL Energy Group Structure in comparison with FBR Neutron Spectrum

2.3 Resonance Self-Shielding Factor

The Bondarenko type resonance self-shielding factors were compiled with nine background cross-sections and four temperatures, i.e., 0, 0.17783, 1, 10, 100, 1000, 10000, 100000 and 1.0×10^6 for infinite dilution) in barns and at 300, 600, 900 and 2100^0K's. The scattering anisotropy was considered up to five-order Legendre polynomials. Fission neutron energy spectra contained in the library are for isotopes ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu.

2.4 Verification of Cross Sections

We compared the multi-group cross-sections collapsed from the point-wise data of JENDL-3.2 with those of the JSSTDL-300 at 300 ^0K. The point-wise data were collapsed into the same 300 energy group structure of JSSTDL-300 with the same weighting function by using CRECTJ5 code. For instance, ^{232}Th(n,γ) cross section is shown in **Fig. 3**. The JSSTDL-300 cross section is slightly different from point-wise data in the energy range above 11.5 MeV where the point-wise cross section is zero. However, we assumed very small cross-sections because PROF-GROUCH-G/B cannot treat zero value.

2.5 Benchmark Tests

Some shielding benchmark tests, e.g., a simple KfK benchmark and JASPER benchmarks, were made to verify accuracy and applicability of the JSSTDL-300 library. For instance, the Karlsruhe Neutron Spectrum Measurement from iron sphere was selected as a simple benchmark test where a typical thin transmission of neutrons can be validated. In this experiment, leakage neutron spectrum was measured for iron sphere of 15, 20, 25, 30,

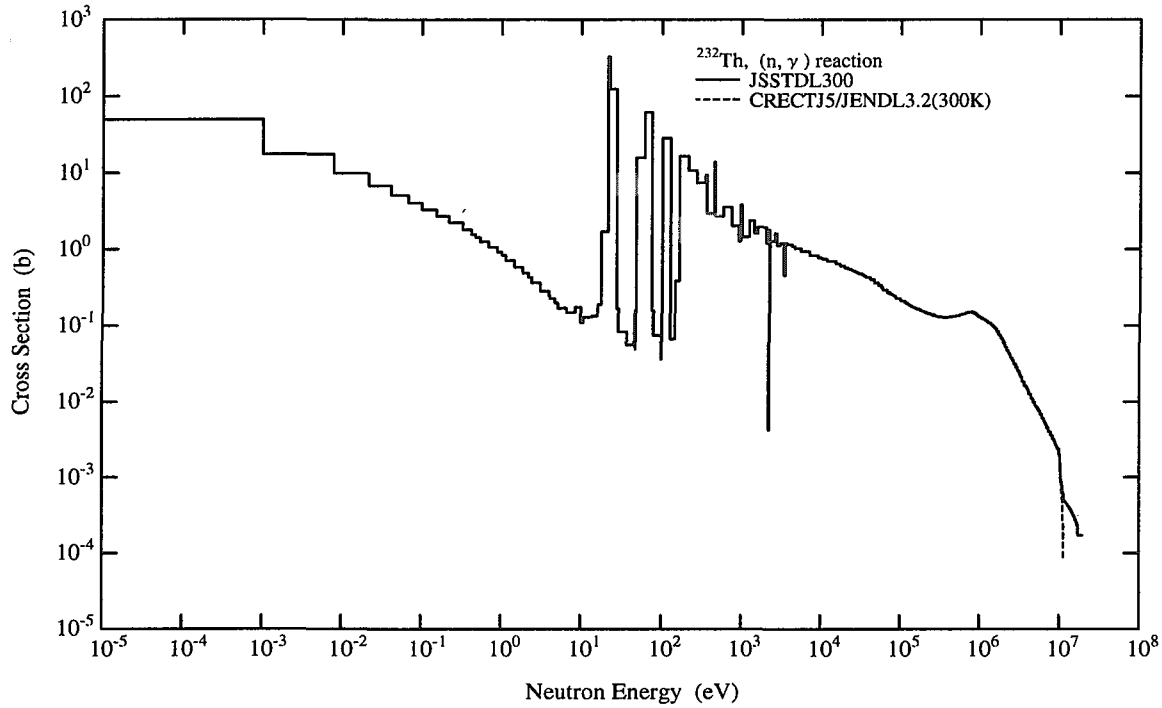

Figure 3: Comparison of ^{232}Th(n,γ) Cross Section of JSSTDL-300's and Point-wise Data

35 and 40 cm in diameter from a ^{252}Cf source in the center. The penetration length for the system corresponds to between 2 and 5 mean-free-path (mfp). The experimental configuration is suitable to examine inelastic reactions of iron. In the present study, a case of 15 cm iron sphere is adopted.

As shown in **Fig. 5**, the calculated leakage neutron current from KFK Sphere Geome-

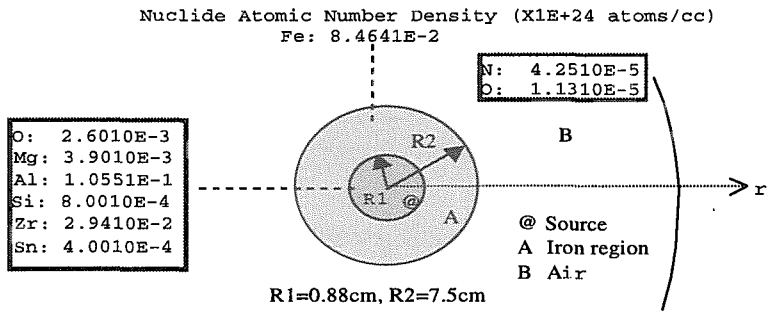

Figure 4: Calculational Model for KFK Iron Benchmark Problem.

try is in good agreement with the experimental one except below about 10 keV where the sharper resonance effects are found in the calculated values.

3 Current Status of Cross Section Library in Japan

Cross section libraries for neutronics calculation codes developed by JAERI are available. Their functions and libraries are briefly reviewed below.

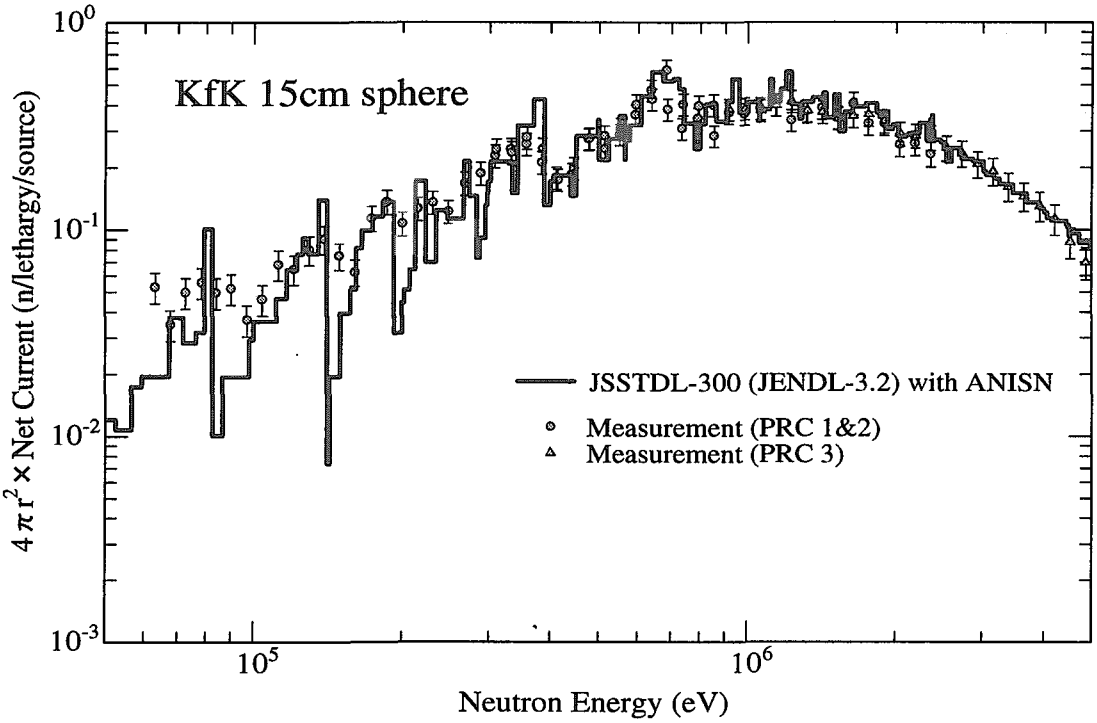

Figure 5: Comparison of Measured and Calculated Neutron Currents from Iron Sphere of KFK Experiment.

JFS-3: Bondarenko type 70 group cross section libraries are provided in the energy range from 0.414 eV to 10 MeV. This library has been used as a standard cross section library for fast reactor physics and core design in Japan so far. The present version is based on the JENDL-3.2. The nuclear data were processed by TIMS/Prof. Grouch-G code system where the effective cross sections in unresolved resonance region were estimated by LADER-method reproducing the evaluated cross section by generating pseudo-resonance parameters. Mutual resonance overlapping effect is expressed by so-called "r-parameter" in the JFS-3.

SRAC: Modular code system SRAC is a general purpose neutronics code system applicable to core analysis of various type reactors and has six kinds of cross section libraries based on the evaluated nuclear data files ENDF/B-IV, -V, -VI, JENDL-2, -3.1 and -3.2. The cross section libraries are provided for the fast energy range from 0.41399 eV to 10 MeV divided by 71 energy groups, and for thermal energy range from 1.0^{-5} eV to 3.9279 eV by 48 energy groups.

The neutronics code system SRAC is integrated by five modular codes such as P_{IJ}-method for 16 types of lattices and S_n-transport calculation modules. In addition to these modular codes, SRAC can calculate superfine group neutron spectrum by using bi-mixture model for slowing-down calculation module PEACO.

MVP: General purpose Monte Carlo code for neutron and photon transport calculations based on continuous energy, which is fully vectorized. MVP point-wise cross section library is produced by LICEM code from evaluated nuclear data file as functions of nuclide and tem-

perature. The point-wise cross section is provided so as to reproduce the evaluated smooth cross section within usually 0.1% error. For unresolved resonance, the point-wise cross section is produced by means of probability-method. The energy range covers the full range from 10^{-5} eV to 20 MeV accepted by JENDL-3.2 as well as ENDF/B-VI.

There are no choice of standard cross section library for such a library as a function of continuous energy since its cross sections are esentially equal to the evaluated nuclear cross sections as a function of temperature.

JSSTDL: General purpose multi-group neutron and photon library based on JENDL-3.2 are available as mentioned in Section 2.

MGCL:General purpose Multi-group Monte Carlo code coupled with 137 group Bondarenko type cross section library based on JENDL-3.2.

MCNP: Cross section library of the general purpose continuous energy Monte Carlo code MCNP is prepared in ACE format by using ACER module of NJOY. This library is essentially equal to the evaluated cross section although the thinning error of union energy grid used in MCNP should be optimized.

4 Study on The Future Plan

4.1 JAERI Updating Plan for The JENDL-3.3

Revision Plans of the public cross section libraries developed by JAERI following the release of the newest version JENDL-3.3 are shown in Table 3. Almost all libraries are continuously updated along with the revision of evaluated nuclear data files.

Table 3: **Current Status of Cross Section Libraries in JAERI**

No.	Code	Purpose	For Benchmark Tests	Version-up by JENDL-3.3	Future plan
1	JFS-3	General	Nuclides for Benchmark Test only will be processed	Will be prepared and released	Considering an extension to high energy file by The Center for Neutron Science. Changes of FORMAT and enlargement of data capacity may be needed. Version up of JFS-3 itself is terminated by the forthcoming JENDL-3.3 release.
2	SRAC	General	No action at present	Production is planed and resultant library released	Continuous correspondence along with JENDL updation
3	MVP	General	No action at present	Production is planed and resultant library released	Continuous correspondence along with JENDL updation
4	JSSTDL	General	Not yet complited.	Production is planed and resultant library released	Continuous correspondence along with JENDL updation
5	MGCL	Criticality Safety	No action at present	Planed but quick responce may be impossible due to modification of soft-ware. The products will be released	Continuous correspondence along with JENDL updation
6	MCNP	Neutron	Library production completed and available	Production is planed and resultant library released	Continuous correspondence along with JENDL updation

4.2 Development of Cross Section Library Production System in JNC

Development of Multi-group cross section library production system in Japan Nuclear Cycle Development Institute(JNC) is in progress for fast breeder reactor as presented in this specialist meeting. NJOY is key processing code supplemented by TIMS-routine in order to precisely estimate the unresolved resonance cross section by means of LADER-method.

4.3 Request for Standard Cross Section Library— Summary of questionnaires

The questionnaires had been sent out to the all members of Standard Cross Section Working Group (SCSLWG) in order to seek for the specifications of future standard cross section library. The summary of questionnaires is shown in Tables 4, 5 and 6

. The standard cross section library should be satisfied the following conditions as (1) documentation describing the nuclear data processing is well furnished, (2) linked to open calculational codes, (3) with enough verification and high reliability, and (4) accepted as a standard method and available as open use.

According to the answers for the questionnaires from working group members, the required standard cross section libraries can be categorized as shown in Table 4;

Number of neutron energy groups may be classified to about three categories as shown

Table 4: **Types of Standard Cross Section Library requested**

Type	Requirement
For fission Reactors:	For detailed LWR and FBR core designs including burn-up calculations: Overall neutron and γ-ray reaction cross sections with resonance self-shielding factors. Furthermore, heating cross section in-cite and DPA cross section are needed. Library should be linked to some target codes. For instance, linked to SRAC, JOINT codes. However, Complete responces to overall users may be impossible.
For Spallation Reaction Neutron Source:	Multi-group cross section library with thermal scattering kernels as function of temperature and molecular and/or crystal structure for the nuclides in the spallation neutron source. For instance, library applicable to MCNP-X and/or S_n-transport calculation codes covering the energy range from 10^{-5} eV to 3.0 GeV

Table 5: **Number of Energy Groups**

Type	Notes
1)	Fine group: $70 \sim 300$ JAERI-Fast and JSSTDL-300
2)	Ultra-fine group: $2000 \sim 3000$ For instance, neutron spectrum calculation by $MC^2 - 2$ code is performed with about 2040 ultra-fine groups with the lethrgy width $\Delta u = 1/120$. As well-known, the standard cross section library in EU has about 1990 groups.
3)	Hyper-fine group: >about 3000 The hyper-fine group like PENDF file of NJOY is in the region of Monte Claro Code and is eqiuvalent to nuclear data themselves depending on temperature.

in Table 5. The JFS-J3 and our JSSTDL-300 belong to the fine group structures. The NJOY PENDF file may be a hyperfine group close to the $MC^2 - 2$ RABLE routine. In general, the finer group structure is flexible for group collapsing to practical multigoup cross section library and can realize the fine structures of neutron spectrum due to resonance effects.

Nuclear data processing codes in public use in Japan are NJOY code and TIMS coupled with Prof. Grouch-G/B, although $MC^2 - 2$ code is partially used by a few users. Recently, NJOY is getting many users but the ladder method for unresolved resonance treatment as in the TIMS code is still preferred as in the case of JNC code system.

The NJOY code is welcome in Japan because of its maintainability and many functions for neutron and/or photon library productions.

Table 6: **Standard Nuclear Data Processing Code**

Case	Code Name	Remarks
(a)	NJOY	(1):Open code and clearly decribed methods, (2):TIMS is not easely handeled, JFS-3-J3.2 → TIMS, (3):NJOY is for general purpose。 (4): NJOY is always guaranteed. (5):and in world wide use, (6):easely handled. (7):Although TIMS/Prof.Grouch-G code system had been developed in Japan and thus more advancement and improvements are wished so as to be updated to NJOY level, man-power maintaining the code system is a probelem.
(b)	NJOY+TIMS	TIMS is for unresolved resonance.

5 Concluding Remarks

The future plan for standard cross section library applicable to many branches; neutron, gamma-ray transport, nuclear transmutation and so on, is not yet established. In prior to detailed specification, extensive requests should be given by general users. The "Standardization" itself is seemed to be worldwide trend but there does not exit strong motivation in Japan at present.

(1)	: Future Plan	Is undergoing a searching the specifications of Standard Cross Section Library.
(2)	: Further Investigation	Users' Needs. ⊙ Feature of Standard Cross Section Lib. ⊙ Exective Group, ⊙ Financial Support.

6 REFERENCES

1) Nakazawa M., et al. : JAERI-M 87-025, p.9 (1987)
2) Koyama K., Yamano N, Minami K. and Miyasaka S. : JAERI-M 7155 (1977)
3) Berger, M.H. and Hubbell,J.H.:"DLC-136/PHOTX V2.0", Available from RSIC, Oak Ridge National Laboratory (1988)
4) $MC^2 - 2$.